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Safety Evaluation Report Related to Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada

Volume 3: Repository Safety After Permanent Closure

Office of Nuclear Material Safety and Safeguards

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NOTE TO READER: In June 2008, the U.S. Department of Energy (DOE) submitted a license application seeking authorization to construct a geologic repository at Yucca Mountain. After docketing the DOE license application, the U.S. Nuclear Regulatory Commission (NRC) staff began documenting its review in a Safety Evaluation Report (SER). In March 2010, DOE filed a motion to withdraw its application before the Atomic Safety and Licensing Board. On September 30, 2010, DOE's Office of Civilian Radioactive Waste Management ceased operations, and assigned its Yucca Mountain-related responsibilities to other offices within DOE. The Atomic Safety and Licensing Board denied DOE's motion to withdraw, and in September 2011, the Commission announced it was evenly divided on whether to overturn or uphold this decision. The Commission directed the Atomic Safety and Licensing Board, in recognition of budgetary limitations, to complete all necessary and appropriate case management activities, and the Atomic Safety and Licensing Board suspended the proceeding on September 30, 2011.

In August 2013, the U.S. Court of Appeals for the District of Columbia Circuit issued a decision granting a *writ of mandamus* and directing the NRC to resume the licensing process for DOE's license application. In November 2013, the Commission directed the NRC staff to complete and issue the SER associated with the license application. Because of the lapse in time and changes within DOE between license application submittal and the issuance of this SER volume, some information in the application does not reflect current circumstances. For example, scientific information continues to be published in areas relevant to the topics considered in the license application. When these situations are relevant to the NRC staff's evaluation of the license application in this volume, the SER identifies and addresses them, as appropriate.

The SER details the NRC staff's review of DOE's license application and supporting information consistent with the NRC regulations and the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa), as supplemented by the Division of High-Level Waste Repository Safety Director's Policy and Procedure Letter 14: Application of YMRP for Review Under Revised Part 63 (NRC 2009ab).

This volume is one of five volumes that comprise the SER. Each volume is to be published separately as it is completed; however, the volume number may not be published in sequence (e.g., Volume 3 is anticipated to be published before Volume 2). The SER volume number and section number within a volume are based on the YMRP. Use of SER section numbers that correspond to the YMRP section numbers facilitated the NRC staff's writing of the SER and allows the reader to easily find the applicable review methods and acceptance criteria within the YMRP. The following table provides the topics and SER sections for each volume.

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Volume 5	**Proposed Conditions on the Construction Authorization and Probable Subjects of License Specifications**	
1	2.5.10.1	Proposed Conditions on the Construction Authorization
2	2.5.10.2	Probable Subjects of License Specifications

References

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

NRC. 2003aa. NUREG–1804, "Yucca Mountain Review Plan—Final Report." Rev. 2. ML032030389. Washington, DC: NRC.

ABSTRACT

Volume 3, Repository Safety After Permanent Closure, of this Safety Evaluation Report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's review and evaluation of the U.S. Department of Energy's (DOE) Safety Analysis Report (SAR), Chapter 2: Repository Safety After Permanent Closure, provided in DOE's June 3, 2008, license application, as updated by DOE on February 19, 2009. In its application, DOE seeks authorization from the Commission to construct a repository at Yucca Mountain. The NRC staff also reviewed information DOE provided in response to the NRC staff's requests for additional information and other information that DOE provided related to the SAR. In particular, SER Volume 3 documents the results of the NRC staff's evaluation to determine whether the proposed repository design complies with the performance objectives and requirements that apply after the repository is permanently closed. The NRC staff finds, with reasonable expectation, that DOE has demonstrated compliance with the NRC regulatory requirements for postclosure safety, including, but not limited to, "Performance objectives for the geologic repository after permanent closure" in 10 CFR 63.113, "Requirements for performance assessment" in 10 CFR 63.114, "Requirements for multiple barriers" in 10 CFR 63.115, and "Postclosure Public Health and Environmental Standards" in 10 CFR Part 63, Subpart L. In particular, the NRC staff finds that the proposed repository at Yucca Mountain (1) is comprised of multiple barriers and (2) based on performance assessment evaluations that are in compliance with applicable regulatory requirements, meets the 10 CFR Part 63, Subpart L limits for individual protection, human intrusion, and separate standards for protection of groundwater.

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EXECUTIVE SUMMARY

1.0 Background

Volume 3, Repository Safety After Permanent Closure, of this Safety Evaluation Report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's review and evaluation of the Safety Analysis Report (SAR), Chapter 2: Repository Safety After Permanent Closure, the U.S. Department of Energy (DOE, or the applicant) provided in its June 3, 2008, license application (LA) submittal (DOE, 2008ab), as updated on February 19, 2009 (DOE, 2009av). The NRC staff also reviewed information DOE provided in response to the NRC staff's request for additional information and other information that DOE provided related to the SAR. In particular, this SER Volume 3 documents the results of the NRC staff's evaluation to determine whether the proposed repository design for Yucca Mountain complies with the performance objectives and requirements that apply after the repository is permanently closed. These performance objectives and requirements can be found in NRC's regulations at 10 CFR Part 63, Subparts E and L. The NRC staff's safety evaluation considers the proposed geologic repository's multiple barriers, both natural and engineered (manmade); and the performance assessments (including model abstractions) used for the individual protection, the separate groundwater protection, and the human intrusion evaluations.

NRC regulations at 10 CFR Part 63 provide site-specific criteria for geologic disposal at Yucca Mountain. Pursuant to 10 CFR Part 63, there are several stages in the licensing process:

- The site characterization stage
- The construction stage
- A period of operations
- Termination of the license

The multi-staged licensing process affords the Commission the flexibility to make decisions in a logical time sequence that accounts for DOE collecting and analyzing additional information over the construction and operational phases of the repository. The period of operations includes (i) the time during which emplacement would occur, (ii) any subsequent period before permanent closure during which the emplaced wastes are retrievable, and (iii) permanent closure. In conducting its review, the NRC staff was guided by the review methods and acceptance criteria outlined in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa), as supplemented by the Division of High-Level Waste Repository Safety Director's Policy and Procedure Letter 14: Application of YMRP for Review Under Revised Part 63 (NRC, 2009ab).

Risk-Informed and Performance-Based Review

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

System Description and Demonstration of Multiple Barriers

NRC regulations at 10 CFR Part 63 require that a geologic repository at Yucca Mountain include multiple barriers, both natural and engineered. Barriers prevent or limit the movement of water or radioactive material. A multiple barrier approach ensures that the overall repository system is robust and not wholly dependent on any single barrier. The NRC requires that DOE identify these barriers when it calculates how the repository will perform. DOE is required to describe the capability of each barrier and provide the technical basis for its description. In its SAR for the proposed repository at Yucca Mountain, DOE identified three barriers: the Upper Natural Barrier, the Engineered Barrier System (EBS), and the Lower Natural Barrier. The Upper Natural Barrier is composed of features above the repository (i.e., topography, surficial soils, and the unsaturated zone) that reduce the quantity and rate of movement of water downward toward the repository, which in turn reduces the rate of movement of water from the radioactive waste in the repository to the accessible environment. The EBS includes different engineering features (e.g., emplacement drifts, drip shields, waste packages and its internal components, and emplacement pallets and inverts) that are designed to (i) enhance the performance of the waste package, preventing radionuclide releases while it is intact; (ii) limit radionuclide releases after the waste package is breached by limiting the amount of water that can contact the waste package; and (iii) limit radionuclide release from the engineered barrier system through sorption processes. The Lower Natural Barrier comprises two features: the unsaturated zone below the repository and the saturated zone, both of which prevent or reduce the rate of radionuclide movement from the repository to the accessible environment through such processes as the slow movement of water and sorption of radionuclides onto mineral surfaces. Each of these barriers includes features that DOE described as important to waste isolation. The NRC staff's review of the multiple barriers is provided in SER Section 2.2.1.1.

Review of Postclosure Total System Performance Assessment

DOE conducted an analysis, through its Total System Performance Assessment (TSPA) computer model, that evaluates the behavior of the high-level waste repository due to the potential release of radionuclides from the repository. The performance assessment provides a method to evaluate the range of features (e.g., geologic rock types, waste package materials), events (e.g., earthquakes, igneous activity), and processes (e.g., corrosion of metal waste packages, sorption of radionuclides onto rock surfaces) that are relevant to the behavior of a repository at Yucca Mountain. The NRC staff reviewed the TSPA analytic models and analyses DOE provided in its SAR.

<u>Scenario Analysis and Event Probability</u>

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

The NRC staff evaluates the applicant's scenario analysis in four separate SER Sections (2.2.1.2.1.3.1 through 2.2.1.2.1.3.4). Section 2.2.1.2.1.3.1 contains the NRC staff's evaluation of both the applicant's methodology to develop a list of FEPs and DOE's list of the FEPs that it considered for inclusion in the performance assessment. In Section 2.2.1.2.1.3.2, the NRC staff evaluates DOE's screening of its list of FEPs, including DOE's technical bases for the exclusion of FEPs from its performance assessment. DOE's formation of scenario classes and the exclusion of specific scenario classes in DOE's performance assessment are evaluated in Sections 2.2.1.2.1.3.3 and 2.2.1.2.1.3.4, respectively.

The NRC staff's evaluation of the applicant's methodology and conclusions on the probability of events included in the performance assessments is addressed in SER Section 2.2.1.2.2. Hence, SER Section 2.2.1.2.2 is aimed at the second of the three risk questions, "How likely is it to happen?" In SAR Section 2.2.2, DOE identified and described those events that exceeded the probability threshold of 1 chance in 100 million per year (10^{-8} per year) of occurring. The NRC staff's evaluation of the applicant's approach for quantifying the event probabilities and the technical basis for determining the probability estimates assigned to each event type with a probability of 10^{-8} per year or higher are evaluated in SER Section 2.2.1.2.2.

Model Abstraction

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

The evaluation of the model abstraction process begins with the review of the repository design and the data characterizing the geology and the performance of the design and proceeds through the development of models used in the performance assessment. The model abstraction review process ends with a review of how the abstracted models are implemented in the TSPA model (e.g., parameter ranges and distributions, integration with model abstractions for other parts of the repository system, representation of spatial and temporal scales, and whether the TSPA model appropriately implements the abstracted model). The NRC staff has separated its model abstraction review into 13 categories that are addressed in SER Sections 2.2.1.3.1 through 2.2.1.3.14. .

Expert Elicitation

Expert elicitation is a formal, structured, and well-documented process for obtaining the judgments of multiple experts on various scientific topics. Pursuant to 10 CFR 63.21(c)(19), DOE must explain how expert elicitation was used in its application. Consistent with YMRP Section 2.5.4, DOE could elect to use the subjective judgments of experts, or groups of experts, to interpret data and address technical issues and inherent uncertainties when assessing the long-term performance of a geologic repository. In its SAR, the applicant used the results of three formal expert elicitations to complement and supplement other sources of scientific and technical information such as data collection, analyses, and experimentation. The NRC staff has reviewed DOE's use of expert elicitation, which includes a technical review of the results of these elicitations.

SER Section 2.5.4 provides the NRC staff's review of the three expert elicitations DOE used in support of its SAR. Expert elicitations were conducted in the areas of seismic hazard (SAR Section 2.2.2.1); igneous activity (SAR Section 1.1.6.2, Section 2.2.2.2, and Section 2.3.11); and saturated zone flow and transport (SAR Section 2.3.9.2).

2.0 Sections of the Postclosure Review

2.1 Multiple Barriers

The NRC staff's review of DOE's System Description and Demonstration of Multiple Barriers is in SER Section 2.2.1.1.

<u>SER Section 2.2.1.1 System Description and Demonstration of Multiple Barriers</u>

This SER section provides the NRC staff's evaluation of DOE's description of the capabilities of the barriers for the repository. A system of multiple barriers is intended to ensure that the repository system is robust and is not wholly dependent on a single barrier. The repository performance objectives in 10 CFR 63.113 require that a geologic repository contain both natural barriers and an engineered barrier system.

The emphasis of the U.S. Nuclear Regulatory Commission (NRC) staff's integrated review of the applicant's performance assessment is not solely focused on the isolated performance of individual barriers, but rather on ensuring that the repository system is robust. The purpose of this SER section is to provide an understanding of how the natural barriers and the engineered barrier system work in combination to enhance the resiliency of the geologic repository. As described previously, DOE identified three barriers: the Upper Natural Barrier, the Engineered Barrier System (EBS), and the Lower Natural Barrier.

The NRC staff has reviewed the SAR and other information submitted in support of the license application and finds, with reasonable expectation, that an engineered barrier system has been designed that, working in combination with natural barriers, satisfies the requirements of 10 CFR 63.113(a) and 10 CFR 63.115(a–c).

2.2 Scenarios in DOE's Total System Performance Assessment

The NRC staff has separated its review of the scenarios used to support DOE's TSPA model in SER Sections 2.2.1.2.1 and 2.2.1.2.2.

<u>SER Section 2.2.1.2.1 Scenario Analysis</u>

This SER section provides the NRC staff's evaluation of the scenario analysis used to support DOE's TSPA model. A scenario analysis is generally composed of four parts (Nuclear Energy Agency, 2001aa). First, a scenario analysis identifies FEPs relevant to the geologic repository system. Second, in a process known as screening, the scenario analysis evaluates and identifies FEPs for exclusion from or inclusion into the performance assessment calculations. Third, included FEPs are considered to form scenarios and scenario classes (i.e., related scenarios) from a reduced set of events. Fourth, the scenario classes are screened for implementation into the TSPA model. Limits on performance assessments are defined in 10 CFR 63.342 including the conditions for exclusion of FEPs on the basis of probability or consequence.

The NRC staff has reviewed the SAR and other information submitted in support of the license application, which includes information required by 10 CFR 63.21(c)(1) and (9), and finds, with reasonable expectation, that relevant requirements of 10 CFR 63.114 and 10 CFR 63.342 are satisfied.

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

This SER section provides the NRC staff's evaluation of information on event probability used to support DOE's TSPA model calculations. The performance assessment used to demonstrate compliance with the individual protection standard for the proposed Yucca Mountain repository must consider events that have at least 1 chance in 100 million per year of occurring. To address this requirement, DOE identified and described those events that exceeded this probability threshold (10^{-8} per year).

The NRC staff reviewed the SAR and other information submitted in support of the license application, which includes information required by 10 CFR 63.21(c)(1) and (9), and finds, with reasonable expectation, that the relevant requirements of 10 CFR 63.114 and 10 CFR 63.342 are satisfied.

2.3 Model Abstractions in DOE's Total System Performance Assessment

The NRC staff has separated its review of the model abstractions used to support DOE's TSPA model in SER Sections 2.2.1.3.1 through 2.2.1.3.14:

SER Section 2.2.1.3.1 Degradation of Engineered Barriers

This SER section provides the NRC staff's evaluation of the chemical degradation of the drip shields and waste packages that would be emplaced in the repository drifts. Chemical degradation is primarily associated with the effect of corrosion processes on the metal surfaces of the drip shields and the waste package outer barriers. The NRC staff's evaluation of the corrosion processes focuses on the following: long-term passive film stability (i.e., passivity), general corrosion, localized corrosion, stress corrosion cracking, early failure, and abstraction and integration of evaluated processes. The drip shields and the waste packages are engineered barriers, a subset of the EBS. The general functions of the EBS are to (i) prevent or significantly reduce the amount of water that contacts the waste, (ii) prevent or significantly reduce the rate at which radionuclides are released from the waste, and (iii) prevent or significantly reduce the rate at which radionuclides are released from the EBS to the Lower Natural Barrier. The complete EBS consists of the emplacement drifts, the drip shields, the waste packages, the naval spent nuclear fuel structure, the waste forms and waste package internal components, and emplacement pallets and inverts.

The NRC staff has reviewed the SAR and other information submitted in support of the license application, which includes information required by 10 CFR 63.21(c)(3),(9),(10) and (15), and finds, with reasonable expectation, that the relevant requirements of 10 CFR 63.114 and 63.342 are satisfied regarding the abstraction of degradation of engineered barriers in the TSPA model.

SER Section 2.2.1.3.2 Mechanical Disruption of Engineered Barriers

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

The NRC staff has reviewed SAR Section 2.3.4 and other information submitted in support of the license application, which includes information required by 10 CFR 63.21(c)(1)–(3), (9), (10), (15), and (19) related to mechanical and structural performance of EBS components, and finds, with reasonable expectation, that the relevant requirements of 10 CFR 63.114 and 63.342 are satisfied regarding the abstraction of mechanical disruption of engineered barriers in the performance assessment.

SER Section 2.2.1.3.3 Quantity and Chemistry of Water Contacting Engineered Barriers and Waste Forms

This SER section provides the NRC staff's evaluation of DOE's abstraction of the repository drift system that may alter the chemical composition and volume of water contacting the drip shield and waste package surfaces. It focuses on key features, events and processes that address (i) the chemistry of water entering the drifts, (ii) the chemistry of water in the drifts (tunnels), and (iii) the quantity of water in contact with the EBS. These three abstraction topics provide input to model the features and performance of the EBS (e.g., drip shields and waste packages) and their contributions to barrier functions. The range of testing environments was derived from a range of potential starting water compositions and from knowledge of near-field and in-drift processes that alter these compositions.

The NRC staff has reviewed the SAR and other information submitted in support of the license application, which includes information required by 10 CFR 63.21(c)(1), (9), (10) and (15), and finds, with reasonable expectation, that the relevant requirements of 10 CFR 63.114 and 63.342 are satisfied regarding the abstraction of the quantity and chemistry of water contacting engineered barriers and waste forms.

SER Section 2.2.1.3.4 Radionuclide Release Rates and Solubility Limits

This SER section provides the NRC staff's evaluation of analytical models and the processes that could result in water transport of radionuclides out of the EBS, including the waste packages and the emplacement inverts, and into the unsaturated zone (the rock mass directly below the repository horizon and above the water table). The NRC staff's evaluation focuses on the following: in-package chemical and physical environment, waste form degradation, concentration limits, availability and effectiveness of colloids, and engineered barrier system radionuclide transport. The EBS and the transport pathway within the drift (repository tunnel) are the initial barriers to radionuclide release. If a waste package is breached and water enters the waste package, the radionuclides contained in the package may be released from the EBS.

The NRC staff has reviewed the SAR and other information submitted in support of the license application, which includes information required by 10 CFR 63.21(c)(3),(9),(15), and finds, with reasonable expectation, that the relevant requirements of 10 CFR 63.114 and 63.342 are satisfied regarding the abstraction of radionuclide release rates and solubility limits.

SER Section 2.2.1.3.5 Climate and Infiltration

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

The NRC staff has reviewed the SAR and other information submitted in support of the license application, which includes information required by 10 CFR 63.21(c)(1), (9),(10),(15), and finds, with reasonable expectation, that the relevant requirements of 10 CFR 63.114, 63.305, and 63.342 are satisfied regarding the abstraction of climate and infiltration.

SER Section 2.2.1.3.6 Unsaturated Zone Flow

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

The NRC staff has reviewed the SAR and other information submitted in support of the license application, which includes information required by 10 CFR 63.21(c)(1), (9), (10), (15), and (19), and finds, with reasonable expectation, that the relevant requirements of 10 CFR 63.114 and 10 CFR 63.342 are satisfied regarding the abstraction of unsaturated zone flow, thermal conditions in the host rock, and in-drift thermohydrological conditions excluding conditions for the engineered components.

SER Section 2.2.1.3.7 Radionuclide Transport in the Unsaturated Zone

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

The NRC staff has reviewed the SAR and other information submitted in support of the license application, which includes information required by 10 CFR 63.21(c)(1), (9), (10), and (15), and finds, with reasonable expectation, that the relevant requirements of 10 CFR 63.114 and 63.342 are satisfied regarding the abstraction of radionuclide transport in the unsaturated zone.

SER Section 2.2.1.3.8 Flow Paths in the Saturated Zone

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

The NRC staff has reviewed the SAR and other information submitted in support of the license application, which includes information required by 10 CFR 63.21(c)(1), (9), (10), (15), and (19), and finds, with reasonable expectation, that the relevant requirements of 10 CFR 63.114 and 10 CFR 63.342 are satisfied regarding the abstraction of flow paths in the saturated zone.

SER Section 2.2.1.3.9 Radionuclide Transport in the Saturated Zone

This SER section provides the NRC staff's evaluation of the model abstraction for transport of radionuclides in the saturated zone. The NRC staff's technical review focuses on (i) how DOE represented the geological, hydrological, and geochemical features of the saturated zone in a framework for modeling the transport processes; (ii) how DOE integrated the saturated zone transport abstraction with other TSPA model abstractions for performance assessment calculations; and (iii) how DOE included and supported the important transport processes of advection and dispersion, sorption, matrix diffusion, colloid-associated transport, and radioactive decay and ingrowth in the saturated zone radionuclide transport abstraction.

The NRC staff reviewed the SAR and other information submitted in support of the license application, which includes information required by 10 CFR 63.21(c)(1), (9), and (15), and finds, with reasonable expectation, that the relevant requirements of 10 CFR 63.114 and 63.342 are satisfied regarding the abstraction of radionuclide transport in the saturated zone.

SER Section 2.2.1.3.10 Igneous Disruption of Waste Packages

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

The NRC staff reviewed the SAR and other information submitted in support of the license application, which includes information required by 10 CFR 63.21(c)(1), (9), and (15), and finds, with reasonable expectation that the relevant requirements of 10 CFR 63.114 and 63.342 are satisfied regarding the abstraction of igneous disruption of waste packages.

SER Section 2.2.1.3.12 Concentration of Radionuclides in Groundwater

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

The NRC staff has reviewed the SAR and other information submitted in support of the license application relevant to the concentration of radionuclides in groundwater, and finds, with reasonable expectation, that the requirements of 10 CFR 63.312(c) are satisfied. The applicant adequately demonstrated that the RMEI uses well water with average concentrations of radionuclides by dividing the annual mass fluxes of radionuclides reaching the accessible environment boundary by the annual water use of 3,000 acre-ft [3.7×10^9 L].

SER Section 2.2.1.3.13 Airborne Transport and Redistribution of Radionuclides

This SER section provides the NRC staff's evaluation of the volcanic ash exposure scenario and the groundwater exposure scenario. First, this SER section provides the NRC staff's evaluation of the airborne transport and deposition of radionuclides expelled by a potential future volcanic eruption and the subsequent redistribution of those radionuclides in soil. Second, this SER section evaluates redistribution of radionuclides in soil that arrive in the accessible environment through groundwater transport.

The NRC staff has reviewed the SAR and other information submitted in support of the license application, which includes information required by 10 CFR 63.21(c)(1),(9), and (15), and finds, with reasonable expectation, that the relevant requirements of 10 CFR 63.114, 63.305, and 63.342 are satisfied regarding the abstraction of airborne transport and redistribution of radionuclides.

SER Section 2.2.1.3.14 Biosphere Characteristics

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

Inhalation pathways include breathing resuspended soil, aerosols from evaporative coolers, and radon gas and its decay products.

The NRC staff has reviewed the SAR and other information submitted in support of the license application, which includes information required by 10 CFR 63.21(c)(1), and finds, with reasonable expectation, that the requirements in 10 CFR 63.305, 63.311(b), and relevant requirements of 10 CFR 63.114, 63.312, and 63.342 are satisfied regarding the biosphere characteristics.

2.4 TSPA Model Calculations

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

SER Section 2.2.1.4.1 Demonstration of Compliance With the Postclosure Public Health and Environmental Standards (Individual Protection)

This SER section provides the NRC staff's evaluation of the applicant's compliance with the individual protection standards. Section 63.311 requires that the average annual dose must not exceed 0.15 mSv/yr [15 mrem/yr] during the initial 10,000 years following disposal and not exceed 1.0 mSv/yr [100 mrem/yr] after 10,000 years up to 1 million years. The performance assessment used for the individual protection calculation considers both likely and unlikely events and the radiological exposure pathways.

The NRC staff has reviewed the SAR and the other information submitted in support of the license application, which includes the information required by 10 CFR 63.21(c)(11), and finds, with reasonable expectation, that the requirements of 10 CFR 63.113(b) are satisfied.

SER Section 2.2.1.4.2 Demonstration of Compliance With the Human Intrusion Standard

This SER section provides the NRC staff's evaluation of the applicant's compliance with the human intrusion standard. The human intrusion standard in Section 63.321 requires the applicant to determine the earliest time after disposal that the waste packages would degrade sufficiently so that a human intrusion from exploratory groundwater drilling could occur without recognition by the drillers. Section 63.321(b) requires that the average annual dose must not exceed 0.15 mSv/yr [15 mrem/yr] during the initial 10,000 years after disposal and not exceed 1.0 mSv/year [100 mrem/yr] after 10,000 years up to 1 million years. The performance assessment used for the human intrusion calculation considers likely events and the radiological exposure pathways.

The NRC staff has reviewed the SAR and the other information submitted in support of the license application, which includes the information required by 10 CFR 63.21(c)(13), and finds, with reasonable expectation, that the requirements of 10 CFR 63.113(d) are satisfied.

SER Section 2.2.1.4.3 Demonstration of Compliance With Separate Groundwater Protection Standards

This SER section provides the NRC staff's evaluation of the applicant's compliance with the groundwater protection standard. The NRC's regulations provide separate standards to protect the groundwater resources in the vicinity of Yucca Mountain and specify the approach for estimating the concentration of radionuclides in groundwater. The groundwater protection standards provide for different limits, depending on the radionuclide. There are three distinct groups of radionuclides with the following limits: (i) radionuclides that are characterized as alpha emitters (e.g., Np-237) are grouped, and the combined concentration must be less than 15 pCi/L (this group explicitly excludes radon and uranium); (ii) radionuclides that are characterized as beta- and photon-emitting radionuclides (e.g., I-129, Tc-99) are grouped together, and the combined concentration cannot result in a dose exceeding 0.04 mSv [4 mrem] per year to the whole body or any organ, on the basis of drinking 2 L [0.53 gal] of water per day at the combined concentration; and (iii) the combined concentration of Ra-226 and Ra-228 cannot exceed a concentration of 5 pCi/L. The performance assessment used for the separate groundwater protection calculation considers likely events and the drinking water exposure pathway.

The NRC staff has reviewed the SAR and other information submitted in support of the license application, which includes information required by 10 CFR 63.21(c)(12), and finds with reasonable expectation, that the requirements of 10 CFR 63.113(c) are satisfied.

SER Section 2.5.4 Expert Elicitation

SER Section 2.5.4 provides the NRC staff's evaluation of the three expert elicitations DOE used in support of its SAR. Expert elicitations were conducted in the areas of seismic hazard (SAR Section 2.2.2.1), igneous activity (SAR Section 1.1.6.2, Section 2.2.2.2, and Section 2.3.11), and saturated zone flow and transport (SAR Section 2.3.9.2).

The NRC staff has reviewed the SAR and other information submitted in support of the license application, and finds, with reasonable expectation, that the requirement in 10 CFR 63.21(c)(19) is satisfied.

3.0 Conclusions

The NRC staff has reviewed and evaluated the DOE's Safety Analysis Report, Chapter 2: Repository Safety After Permanent Closure and the other information submitted in support of its license application and has found that DOE submitted applicable information required by 10 CFR 63.21. The NRC staff has also found with reasonable expectation, that (i) the proposed Yucca Mountain repository design meets the applicable performance objectives in Subpart E, including the requirement that the repository be composed of multiple barriers and (ii) based on performance assessment evaluations that are in compliance with applicable regulatory requirements, meets the 10 CFR Part 63, Subpart L limits for individual protection, human intrusion, and separate standards for protection of groundwater.

4.0 References

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ACRONYMS AND ABBREVIATIONS

AFM	active fracture model
AMR	analysis and model reports
APE	annual probability of exceedance
ASTM	American Society for Testing and Materials
BDCF	Biosphere Dose Conversion Factors
BSC	Bechtel SAIC Company, LLC
BWR	boiling water reactor
CDSP	codisposal package
CFR	Code of Federal Regulations
CFu	Crater Flat undifferentiated
CHn	Calico Hills nonwelded
CNWRA®	Center for Nuclear Waste Regulatory Analyses
CRWMS-M&O	Civilian Radioactive Waste Management-Management & Operation
CSNF	commercial spent nuclear fuel
DHLW	defense high-level waste
DOE	U.S. Department of Energy
DVRGFSM	Death Valley Regional Groundwater Flow System Model
EBS	engineered barrier system
EPA	U.S. Environmental Protection Agency
ERD	Error Resolution Document
ERMYN	Environmental Radiation Model for Yucca Mountain Nevada
EWDP	Early Warning Drilling Program
FAR	Fortymile Wash Ash Redistribution
FEPs	features, events, and processes
FEHM	finite element heat transfer code
GI	Geologic Information
GROA	geologic repository operations area
ITWI	important to waste isolation
LA	license application
MASSIF	Mass Accounting System for Soil Infiltration and Flow
MCO	multicanister overpack
MDEB	mechanical disruption of engineered barriers
MIC	microbially influenced corrosion
NC–EWDP	Nye County Early Warning Drilling Program
NOAA	U.S. National Oceanic and Atmospheric Administration
NRC	U.S. Nuclear Regulatory Commission
OECD	Organization for Economic Co-operation and Development
PCSA	preclosure safety analysis
PFDHA	probabilistic fault displacement hazard analysis
PGA	peak ground acceleration
PGV	peak ground velocity
PSHA	probabilistic seismic hazard assessment
PTn	Paintbrush Tuff nonwelded
PVHA	probabilistic volcanic hazard assessment
PVHA-U	probabilistic volcanic hazard assessment-update
PWR	pressurized water reactor

ACRONYMS AND ABBREVIATIONS (continued)

QA	quality assurance
RAI	request for additional information
RB	repository block
RIPB	risk-informed, performance-based
RMEI	reasonably maximally exposed individual
RMS	root-mean-square
RST	residual stress threshold
SA	spectral accelerations
SAR	Safety Analysis Report
SCA	Seismic Consequence Abstractions
SCC	stress corrosion cracking
SDFR	slip-dissolution aging and film-rupture
SER	Safety Evaluation Report
SNF	spent nuclear fuel
SNL	Sandia National Laboratories
SSHAC	Senior Seismic Hazard Analysis Committee
SZEE	saturated zone flow and transport expert elicitation
TAD	transportation, aging, and disposal
TCw	Tiva Canyon welded
TEDE	total effective dose equivalent
TSPA	Total System Performance Assessment
TSw	Topopah Spring welded
UDEC	universal distinct element code
USGS	U.S. Geological Survey
UZ	unsaturated zone
WAPDEG	Waste Package Degradation
YMRP	Yucca Mountain Review Plan

INTRODUCTION

Volume 3, Postclosure: Repository Safety After Permanent Closure, of this Safety Evaluation Report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's review and evaluation of the Safety Analysis Report (SAR) the U.S. Department of Energy (DOE) provided in its June 3, 2008, license application (LA) submittal (DOE, 2008ab), as updated on February 19, 2009 (DOE, 2009av). The NRC staff also reviewed information DOE provided in response to the NRC staff's requests for additional information and other information that DOE provided related to the SAR. In particular, this SER Volume 3 documents the results of the NRC staff's evaluation to determine whether the proposed repository design for Yucca Mountain complies with the performance objectives and requirements that apply after the repository is permanently closed. These performance objectives and requirements can be found in NRC's regulations at 10 CFR Part 63, Subparts E and L. The NRC staff's safety evaluation considers the proposed geologic repository's multiple barriers, both natural and engineered (manmade); and the performance assessments (including model abstractions) used for the individual protection, the separate groundwater protection, and the human intrusion evaluations.

Other portions of the NRC staff's safety review have been, or will be, documented in other volumes. SER Volume 1, NUREG–1949 (NRC, 2010aa) documents the results of the NRC staff's review of DOE's General Information. SER Volume 2 will document the results of the NRC staff's review and evaluation of DOE's compliance with preclosure safety objectives and requirements. SER Volume 4 will document the results of the NRC staff's review and evaluation of DOE's demonstration of compliance with administrative and programmatic requirements. SER Volume 5 will document the NRC staff's review and evaluation of probable subjects of license specifications and proposed conditions of construction authorization.

NRC's regulations at 10 CFR Part 63 provide site-specific criteria for geologic disposal at Yucca Mountain. Pursuant to 10 CFR Part 63, there are several stages in the licensing process: the site characterization stage, the construction stage, a period of operations, and termination of the license. The multi-staged licensing process affords the Commission the flexibility to make decisions in a logical time sequence that accounts for DOE collecting and analyzing additional information over the construction and operational phases of the repository. The period of operations includes (i) the time during which emplacement would occur, (ii) any subsequent period before permanent closure during which the emplaced wastes are retrievable, and (iii) permanent closure. In addition, 10 CFR Part 63 represents a risk-informed, performance-based regulatory approach to the review of geological disposal. This risk-informed, performance-based regulatory approach uses risk insights, engineering analysis and judgments, performance history, and other information to focus on the most important activities and to focus the review to areas most significant to safety or performance. Therefore, the SER includes discussions regarding how the NRC staff used risk information in its review of DOE's application. In conducting its review, the NRC staff was guided by the review methods and acceptance criteria outlined in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa), as supplemented by the Division of High-Level Waste Repository Safety Director's Policy and Procedure Letter 14: Application of YMRP for Review Under Revised Part 63 (NRC, 2009ab).

Risk-Informed and Performance-Based Review

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

To support its risk-informed, performance-based review, the NRC staff initially reviewed DOE's information on the repository's natural and engineered barriers that DOE identified as important to waste isolation in the performance assessment. The SAR describes each barrier's capability and provides the technical basis for that capability. This information describes DOE's understanding of each barrier's capability to prevent or substantially delay the movement of water or radioactive materials. The NRC staff's review of DOE's information regarding the repository barriers presented in SER Section 2.2.1.1 provides an understanding of each barrier's importance to waste isolation to help focus the NRC staff's review. Particular parts of the NRC staff's review are emphasized on the basis of the risk insights (i.e., those attributes of the repository system most important to repository performance). Additionally, the NRC staff has considered independent risk insights from previous performance assessments conducted for the Yucca Mountain site, detailed process modeling efforts, laboratory and field experiments, and natural analog studies and has identified this information, as appropriate.

System Description and Demonstration of Multiple Barriers

NRC regulations at 10 CFR Part 63 require that a geologic repository at Yucca Mountain include multiple barriers, both natural and engineered. Barriers prevent or limit the movement of water or radioactive material. A multiple barrier approach ensures that the overall repository system is robust and not wholly dependent on any single barrier. The NRC requires that DOE identify these barriers when it calculates how the repository will perform. DOE is required to describe the capability of each barrier and provide the technical basis for its description. In its SAR for the proposed repository at Yucca Mountain, DOE identified three barriers: the Upper Natural Barrier, the Engineered Barrier System (EBS), and the Lower Natural Barrier. The Upper Natural Barrier is composed of features above the repository (i.e., topography, surficial soils, and the unsaturated zone) that reduce the movement of water downward toward the repository, which in turn reduces the rate of movement of water from the radioactive waste in the repository to the accessible environment. The EBS includes different engineering features (e.g., emplacement drifts, drip shields, waste packages and its internal components, and emplacement pallets and inverts) that are designed to (i) enhance the performance of the waste package, preventing radionuclide releases while it is intact; (ii) limit radionuclide releases after the waste package is breached by limiting the amount of water that can contact the waste package; and (iii) limit radionuclide release from the engineered barrier system through sorption processes. The Lower Natural Barrier comprises two features: the unsaturated zone below the repository and the saturated zone, both of which prevent or reduce the rate of radionuclide movement from the repository to the accessible environment through such processes as the slow movement of water and sorption of radionuclides onto mineral surfaces. Each of these barriers includes features that DOE described as important to waste isolation. The NRC staff's review is provided in SER Section 2.2.1.1.

Review of Postclosure Total System Performance

DOE conducted an analysis, through its Total System Performance Assessment (TSPA) computer model, that evaluates the behavior of the high-level waste repository due to the potential release of radionuclides from the repository. The performance assessment provides a method to evaluate the range of features (e.g., geologic rock types, waste package materials), events (e.g., earthquakes, igneous activity), and processes (e.g., corrosion of metal waste packages, sorption of radionuclides onto rock surfaces) that are relevant to the behavior of a repository at Yucca Mountain. The NRC staff reviewed the TSPA analytic models and analyses DOE provided in its SAR.

<u>Scenario Analysis and Event Probability</u>

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

The NRC staff uses YMRP Section 2.2.1.2.1 to evaluate the applicant's scenario analysis, which is documented in four separate SER sections (2.2.1.2.1.3.1 through 2.2.1.2.1.3.4). Section 2.2.1.2.1.3.1 contains the NRC staff's evaluation of both the applicant's methodology to develop a list of FEPs and DOE's list of the FEPs that it considered for inclusion in the performance assessment analyses. In Section 2.2.1.2.1.3.2, the NRC staff evaluates DOE's screening of its list of FEPs, including DOE's technical bases for the exclusion of FEPs from its performance assessment. DOE's formation of scenario classes and the exclusion of specific scenario classes in DOE's performance assessment analyses are evaluated in SER Sections 2.2.1.2.1.3.3 and 2.2.1.2.1.3.4, respectively.

The NRC staff's evaluation of the applicant's methodology and conclusions on the probability of events included in the performance assessments is addressed in SER Section 2.2.1.2.2. Hence, SER Section 2.2.1.2.2 is aimed at the second of the three risk questions, "How likely is it to happen?"

<u>Model Abstraction</u>

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

The evaluation of the model abstraction process begins with the review of the repository design and the data characterizing the geology and the performance of the design and proceeds through the development of models used in the performance assessment. The model abstraction review process ends with a review of how the abstracted models are implemented in the TSPA model (e.g., parameter ranges and distributions, integration with model abstractions for other parts of the repository system, representation of spatial and temporal scales, and whether the performance assessment model appropriately implements the abstracted model).

The TSPA is a complex analysis with many parameters, and DOE may use conservative assumptions to address uncertainties or justify a simplified modeling approach. DOE provided a technical basis for the selection of models and parameter ranges or distributions. The NRC staff's evaluation of the technical bases supporting models and parameter ranges or distributions considers whether the approach results in calculated doses that would underestimate, rather than overestimate, the dose to the reasonably maximally exposed individual (RMEI). In particular, DOE's assumption of conservatism as a basis for simplifying models and parameters is evaluated by the NRC staff to verify that any simplifications do not unintentionally result in nonconservative results (i.e., underestimate dose to the RMEI).

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

The NRC staff has separated its model abstraction review into 13 categories that are addressed in SER Sections 2.2.1.3.1 through 2.2.1.3.14. Two of the topics in the YMRP are discussed as a single topic in the SER; however, the numbering system in the YMRP was retained to the maximum extent possible (e.g., Biosphere Characteristics is Section 2.2.1.3.14 in the YMRP and the SER). The review of Airborne Transport of Radionuclides (YMRP Section 2.2.1.3.11) and Redistribution of Radionuclides in Soil (YMRP Section 2.2.1.3.13) are discussed in SER Section 2.2.1.3.13 because the NRC staff considers a single discussion of these two topics provides for more clarity. Therefore, the SER does not contain a section numbered 2.2.1.3.11.

Expert Elicitation

Expert elicitation is a formal, structured, and well-documented process for obtaining the judgments of multiple experts on various technical areas. Pursuant to 63.21(c)(19), DOE must explain how expert elicitation was used. Consistent with YMRP Section 2.5.4, DOE could elect to use the subjective judgments of experts, or groups of experts, to interpret data and address technical issues and inherent uncertainties when assessing the long-term performance of a geologic repository. In its SAR, the applicant used the results of three formal expert elicitations to complement and supplement other sources of scientific and technical information such as data collection, analyses, and experimentation. The NRC staff has reviewed DOE's use of expert elicitation, which includes a technical review of the results of these elicitations.

SER Section 2.5.4 provides the NRC staff's review of the three expert elicitations DOE used in support of its SAR. Expert elicitations were conducted in the areas of seismic hazard

(SAR Section 2.2.2.1); igneous activity (SAR Section 1.1.6.2, Section 2.2.2.2, and Section 2.3.11); and saturated zone flow and transport (SAR Section 2.3.9.2).

Sections of the Postclosure Review

The individual sections documenting the NRC staff review are:

1. System Description and Demonstration of Multiple Barriers (SER Section 2.2.1.1.1)

2. Scenario Analysis (SER Section 2.2.1.2.1)

3. Identification of Events With Probabilities Greater Than 10^{-8} Per Year (SER Section 2.2.1.2.2)

4. Degradation of Engineered Barriers (SER Section 2.2.1.3.1)

5. Mechanical Disruption of Engineered Barriers (SER Section 2.2.1.3.2)

6. Quantity and Chemistry of Water Contacting Engineered Barriers and Waste Forms (SER Section 2.2.1.3.3)

7. Radionuclide Release Rates and Solubility Limits (SER Section 2.2.1.3.4)

8. Climate and Infiltration (SER Section 2.2.1.3.5)

9. Unsaturated Zone Flow (SER Section 2.2.1.3.6)

10. Radionuclide Transport in the Unsaturated Zone (SER Section 2.2.1.3.7)

11. Flow Paths in the Saturated Zone (SER Section 2.2.1.3.8)

12. Radionuclide Transport in the Saturated Zone (SER Section 2.2.1.3.9)

13. Igneous Disruption of Waste Packages (SER Section 2.2.1.3.10)

14. Concentration of Radionuclides in Groundwater (SER Section 2.2.1.3.12)

15. Airborne Transport and Redistribution of Radionuclides (SER Section 2.2.1.3.13)

16. Biosphere Characteristics (SER Section 2.2.1.3.14)

17. Demonstration of Compliance With the Postclosure Public Health and Environmental Standards (Individual Protection) (SER Section 2.2.1.4.1)

18. Demonstration of Compliance With the Human Intrusion Standard (SER Section 2.2.1.4.2)

19. Demonstration of Compliance With Separate Groundwater Protection Standards (SER Section 2.2.1.4.3)

20. Expert Elicitation (SER Section 2.5.4)

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CHAPTER 1

2.2.1.1 System Description and Demonstration of Multiple Barriers

2.2.1.1.1 Introduction

The performance objectives in 10 CFR Part 63 for the repository after permanent closure require that the geologic repository must include multiple barriers, consisting of both natural barriers and an engineered barrier system. Natural and engineered barriers isolate waste by preventing or substantially reducing the rate of movement of water or radionuclides from the Yucca Mountain repository to the accessible environment. A comprehensive description of the capabilities of the natural and engineered barriers would identify the risk-significant attributes for the repository performance. Safety Evaluation Report (SER) Section 2.2.1.1 evaluates the U.S. Department of Energy's (DOE's or applicant's) description of the capabilities of the barriers in the geologic repository. The technical basis for the barrier capability is evaluated in SER Section 2.2.1.3.

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

The emphasis of the U.S. Nuclear Regulatory Commission (NRC) staff's integrated review of the applicant's performance assessment is not solely focused on the isolated performance of individual barriers, but rather on ensuring that the repository system is robust. The purpose of this SER Section is to provide an understanding of how the natural barriers and the engineered barrier system work in combination to enhance the resiliency of the geologic repository. To provide an understanding of integrated repository performance, 10 CFR 63.115 requires the applicant to

- Identify barriers considered important to waste isolation
- Describe each barrier's capability
- Provide a technical basis for that capability which is based on and consistent with the technical basis for the performance assessment used to demonstrate compliance with the performance objectives in 10 CFR 63.113(b) and (c)

This risk information provides DOE's understanding of each barrier's capability to prevent or substantially delay the movement of water or radioactive material. The NRC staff can use this risk information to implement a risk-informed approach in its review of the applicant's performance assessment calculations.

The NRC staff's evaluation is based on information provided in the Safety Analysis Report (SAR) (DOE, 2008ab), as supplemented by DOE responses to the NRC staff's requests for additional information (DOE, 2009an,bu). DOE provided a description of the barrier capabilities in SAR Chapter 2.1. This description, supplemented by DOE's responses to the NRC staff's requests for additional information, is used by the NRC staff in its review of the technical bases for the performance assessment, as documented in SER Section 2.2.1.3. SER Section 2.2.1.1 focuses on the adequacy of DOE's description of the barrier capabilities. As discussed in the

NRC staff's Yucca Mountain Review Plan (YMRP) Section 2.2.1 (NRC, 2003aa), the multiple barrier review focuses on each barrier's importance to waste isolation. Following the guidance in the YMRP Section 2.2.1, the NRC staff evaluated the information required by 10 CFR 63.21(c)(1), (9), (10), (14), and (15).

2.2.1.1.2 Regulatory Requirements

The regulatory requirements applicable to multiple barriers are found in 10 CFR 63.113(a) and 10 CFR 63.115(a–c). These require an applicant to

- Ensure that the geologic repository includes multiple barriers, consisting of both natural barriers and an engineered barrier system
- Identify those features of the repository that are considered barriers important to waste isolation (ITWI)
- Describe the capabilities of those barriers, taking into account uncertainties in characterizing and modeling the behavior of the barriers
- Provide a technical basis for the description of the capability that is based on and consistent with the technical basis for the performance assessment used to demonstrate compliance with 10 CFR 63.113(b) and (c)

Definitions and discussions of important terms and concepts, such as "barrier" and "important to waste isolation," are located in 10 CFR 63.2 and 10 CFR 63.102(h). For example, 10 CFR 63.2 states that the term "barrier" means any material, structure, or feature that, for a period to be determined by NRC, prevents or substantially reduces the rate of movement of water or radionuclides from the Yucca Mountain repository to the accessible environment, or prevents the release or substantially reduces the release rate of radionuclides from the waste.

The NRC staff reviewed information provided in the SAR (DOE, 2008ab), as supplemented by DOE responses to NRC staff's requests for additional information (DOE, 2009an,bu) using the review methods and acceptance criteria provided in YMRP Section 2.2.1.1. The three acceptance criteria that are based on meeting the requirements of 10 CFR 63.115 and 10 CFR 63.113(a) are

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

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

2.2.1.1.3 Technical Review

Description of DOE Approach

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

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

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

- The specific function of each barrier component and how the barrier component carries out its functions
- The time period over which the barrier functions and how DOE expects the capability of the barrier to evolve over time
- How uncertainty in the barrier capability has been accounted for in the performance assessment
- The impact of disruptive events on the barrier, if any
- A quantitative evaluation of the barrier capability to carry out its barrier functions

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

2.2.1.1.3.1 Identification of Barriers

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

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

DOE's safety classification identified in SAR Table 2.1-1 indicates whether each individual feature or component is considered important to waste isolation and whether each feature or component is clearly linked to its capability and to the upper natural barrier, engineered barrier system, or the lower natural barrier. Therefore, because DOE indicated whether features or components are important to waste isolation and identified their capabilities, the NRC staff concludes that DOE has identified the barriers that are relied on to achieve compliance with 10 CFR 63.113. Because the list in SAR Table 2.1-1 includes features from both the engineered and natural systems, the NRC staff concludes that DOE identified barriers that include at least one feature from the engineered system and one from the natural system.

DOE also identified three engineered system components as important to waste isolation on the basis of their capability to reduce the probability of criticality. DOE used these system components to screen out the criticality event class from the performance assessment because of the low probability of occurrence (SAR Table 2.2-5; SNL, 2008ab). NRC staff's evaluation of DOE's technical basis for screening out the criticality event class can be found in SER Section 2.2.1.2.1.

2.2.1.1.3.2 Description and Technical Basis for Barrier Capability

NRC Staff's Review Process

The NRC staff's review of the description and technical basis for barrier capability is based on a list of 22 individual features presented in SAR Table 2.1-1. For purposes of evaluation, the NRC staff consolidated these features to yield nine features as shown in the second column of SER Table 1-1. The NRC staff consolidated the 22 barrier features because it found that several features in the second column of SAR Table 2.1-1 are related. For example, the emplacement drift is referred to twice. The NRC staff, therefore, consolidated the two emplacement drift entries into a grouped entry titled "Emplacement Drift." Also, 11 of the features listed in SAR Table 2.1-1 were prefaced with the term "Waste Form and Waste Package Internals." The NRC staff grouped all of these 11 features into 1 component titled "Waste Form and Waste Package Internals." The NRC staff included cladding into this grouped category because the cladding is a component that contains the waste form and is internal to the waste package. The NRC staff also noted that neither the emplacement pallet nor the invert was classified as important to waste isolation in SAR Table 2.1-1, and that both components serve to support the waste package. The NRC staff, therefore, consolidated these two features into a single barrier feature titled "Emplacement Pallet and Invert." The resulting consolidated list is consistent with the grouping that DOE used in SAR Section 2.1.2.2 in its summary of the features, processes, and characteristics of the engineered barrier system that are important to waste isolation.

Table 1-1. Summary of NRC Staff's Barrier Component Review			
Barrier	**Barrier Feature**	**SAR Table 2.1-1 ITWI* Components**	**SAR Table 2.1-1 Non-ITWI Components**
Upper Natural Barrier	Topography and Surficial Soils	Topography and Surficial Soils	None
Upper Natural Barrier	Unsaturated Zone Above the Repository	Unsaturated Zone Above the Repository	None
Engineered Barrier System (EBS)	Emplacement Drift	Emplacement Drift	Emplacement Drift: Nonemplacement Openings, Closure, Ground Support, and Ventilation System
EBS	Drip Shield	Drip Shield	None
EBS	Waste Package	Waste Package	None
EBS	Waste Form and Waste Package Internals	• Transport, aging, and disposal (TAD) canister • Naval canister • Commercial spent nuclear fuel (SNF) and high-level waste glass • Naval SNF • Naval SNF canister system components† • TAD canister internals† • DOE SNF canister internals†	• DOE SNF canister • High-level waste canister • Codisposal package internals • DOE SNF • Cladding
EBS	Emplacement Pallet and Invert	None	• Waste Package Pallet • Invert
Lower Natural Barrier	Unsaturated Zone Below the Repository	Unsaturated Zone Below the Repository	None
Lower Natural Barrier	Saturated Zone	Saturated Zone	None
*ITWI stands for "important to waste isolation." †DOE identified these components as important to waste isolation solely in relation to their capability to reduce the probability of criticality.			

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

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

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

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

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

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

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

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

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

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

- Describes whether the barrier capability DOE described is consistent with the definition of a barrier in 10 CFR 63.2

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

- Describes whether the identified capabilities are consistent with the results from the total system performance assessment (in reviewing these analyses, the NRC staff examined whether the numerical results used to illustrate barrier capability were consistent with the intermediate results used to compute the radiation dose in the total system performance assessment calculation)

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

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

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

2.2.1.1.3.2.1 Upper Natural Barrier: Topography and Surficial Soils

The NRC staff reviewed DOE's description of the barrier capability of the topography and surface soils. DOE described the barrier capabilities of the topography and the surficial soils to prevent or substantially reduce the rate of water movement qualitatively in SAR Section 2.1.2.1.1 and quantitatively in SAR Section 2.1.2.1.6.1. DOE supplemented this description in DOE (2009an, Enclosure 1). DOE used net infiltration as a percentage of annual precipitation to quantify the barrier capability of topography and surficial soils. The NRC staff concludes that the capabilities DOE described are consistent with the definition of a barrier at 10 CFR 63.2 because DOE described a capability that substantially reduces infiltration into the unsaturated zone, which in turn reduces the rate of movement of water from the nuclear waste in the repository to the accessible environment.

DOE's climate and infiltration analyses are summarized in SAR Tables 2.3.1-2, 2.3.1-3, and 2.3.1-4. DOE concluded in SAR Section 2.1.2.1.1 and in DOE (2009an, Enclosure 1) that for approximately 10,000 years following closure of the repository, a limited amount of water would infiltrate the unsaturated zone above the repository at Yucca Mountain. DOE attributed the low rate of infiltration to low precipitation that is substantially further reduced by high rates of evapotranspiration (e.g., uptake by plants, surface evaporation) and surface runoff. In SAR Section 2.1.2.1.1, DOE stated that the average net infiltration rate estimates range from about 3 to 17 percent of the total precipitation, depending upon the climate state and the infiltration scenario. For the post-10,000-year period, DOE stated that it used the deep percolation rate distribution specified in the proposed 10 CFR 63.342(c)(2) rule. In DOE (2009cb, Enclosure 6), DOE stated that use of the distribution of deep percolation specified in the final 10 CFR 63.342(c)(2) rule led to an insignificant increase in radiation dose.

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

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

In SAR Section 2.1.2.1.4, DOE described sources of uncertainty that are considered in the climate and infiltration model. Sources of uncertainty include (i) the interpretation of the geologic record of past climates, (ii) the parameters describing evapotranspiration, (iii) the applicability of models, and (iv) the characteristics of the Yucca Mountain site. DOE also addressed the uncertainty in the barrier capability by describing results from its infiltration model demonstrating the probability of different infiltration scenarios (SAR Section 2.3.2.4.1.2.4.5 and Table 2.3.2-27). DOE demonstrated in DOE (2009bo, Enclosure 5) that adjusting the probability weighting of these scenarios based on deep subsurface observations of chloride and temperature (SAR 2.3.2.4.1.2.4.5 and Table 2.3.2-27) reduced the average infiltration fluxes in the initial 10,000 years following permanent closure by approximately 50 percent. DOE addressed infiltration uncertainties in the post-10,000-year period in SAR Section 2.3.2.4.1.2.4.2 by using a weighting of net infiltration scenarios to yield a distribution of deep percolation fluxes comparable to the distribution specified in the proposed 10 CFR 63.342(c)(2). DOE (2009cb, Enclosure 6) stated that use of the distribution of deep percolation specified in the final 10 CFR 63.342(c)(2) rule led to an insignificant increase in radiation dose. Because DOE described the sources of uncertainty and demonstrated the range of uncertainty in its infiltration estimates, the NRC staff concludes that DOE has adequately addressed uncertainty in its descriptions of barrier capability. SER Section 2.2.1.3.6.3.2 documents the NRC's staff's evaluation of DOE's approach to and representation of infiltration uncertainty in the performance assessment in the first 10,000 years and in the post-10,000-year period.

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

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

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

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

2.2.1.1.3.2.2 Upper Natural Barrier: Unsaturated Zone Above the Repository

The NRC staff reviewed the DOE description of the barrier capability of the unsaturated zone above the repository. DOE described the capability of the unsaturated zone above the repository to prevent or substantially reduce seepage qualitatively in SAR Section 2.1.2.1.2 and quantitatively in SAR Section 2.1.2.1.6.2. DOE supplemented this description in

DOE (2009an, Enclosures 1 and 2). The NRC staff concludes that the capabilities DOE described are consistent with the definition of a barrier at 10 CFR 63.2 because they describe a capability to prevent or substantially reduce seepage of water into the emplacement drifts, which in turn substantially reduces the rate of water movement from the nuclear waste in the repository to the accessible environment.

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

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

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

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

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

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

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

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

2.2.1.1.3.2.3 Engineered Barrier System: Emplacement Drift

The NRC staff reviewed the DOE description of the barrier capability of the emplacement drift. DOE discussed the barrier capabilities of the emplacement drift in SAR Section 2.1.2.2 under the discussion titled "Emplacement Drift" and in DOE (2009an, Enclosures 1 and 3). DOE stated that the capability of the emplacement drift to prevent or substantially reduce the movement of water is associated with the capillary barrier discussed under the upper natural barrier in SAR Section 2.1.2.1. DOE associated the capability of the drift to prevent or reduce the rate of movement of radionuclides with the effect of temperature and water chemistry on various processes affecting the degradation of the other engineered barrier system (EBS) components (e.g., drip shield, waste package, and waste form) and radionuclide transport. DOE (2009an, Enclosures 1 and 3) specifically identified and discussed the roles of individual processes in the capability of the emplacement drifts. The NRC staff finds that DOE has described the emplacement drifts capabilities as follows:

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

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

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

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

The NRC staff finds that DOE has adequately described the capabilities of the emplacement drift with respect to drift seepage because DOE described the effect of an intact and collapsed drift on the performance of the capillary barrier associated with the unsaturated zone above the repository. The NRC staff further finds that DOE has adequately described the capabilities of the emplacement drift as a barrier important to waste isolation with respect to the effect of the in-drift environment because DOE described the effect of the in-drift environment on degradation and transport processes within the drift that are described as aspects of the other components of the engineered barrier system. Therefore, the NRC staff concludes that the capabilities DOE described, with respect to drift seepage and the effect of the in-drift environment, are consistent with the definition of a barrier at 10 CFR 63.2 because these capabilities substantially reduce the rate of movement of water or radionuclides.

DOE discussed the time period over which the emplacement drift functions in SAR Section 2.1.2.2.3 and in DOE (2009an, Enclosure 3). DOE described the evolution of the mechanical stability of the drift and the in-drift environment and discussed how these changes affect the major processes associated with emplacement drift performance. The NRC staff concludes that these evaluations, along with the discussion of the time period over which DOE expects the drift to degrade due to seismic events, provide an adequate description of the time period over which the emplacement drift performs its function because DOE described both the timing and effect of seismically induced drift degradation. In SER Section 2.2.1.3.2.3.2, the

NRC staff addresses the adequacy and uncertainty in the capabilities of the emplacement drift related to the mechanical integrity of the drift opening.

DOE discussed the uncertainty in the performance of the emplacement drift in SAR Sections 2.1.2.2.4 and 2.3.4.4.8 and in DOE (2009an, Enclosure 3). DOE indicated that the uncertainties in the environmental conditions are a primary source of uncertainty in the performance of the engineered barrier system. In SAR Section 2.3.4.4.8, DOE discussed the sources and treatment of uncertainty in the evaluation of rockfall and demonstrated the effect of these uncertainties in SAR Figure 2.1-14, which demonstrates the range of uncertainties in the expected fraction of the drift filled with rubble. The NRC staff concludes that these discussions, supplemented by probabilistic outputs of the rockfall model showing the range of times for rubble to accumulate within the drift (SAR Section 2.1.2.2.6, Figure 2.1-14), demonstrate that DOE has adequately addressed uncertainty in its descriptions of barrier capability.

DOE summarized the technical bases of the engineered barrier system capability, which included the emplacement drift, in SAR Section 2.1.3.2. In its discussion, DOE indicated which TSPA models it used to support the description of the barrier capability of the engineered barrier system. DOE based its description of the barrier capability of the emplacement drift on three TSPA submodels: (i) the ambient and thermal seepage models described in SAR Section 2.3.3, (ii) the engineered barrier system mechanical degradation model described in SAR Section 2.3.4, and (iii) the in-drift chemical and physical environment model described in SAR Section 2.3.5.

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

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

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

2.2.1.1.3.2.4 Engineered Barrier System: Drip Shield

The NRC staff reviewed the description of the barrier capability of the drip shield. DOE discussed the capability of the drip shield to prevent or substantially reduce contact of seepage with the waste package in SAR Section 2.1.2.2 under the discussion titled “Drip Shield,” in SAR Section 2.1.2.2.1, and quantitatively in SAR Section 2.1.2.2.6. DOE supplemented its discussion in DOE (2009an, Enclosure 1). DOE addressed the drip shield’s capability to prevent seepage water from contacting the waste package during the thermal period in SAR Section 2.1.2.2. The thermal period is the time period when the temperature of the host rock and EBS are above the ambient temperature of the host rock (SER Section 2.2.1.3.6). During the thermal period, seepage water, if contacting the waste package, could lead to water chemistry that may initiate localized corrosion. The NRC staff concludes that the capabilities DOE described are consistent with the 10 CFR 63.2 definition of a barrier because they describe a capability to prevent or substantially reduce the rate of movement of water.

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

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

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

DOE described the sources of uncertainty in the drip shield capability in SAR Section 2.1.2.2.4 and quantitatively demonstrated the effect of uncertainties in SAR Section 2.1.2.2.6. These include, for example, uncertainties in the environmental conditions affecting the drip shield. DOE described specific analyses of uncertainty in the model abstractions for drip shield degradation in SAR Sections 2.3.4.5 and 2.3.6.8. The NRC staff concludes that these discussions, supplemented by probabilistic outputs of the drip shield degradation model showing the range of times for failure of the drip shield (SAR Section 2.1.2.2.6, Figure 2.1-11), demonstrate that DOE has adequately addressed uncertainty in its descriptions of barrier capability.

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

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

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

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

2.2.1.1.3.2.5 Engineered Barrier System: Waste Packages

The NRC staff reviewed the description of the barrier capability of the waste packages. DOE discussed the capability of the waste package to prevent or substantially reduce contact of seepage with the waste form in SAR Section 2.1.2.2 under the discussion titled "Waste Package," in SAR Sections 2.1.2.2.1 and 2.1.2.2.2, and quantitatively in SAR Section 2.1.2.2.6. DOE (2009an, Enclosures 1 and 4) supplemented this discussion. NRC staff concludes that the capabilities DOE described are consistent with the definition of a barrier at 10 CFR 63.2 because they describe a capability to substantially reduce the rate of water or radionuclide movement.

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

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

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

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

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

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

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

The NRC staff compared the technical bases descriptions in SAR Sections 2.1.2.2 and 2.1.3.2 with SAR Sections 2.3.6.6.3, 2.3.6.3.3, 2.3.6.4.3, 2.3.6.5.3, 2.3.4.5, and 2.3.7.12.3.1 and concludes that the technical bases descriptions are consistent among these SAR sections. Further, the NRC staff compared the quantitative representation of the barrier capability in SAR Sections 2.1.2.2 and 2.1.3.2 with the results of the six waste package TSPA submodels DOE described in SAR Sections 2.3.6.6.3, 2.3.6.3.3, 2.3.6.4.3, 2.3.6.5.3, 2.3.4.5, and 2.3.7.12.3.1 and concludes that the waste package lifetime, waste package failed area, and waste package water flow estimates are consistently represented among these SAR sections. The NRC staff therefore concludes that the technical bases for the description of barrier capability in SAR Section 2.1.2.2 are consistent with the technical bases of the waste package TSPA submodels.

SER Sections 2.2.1.3.1, 2.2.1.3.2, and 2.2.1.3.3 address the NRC staff's evaluation of the adequacy of the technical bases used to support the barrier capability of the waste packages. The NRC staff concludes in SER Sections 2.2.1.3.1, 2.2.1.3.2, and 2.2.1.3.3 that DOE provided technical bases for the waste package TSPA submodels and for the ranges of values for the waste package lifetime, waste package failed area, and waste package water flow used in the performance assessment that are adequate for their intended use. The NRC staff therefore concludes that DOE's technical bases for the description of the barrier capability of the waste packages are commensurate with the barrier capability described in SAR Section 2.1.2.2 and in DOE (2009an, Enclosures 1 and 4).

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

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

The NRC staff reviewed the description of the barrier capability of the waste form and waste package internal components. DOE provided a qualitative description of how the waste form and waste package internal components limit the release of radionuclides from a failed waste package in SAR Section 2.1.2.2 under the discussion titled "Waste Form and Waste Package Internals," in SAR Section 2.1.2.2.2, and in DOE (2009an, Enclosures 1 and 5). DOE described the performance of the waste package internal components quantitatively in SAR Section 2.1.2.2.6. DOE discussed the impacts of these processes in an aggregated fashion, using a metric that indicates the extent to which radionuclides are retained within the entire engineered system over time. Specifically, the approach identifies, for selected radionuclides, the decayed cumulative release from the engineered system (this means, the amount of radionuclides existing within either the lower natural barrier or the accessible environment relative to the total inventory in the entire system). The capabilities DOE described are consistent with the definition of a barrier at 10 CFR 63.2 because DOE described a capability to substantially reduce the rate of release and movement of radionuclides from the waste package.

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

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

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

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

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

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

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

The time period over which the waste form and waste package internals limit the release of radionuclides is described in SAR Section 2.1.2.2.3, in which DOE described the degradation rates of the different waste forms. DOE supplemented this information in DOE (2009an, Enclosure 5), which identified important processes controlling releases at different times in a discussion of selected TSPA model realizations and described how DOE expects the significance of various processes (e.g., solubility, radionuclide sorption) to change with time. SAR Section 2.1.2.2.6 demonstrated the change in the effectiveness of the capability by providing intermediate results from the TSPA model that demonstrated the time-dependent performance of the engineered barrier system to retain selected radionuclides. The NRC staff concludes that this information adequately describes the time period over which the waste form and waste package internal components perform their barrier functions because the significance of different processes at different times is discussed.

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

DOE summarized the technical bases of the barrier capability description of the engineered barrier system, which included the waste form and waste package internal components, in

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

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

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

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

2.2.1.1.3.2.7 Engineered Barrier System: Emplacement Pallet and Invert

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

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

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

The NRC staff reviewed DOE's description of barrier capability for the emplacement pallet and invert. The NRC staff determined that DOE did not include the emplacement pallet as part of the transport path length from waste package to the invert and, as a result, attributed no barrier capability to the emplacement pallet in the performance assessment. Therefore, because DOE did not attribute barrier capability to the emplacement pallet in its performance assessment, the NRC staff concludes that DOE has adequately characterized the emplacement pallet as not important to waste isolation. Based upon examination of the engineered barrier system radionuclide transport abstraction described in SAR Section 2.3.7 and evaluated in SER Section 2.2.1.3.4, the NRC staff determined that the invert does not have a significant delaying effect on radionuclide transport from the waste package to the unsaturated zone below the repository footprint. Therefore, the NRC staff concludes that DOE appropriately considered the invert as not important to waste isolation, because the capabilities of the invert would not have a significant delaying effect on radionuclide transport.

2.2.1.1.3.2.8 Lower Natural Barrier: Unsaturated Zone Below the Repository

The NRC staff reviewed DOE's description of the barrier capability of the unsaturated zone below the repository. DOE identified the unsaturated zone beneath the repository as important to waste isolation because it prevents or substantially reduces the rate of movement of radionuclides (SAR Table 2.1-1). DOE provided a qualitative description of the barrier capabilities of the unsaturated zone below the repository in SAR Section 2.1.2.3.1. In SAR Section 2.1.2.3.6 and in DOE (2009an, Enclosures 1, 6, and 7), DOE quantified the barrier capability with calculations of radionuclide travel times and reduction of radionuclide activity between the repository and the water table. The NRC staff concludes that the barrier capabilities DOE described are consistent with the definition of a barrier at 10 CFR 63.2 because DOE described a capability to substantially reduce the rate of movement of radionuclides from the repository to the water table.

DOE explained in SAR Section 2.1.2.3.1 that downward flow from the repository occurs primarily in well-connected fracture networks in the Topopah Spring welded tuff. DOE explained in SAR Section 2.1.2.3.6 that radionuclides leaving the emplacement drift invert will enter either the repository host rock matrix (primarily under nondripping conditions, where advective flows through the invert are negligible) or the fractures (primarily under dripping conditions, where advective flows through the invert are relatively high). In DOE's unsaturated zone transport abstraction (SAR Section 2.3.8), radionuclides may be retarded by sorption in the matrix but not in fractures. However, radionuclides can migrate from fractures to the rock matrix by matrix diffusion. DOE identified matrix diffusion, coupled with sorption in the matrix, as contributing to barrier performance in the fracture-dominated flow paths. DOE explained in

SAR Sections 2.1.2.3.1 and 2.1.2.3.6 that radionuclide travel times through the lower unsaturated zone are fast in the northern part of the repository area because fracture-dominated flow from the repository host rock encounters a low-permeability, sparsely fractured rock unit, the zeolitic Calico Hills nonwelded tuff, and the flow is diverted laterally along the interface into transmissive faults that connect with the water table. In contrast, DOE stated that in the southern part of the repository area, the fracture-dominated flow from the repository host rock passes into the vitric Calico Hills nonwelded tuff, a permeable rock unit that is dominated by matrix flow conditions. DOE also stated that low flow velocities and the opportunity for sorption in the rock matrix result in long transport times through the unsaturated zone in the southern part of the repository area, particularly for radionuclides that can undergo sorption in the matrix.

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

On the basis of its review of information DOE provided in SAR Section 2.1.2.3, the NRC staff finds that DOE’s description of the barrier capability of the unsaturated zone below the repository is consistent with the results from the performance assessment calculation because DOE described these capabilities using analyses based on TSPA models and input data as well as referring to intermediate results from the performance assessment model.

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

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

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

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

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

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

2.2.1.1.3.2.9 Lower Natural Barrier: Saturated Zone

The NRC staff reviewed the description of the barrier capability of the saturated zone. DOE described the barrier capabilities of the saturated zone qualitatively in SAR Section 2.1.2.3.2 and used median transport times and reduction of radionuclide activity between the water table below the repository footprint and the accessible environment to quantify the barrier capability in SAR Section 2.1.2.3.6 and in DOE (2009an, Enclosures 1, 6, and 7). The NRC staff concludes that the capabilities DOE described are consistent with the definition of a barrier at 10 CFR 63.2 because they describe a capability to substantially reduce the rate of movement of radionuclides from the water table below the repository footprint to the accessible environment.

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

DOE provided quantitative information on the barrier capability of the saturated zone in SAR Sections 2.1.2.3.6 and 2.3.9.3.4.1. SAR Figures 2.3.9-16 and 2.3.9-45 through 2.3.9-47 illustrated the combined effects of matrix diffusion and sorption in delaying radionuclide transport to the accessible environment. Median transport times ranged from about 10 to 10,000 years for nonsorbing radionuclides (e.g., Tc-99) and from 100 to 100,000 years for moderately sorbing radionuclides (e.g., Np-237). Median transport times generally exceeded 10,000 years for highly sorbing radionuclides (e.g., Pu-239). The median transport times for radionuclides irreversibly attached onto colloids ranged from 100 to 600,000 years. In DOE (2009an, Enclosures 6 and 7), DOE used TSPA model results to provide quantitative information on the barrier capability of the saturated zone in terms of reduction of radionuclide activity between the release from the unsaturated zone into the water table and the release into the accessible environment. DOE presented information on the performance of the saturated zone in DOE (2009an, Enclosure 7, Tables 1-5 through 1-8). This information indicates that DOE expects activities of soluble, short half-life radionuclides (e.g., Cs-137 and Sr-90) to drop by 100 percent during transport to the accessible environment. For radionuclides with moderate to strong sorption and long half-life (e.g., Np-237 and Pu-242), DOE calculated the activities to drop by 70 to 98 percent during the 10,000-year period and by 20 to 50 percent during the post-10,000-year period over the transport time in the saturated zone to the accessible environment.

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

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

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

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

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

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

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

2.2.1.1.4 NRC Summary of Evaluation Findings for Multiple Barriers

2.2.1.1.4.1 Identification of Barriers

As discussed in SER Section 2.2.1.1.3.1, DOE has identified specific features and components that are relied upon for repository performance. DOE has linked these features and components to a description of their capabilities. These features and components include at least one barrier from the engineered system and one from the natural system. Therefore, the NRC staff finds that DOE adequately identified the barriers.

2.2.1.1.4.2 Description of Barrier Capability To Isolate Waste Upper Natural Barrier

The NRC staff reviewed the description of the barrier capabilities of the upper natural barrier. This barrier comprises two features: (i) the topography and surface soils at the repository location and (ii) the unsaturated zone above the repository. On the basis of the evaluations documented in SER Sections 2.2.1.1.3.2.1 and 2.2.1.1.3.2.2, NRC staff concludes that the capability of the upper natural barrier has been adequately described. The descriptions are consistent with the definition of a barrier in 10 CFR 63.2 because they address a capability to prevent or substantially reduce the rate of movement of water from the Yucca Mountain repository to the accessible environment. The descriptions of the capabilities are consistent with the results from the total system performance assessment because they are described by using component-specific intermediate results from the performance assessment or by analyses based on models and data used in the performance assessment. Information on the time period over which the upper natural barrier feature performs its intended function has been provided. DOE has adequately addressed uncertainty in the description of barrier capability by identifying sources of uncertainty and describing how these uncertainties affect repository performance. Therefore, the NRC staff finds that DOE has adequately described the capability of the upper natural barrier to prevent or substantially reduce the rate of movement of water or radionuclides from the Yucca Mountain repository to the accessible environment.

Engineered Barrier

The NRC staff reviewed the description of the barrier capabilities of the engineered barrier system. This barrier comprises four components that are important to waste isolation: (i) the emplacement drift, (ii) the drip shield, (iii) the waste package, and (iv) the waste form and waste package internals. On the basis of the evaluation documented in SER Sections 2.2.1.1.3.2.3 through 2.2.1.1.3.2.7, the NRC staff finds that the capability of the engineered barrier system has been adequately described. The descriptions are consistent with the definition of a barrier in 10 CFR 63.2 because they identify a capability to prevent or substantially reduce the rate of movement of water or radionuclides from the Yucca Mountain repository to the accessible environment. The described capabilities are consistent with the results from the total system performance assessment because they are described by referring to component-specific intermediate results from the performance assessment or by analyses based on models and data used by the performance assessment. Information on the time period over which the features of the engineered barrier system perform their intended functions has been provided. DOE has adequately addressed uncertainty in the description of the barrier capability by identifying sources of uncertainty and describing how these uncertainties affect repository performance. Therefore, the NRC staff finds that DOE has adequately described the capability of the engineered barrier system to prevent or substantially reduce the rate of movement of water or radionuclides from the Yucca Mountain repository to the accessible environment.

Lower Natural Barrier

The NRC staff reviewed the description of the barrier capabilities of the lower natural barrier. This barrier comprises two features: the unsaturated zone below the repository and the saturated zone. On the basis of the evaluation documented in SER Sections 2.2.1.1.3.2.8 and 2.2.1.1.3.2.9, the NRC staff concludes that the capability of the lower natural barrier has been adequately described. The described capabilities are consistent with the definition of a barrier in 10 CFR 63.2 because they address a capability to prevent or substantially reduce the rate of movement of radionuclides from the Yucca Mountain repository to the accessible environment. The described capabilities are consistent with the results from the total system performance assessment because they are described by reference to input data or component-specific intermediate results from the performance assessment or by analyses based on models and data used in the performance assessment. Information on the time period over which the features of the lower natural barrier system perform their intended functions has been adequately addressed. DOE has adequately addressed uncertainty in the description of the barrier capability by identifying sources of uncertainty and describing how these uncertainties affect repository performance. Therefore, the NRC staff concludes that DOE has adequately described the capability of the lower natural barrier to prevent or substantially reduce the rate of movement of water or radionuclides from the Yucca Mountain repository to the accessible environment.

2.2.1.1.4.3 Technical Basis for Barrier Capability

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

2.2.1.1.4.4 Evaluation Findings

The NRC staff has reviewed the SAR and other information DOE submitted in support of its license application and finds, with reasonable expectation, that the requirements of 10 CFR 63.113(a) are satisfied. An engineered barrier system has been designed that, working in combination with natural barriers, satisfies the requirement for a system of multiple barriers, in compliance with the postclosure performance objectives. DOE's proposed barrier system includes multiple barrier features and components in the upper natural barrier (the topography and surface soils at the repository location and the unsaturated zone above the repository), engineered barrier (emplacement drift, drip shield, waste package, and waste form and waste package internals), and lower natural barrier (the unsaturated zone below the repository and the saturated zone). The NRC staff concludes that this system of multiple barriers is not wholly dependent on a single barrier because, as described in the evaluation section above, each barrier DOE identified as important to waste isolation has the capability to prevent or substantially reduce the rate of movement of water or radionuclides from the repository to the accessible environment.

The NRC staff has reviewed the SAR and other information DOE submitted in support of its license application and finds, with reasonable expectation, that the requirements at 10 CFR 63.115(a–c) are satisfied. The design features of the engineered barrier system and the natural features of the geologic setting that are considered barriers important to waste isolation have been identified. A description has been provided of the capability of barriers identified as important to waste isolation, and the NRC staff concludes that the description is consistent with the definition of a barrier at 10 CFR 63.2 because it describes a capability to prevent or substantially reduce the rate of water or radionuclide movement. The NRC staff further concludes that the description takes into account uncertainties in characterizing and modeling the barriers, and the technical basis for this description has been provided that is based on and consistent with the technical basis for the performance assessment.

2.2.1.1.5 References

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CHAPTER 2

2.2.1.2.1 Scenario Analysis

2.2.1.2.1.1 Introduction

Safety Evaluation Report (SER) Section 2.2.1.2.1 provides the U.S. Nuclear Regulatory Commission (NRC) staff's evaluation of the U.S. Department of Energy's (DOE or the applicant) scenario analysis used in its performance assessment. The NRC staff evaluated information in the Safety Analysis Report (SAR) (DOE, 2009av) as supplemented by the DOE responses to the NRC staff's requests for additional information (RAIs) (DOE, 2009ab,ae,af,ah–aj,al,bo,bv–bz,ca–cj,co,cq,gp,gq, 2010ad,ah).

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

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

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

A performance assessment is defined, in part, in 10 CFR 63.2 as an analysis that identifies the features, events, processes (except human intrusion), and sequences of events and processes (except human intrusion) that might affect the Yucca Mountain disposal system and their probabilities of occurring. A functional overview of the performance assessment used to demonstrate compliance with the postclosure performance objectives is presented in 10 CFR 63.102(j). Section 63.102(j) also contains criteria for including FEPs on the basis of consequence [those expected to materially affect compliance with 10 CFR 63.113(b) or be potentially adverse to performance] in the performance assessment. Section 63.102(j) provides that events (event classes or scenario classes) which are very unlikely (less than 1 chance in 10,000 over 10,000 years) can be excluded from the analysis on the basis of probability.

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

2.2.1.2.1.2 Regulatory Requirements

The postclosure performance objectives of 10 CFR 63.113 stipulate that a performance assessment must be used to demonstrate compliance with (i) the individual protection standard after permanent closure (10 CFR 63.311); (ii) the human intrusion standard (10 CFR 63.321 and 63.322); and (iii) the separate standards for protection of groundwater (10 CFR 63.331). Requirements for any performance assessment used to demonstrate compliance with 10 CFR 63.113 for 10,000 years after disposal are presented in 10 CFR 63.114(a). Section 63.114(a)(4) requires that the performance assessment consider only FEPs consistent with the limits on performance assessment specified at 10 CFR 63.342. Section 63.114(a)(5)–(6) requires the applicant to provide the technical basis for either inclusion or exclusion of FEPs and also defines criteria for inclusion of the FEPs into the performance assessment [specific FEPs must be evaluated in detail if the magnitude and time of the resulting radiological exposures to the reasonably maximally exposed individual, or radionuclide releases to the accessible environment, for 10,000 years after disposal, would be significantly changed by their omission].

Section 63.113 also requires that the performance assessments used to demonstrate compliance with the individual protection standard after permanent closure, the human intrusion standard, and the separate standards for protection of groundwater must also meet the requirements of 10 CFR 63.342. The limits on performance assessments are defined in 10 CFR 63.342. According to 10 CFR 63.342(a), the performance assessment for 10,000 years after disposal to show compliance with 10 CFR 63.311(a)(1), 63.321(b)(1), and 63.331 shall not include FEPs with less than 1 chance in 100,000,000 per year of occurring. Also, 10 CFR 63.342(a) provides that the performance assessments need not evaluate the impacts resulting from any FEP or sequence of events and processes with a higher chance of occurring if the results of the performance assessments would not be changed significantly in the initial 10,000-year period after disposal.

An additional basis for excluding FEPs in the performance assessments used to demonstrate compliance with 10 CFR 63.321(b)(1) and 63.331 during the first 10,000 years after disposal is provided in 10 CFR 63.342(b). For those performance assessments, 10 CFR 63.342(b) states that unlikely FEPs or sequences of events and processes (i.e., those that are estimated to have less than 1 chance in 100,000 per year of occurring and at least 1 chance in 100 million per year of occurring) can be excluded from the performance assessment.

Section 63.342(c) specifies how to project the continued effects of FEPs beyond 10,000 years in the performance assessment models to show compliance with 10 CFR 63.311(a)(2) and 63.321(b)(2). Section 63.342(c) requires that DOE's performance assessment shall project the continued effects of the features, events, and processes included in 10 CFR 63.342(a) beyond the 10,000-year post disposal period through the period of geologic stability. Section 63.342(c)(1) requires that DOE assess the effects of seismic and igneous activity scenarios, subject to the probability limits in 10 CFR 63.342(a) for very unlikely features, events, and processes, or sequences of events and processes. Section 63.342(c)(1)(i) states that the seismic analysis may be limited to the effects caused by damage to the drifts in the repository, failure of the waste packages, and changes in the elevation of the water table under Yucca Mountain (i.e., the magnitude of the water table rise under Yucca Mountain). Section 63.342(c)(1)(ii) specifies limitations for the igneous activity analysis and igneous event. Section 63.342(c)(2) requires that DOE assess the effects of climate change; it also specifies that the climate change analysis may be limited to the effects of increased water flow through the repository as a result of climate change, and the resulting transport and release of radionuclides to the accessible environment. In addition, 10 CFR 63.342(c)(2) specifies that the nature and degree of climate change may be represented by constant-in-time climate conditions. Section 63.342(c)(3) requires that DOE assess the effects of general corrosion on engineered barriers and specifies that DOE may use a constant representative corrosion rate throughout the period of geologic stability or a distribution of corrosion rates correlated to other repository parameters.

The cited regulations contain criteria for excluding FEPs and scenario classes on the basis of probability or consequence from performance assessments used to demonstrate compliance with the 10 CFR Part 63 standards. Guidance in YMRP Section 2.2.1.2.1.3, p. 2.2-9 provides that specific FEPs and scenario classes can be excluded on the basis that they are specifically ruled out by regulation or are contrary to stated regulatory assumptions. For example, 10 CFR 63.305 defines the required characteristics of the reference biosphere. FEPs that are contrary to these required characteristics can be excluded.

The NRC staff's evaluation of the applicant's scenario analysis in the SAR and other information submitted in support of the license application, including information required by 10 CFR 63.21(c)(1) and (9), follows the methodologies and acceptance criteria identified in YMRP Section 2.2.1.2.1 (NRC, 2003aa), as supplemented by additional guidance for the period beyond 10,000 years after disposal (NRC, 2009ab). The guidance in YMRP Section 2.2.1.2.1 provides four criteria that DOE may use to demonstrate compliance with 10 CFR 63.114(a)(4)–(a)(6).

- The identification of a list of FEPs is adequate.
- Screening of the list of FEPs is appropriate.
- Formation of scenario classes using the reduced set of events is adequate.
- Screening of scenario classes is appropriate.

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

2.2.1.2.1.3 Technical Review

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

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

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

Identification of a list of FEPs is the initial step in the scenario analysis and is aimed at assembling a list that includes all FEPs with the potential to influence repository performance. The NRC staff's technical review of the identification of the list of FEPs follows the methodology established in YMRP Section 2.2.1.2.1.2.

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

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

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

NRC Staff's Review

The NRC staff evaluated the adequacy of the list of FEPs. The FEPs were classified by technical area (Leslie, 2010aa), following a similar approach as used in NRC (2005aa, Table 5.1.2.1-2). In numerous instances, the same FEP was classified as pertaining to several technical areas, to cover broad aspects, consequences, and couplings associated with that FEP (Leslie, 2010aa). The objective of assigning an FEP to multiple technical areas was to attain a thorough and integral review of the list of FEPs covering multiple technical perspectives and to facilitate identifying aspects potentially overlooked by the existing FEPs. The NRC staff evaluated the description of the scope for the individual FEPs, the screening decision of the individual FEPs, the technical basis for excluding FEPs, and the disposition of the included FEPs. The NRC staff's review of the identification of the list of FEPs was based on knowledge gained reviewing the Yucca Mountain site and regional characterization data, including previous independent Yucca Mountain-related studies and prelicensing interactions (documented, for example, in NUREG–1762; NRC, 2005aa), and DOE's description of the modes of degradation, deterioration, and alteration of the engineered barriers. An NRC staff's prelicense application review of DOE's identification of FEPs (Pickett and Leslie, 1999aa) was also considered. The NRC staff also used available, internationally developed generic lists of FEPs (Nuclear Energy Agency, 1997ab) to determine the completeness of the DOE list of FEPs.

The NRC staff found in NRC (2005aa, Section 5.1.2.1.4.1) that the FEPs list for site recommendation was based on a Nuclear Energy Agency international database of FEPs

(Nuclear Energy Agency, 1997ab). Using SNL (2008ab, Appendix G), a cross comparison of the FEPs lists for site recommendation and license application, the NRC staff verified that the FEPs list for the license application appropriately encompasses the FEPs list for site recommendation. Using SNL (2008ab, Appendix F), tables that map the license application FEPs into the Organisation for Economic Co-operation and Development FEPs and vice versa, the NRC staff confirmed that the license application list of FEPs encompasses the Organisation for Economic Co-operation and Development features, events, and processes.

Because the FEPs list for the license application encompasses generic comprehensive lists of internationally approved FEPs (Nuclear Energy Agency, 2006aa, 2000aa, 1997ab) and is consistent with the site characterization data and the license application design features, the NRC staff finds DOE's list of FEPs acceptable.

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

Based on the information in SAR Section 2.2.1.1.1 and references cited therein, the NRC staff finds that

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

- The list of FEPs in SAR Table 2.2-5 is consistent with the site characterization data

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

Therefore, the NRC staff finds acceptable the DOE's identification of a list of FEPs.

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

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

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

In SAR Section 2.2.1.2 DOE developed criteria for exclusion on the basis of low probability, of low consequence, or by regulation. DOE used the NRC's proposed 10 CFR 63.342(a)

(NRC, 2005af) as its low probability screening criterion. Proposed 10 CFR 63.342(a) identified that performance assessments shall not include consideration of very unlikely FEPs [i.e., those that are estimated to have less than 1 chance in 10,000 of occurrence within 10,000 years of disposal (less than 1 chance in 100,000,000 per year)]. DOE identified in DOE (2009cb, Enclosure 1) that it considered the probability criterion to screen all types of FEPs, rather than selectively applying the probability criterion to events only (i.e., the probability criterion was also considered for features and processes). DOE used the proposed 10 CFR 63.114(a)(5), 10 CFR 63.114(a)(6), and 10 CFR 63.342(a) (NRC, 2005af) as the basis for its low consequence screening criterion. DOE stated that low consequence means omission of an FEP would not result in a significant change in the magnitude or timing of the radiological exposures to the reasonably maximally exposed individual or radionuclide releases to the accessible environment. "By regulation" means that FEPs can be excluded if they are inconsistent with the characteristics, concepts, and definitions specified in 10 CFR Part 63, as described in SNL (2008ab, Section 6.1).

In SAR Section 2.1.2.2 the applicant described that the regulations require inclusion of certain FEPs in the performance assessment evaluations which are conducted to demonstrate compliance with the individual protection standards for the period after the first 10,000 years following disposal, but within the period of geologic stability. The applicant described the proposed 10 CFR 63.342 requirements (NRC, 2005af). The applicant described that FEPs associated with the requirements were evaluated for inclusion in the appropriate performance assessments. The applicant stated that no changes to screening decisions were necessary to address the inclusion of FEPs specified by the proposed 10 CFR 63.342(c)(1), (2), and (3). The applicant restated this issue in two parts. First, the applicant stated that the FEPs required by regulation to be included in the performance assessments for the period after the first 10,000 years following disposal, but within the period of geologic stability, are also included in the performance assessments for the 10,000 years after disposal. Second, the applicant stated FEPs that are excluded from the performance assessments for the 10,000 years after disposal remain excluded in the performance assessments for the period after the first 10,000 years following disposal, but within the period of geologic stability. In SAR Section 2.1.2.2 the applicant identified the specific included FEPs that address the proposed 10 CFR 63.342(c)(1)(i) regulatory requirement and described that excluded FEP 1.2.03.02.0B, Seismic Induced Rockfall Damages EBS [Engineered Barrier System] Components, was also evaluated with respect to proposed 10 CFR 63.342(c)(1)(i). DOE (2009cb, Enclosure 6) identified that the excluded FEP 1.2.10.01.0A, Hydrologic Response to Seismic Activity, addresses 10 CFR 63.342(c)(1)(i). In SAR Section 2.1.2.2 the applicant also identified the included FEPs that address 10 CFR 63.342(c)(1)(ii), 10 CFR 63.342(c)(2), and 10 CFR 63.342(c)(3).

The applicant in DOE (2009cb, Enclosure 6) performed a detailed comparison between the proposed 10 CFR Part 63 (NRC, 2005af) rule and the final 10 CFR Part 63 rule that became effective on April 13, 2009, and identified material changes in the final rule and how those changes may materially impact the license application. DOE (2009cb, Enclosure 6) specifically discussed (i) the post-10,000-year 10 CFR Part 63 individual protection standard (350 mrem vs. 100 mrem); (ii) arithmetic mean of projected doses; (iii) water table rise due to seismic activity [an additional requirement added to 10 CFR 63.342(c)(1)(i) in the final rule]; (iv) changes to the range of deep percolation rates; and (v) dosimetry. In addition, DOE (2009cb, Enclosure 6, Section 1.6) evaluated the potential impacts of all changes identified in DOE (2009cb, Enclosure 6, Appendix, Table A-1) [e.g., the specific words in 10 CFR 63.342(a) and 10 CFR 63.342(b) for probability of very unlikely events and unlikely events, respectively, changed] and concluded that none of the conclusions in the license application require modification as a result of the final rule.

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

NRC Staff's Review

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

The NRC staff used YMRP Section 2.2.1.2.1.3, Acceptance Criterion 2, as supplemented by additional guidance for the period beyond 10,000 years after disposal (NRC, 2009ab), to evaluate whether the screening of the list of FEPs is appropriate. The YMRP Section 2.2.1.2.1.3 acceptance criterion evaluates (i) whether the applicant identified all FEPs that have been excluded, (ii) whether the applicant provided justification for exclusion of those FEPs, and (iii) whether the applicant provided adequate technical basis for exclusion of those FEPs.

The NRC staff finds that the applicant has identified all FEPs related to either the geologic setting or to the degradation, deterioration, or alteration of engineered barriers (including those processes that would affect the performance of natural barriers) which have been excluded. SAR Table 2.2-5 listed all of the FEPs the applicant considered, and it identified the excluded FEPs. With regard to YMRP Section 2.2.1.2.1.3, Acceptance Criterion 2, Subcriterion (2), the NRC staff finds acceptable the applicant's criteria for exclusion on the basis of low probability, low consequence, or by regulation, because these criteria are consistent with regulatory requirements for scenario analysis discussed in SER Section 2.2.1.2.1.2. The applicant clearly stated in SAR Table 2.2-5 the criterion it applied for exclusion of each FEP; therefore, the NRC staff finds that the applicant has provided acceptable justification for the excluded FEPs (on the basis of low probability, low consequence, or by regulation).

With regard to YMRP Section 2.2.1.2.1.3, Acceptance Criterion 2, Subcriterion (3), the NRC staff's evaluation of the technical basis for the exclusion of FEPs is provided in the remainder of this section. First, the NRC staff evaluates the applicant's information on screening of FEPs for the period after 10,000 years following disposal, but within the period of geologic stability. Then, the NRC staff evaluates the applicant's information on screening of FEPs for the 10,000 years after disposal.

The NRC staff reviewed the applicant's information in SAR Section 2.1.2.2 on the screening of FEPs for the period after 10,000 years following disposal, but within the period of geologic stability using the guidance for the period beyond 10,000 years after disposal (NRC, 2009ab). The NRC staff finds the following applicant statements acceptable because they are consistent with 10 CFR 63.342(c): (i) FEPs that are required by regulation to be included in the performance assessments for the period after the first 10,000 years following disposal, but within the period of geologic stability, are also included in the performance assessments for the 10,000 years after disposal and (ii) FEPs that are excluded from the performance assessments for the 10,000 years after disposal remain excluded in the performance assessments for the period after the first 10,000 years following disposal, but within the period of geologic stability. The NRC staff finds acceptable the list of included FEPs, which the applicant identified in SAR Section 2.1.2.2, that address the proposed 10 CFR 63.342(c)(1)(i), 10 CFR 63.342(c)(1)(ii), 10 CFR 63.342(c)(2), and 10 CFR 63.342(c)(3) requirements, because the FEPs listed are consistent with the requirements of the final 10 CFR 63.342(c)(1)–(3). In particular, the NRC staff finds acceptable the applicant's identification of the two excluded FEPs {FEP 1.2.03.02.0B, Seismic Induced Rockfall Damages EBS [Engineered Barrier System] Components, and FEP 1.2.10.01.0A, Hydrologic Response to Seismic Activity} because they are consistent with the final 10 CFR 63.342(c)(1)(i). The NRC staff finds that the included FEPs identified in SAR Section 2.1.2.2, p.2.2-19–20, and the two excluded FEPs previously mentioned, are acceptable because they are consistent with the final 10 CFR 63.342(c)(1)–(3). Therefore, the NRC staff finds acceptable the DOE's identification of FEPs applicable to the final 10 CFR 63.342(c)(1)–(3).

The NRC staff evaluates the adequacy, and consistency with the regulatory requirements, of the applicant's analyses of included FEPs in SER Sections 2.2.1.3.1–2.2.1.3.14 and Sections 2.2.1.4.1–2.2.1.4.3. The NRC staff evaluates the applicant's technical basis for the exclusion of FEPs in this section.

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

The NRC staff's review is supported by years of prelicensing interactions and application of risk-informed methods. For example, NUREG–1762 (NRC, 2005aa) includes a detailed list of FEPs reviewed by the NRC staff and an appendix documenting the NRC staff's technical comments and corresponding DOE responses during technical exchanges. For the license application review, the NRC staff evaluated the applicant's screening analyses for specific FEPs in relevant technical areas. For instance, FEPs related to water chemistry were reviewed by multidisciplinary teams of NRC staff on Quantity and Chemistry of Water Contacting Engineered

Barriers and Waste Forms, on Degradation of Engineered Barriers, and on Radionuclide Release and Solubility Limits (Leslie, 2010aa). The teams then discussed overlapping FEPs and documented technical questions in RAIs (e.g., NRC, 2009ad,ae). The technical review of a limited number of FEPs (approximately 10 percent) is documented in this SER section. On the basis of the NRC staff's detailed and multidisciplinary review (Leslie, 2010aa), the NRC staff finds that the excluded FEPs which are not discussed in this section (i.e., the remaining 90 percent of the excluded FEPs) were adequately defined and that adequate technical bases were provided to support the applicant's exclusion decision.

The next three sections describe (i) the review that supports the NRC staff's conclusion and (ii) examples supporting the NRC staff's conclusion for FEPs that are not discussed in this section and that were excluded by regulation, probability, and consequence.

Exclusion by Regulation

For each FEP that was excluded by regulation and that is not explicitly addressed in SER Section 2.2.1.2.1.3.2, the NRC staff checked to see whether DOE provided an appropriate regulatory citation to exclude the FEP by regulation. The NRC staff also reviewed the adequacy of the technical basis (SNL, 2008ab) for each FEP by comparing the technical basis asserted by the applicant to the cited regulation and ensuring the technical basis was consistent with the cited regulation. The NRC staff finds for those FEPs that the applicant excluded by regulation, the applicant's screening basis and technical justification is consistent with applicable regulations. For example, the NRC staff finds that DOE adequately excluded FEP 1.1.10.00.0A, Administrative Control of the Repository Site, on the basis of the regulation, by citing 10 CFR 63.102(k). Section 10 CFR 63.102(k) states that it is not appropriate to include consideration of human intrusion into a performance assessment evaluation for purposes of addressing compliance with 10 CFR 63.113(b). Instead, 10 CFR 63.102(k) requires consideration of human intrusion in a stylized manner, which does not give credit to passive or active institutional controls, consistent with the applicant's rationale to exclude FEP 1.1.10.00.0A. Similarly, for the other FEPs excluded on the basis of the regulation, as described earlier in SER Section 2.2.1.2.1.3.2, the NRC staff conducted a detailed and multidisciplinary review and found that the applicant provided adequate technical bases.

Exclusion by Probability

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

Exclusion by Low Consequence

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

In addition to the screening justifications previously mentioned, the NRC staff determines that the use of repository design and controlled parameters is also acceptable as a technical basis to complement the screening justifications for the following reasons. The NRC staff finds the applicant’s commitment to control parameters identified in SAR Table 1.9-9, through use of management systems, provides a basis for the repository design considered in the development of screening justifications for FEPs. The NRC staff also finds that SAR Table 2.2-3 is an adequate mechanism to track interdependencies and identify FEPs with screening technical bases which would need reevaluation if some parameters depart from initial design considerations. Furthermore, the NRC staff finds acceptable the applicant’s use of repository design and controlled parameters to define the scope of FEPs, as well as to define the initial states or boundary conditions of systems analyzed in the performance assessment. The NRC staff concludes that the design information and the design assumptions are appropriate to develop performance assessments to demonstrate compliance with 10 CFR 63.113. The NRC staff also concludes that the extent of information the applicant provided in regard to repository design for postclosure performance assessments is consistent with the performance-based approach of 10 CFR Part 63.

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

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

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

NRC Staff's Review

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

The NRC staff finds that DOE's previously listed considerations and assumptions [assumption (i): an open borehole without grout acts as a capillary barrier to unsaturated flow; assumption (ii): the cross-sectional area of the rock bolt borehole, onto which flow may be incident, is small; and assumption (iii): water that may have flowed into the borehole can imbibe back into the rock matrix elsewhere along the borehole length] acceptable as bases for supporting the exclusion of the FEP for the following reasons. DOE provided the technical bases supporting the considerations and assumptions in a discussion of the results from the applicant's seepage modeling exercise (SNL, 2008ab; DOE, 2009cb, Enclosures 2 and 7). The NRC staff evaluated the technical bases and analyzed the consistency of observations from field tests and site characterization with results from the DOE seepage modeling exercise for boreholes. First, observations of temperature fluctuations from the heater tests may be indicative of water flowing in boreholes at host-rock temperatures above boiling (Green, et al., 2008aa). Second, post-test visual observations indicate water entered the drifts and did not absorb back into the wall of the borehole, though the timing and temperature at which this occurred is not known (Green, et al., 2008aa). Third, secondary mineralization in large aperture (open) fractures, which DOE attributed to percolating water under ambient conditions, suggests capillary diversion may not keep water from entering boreholes. Fourth, observations of liquid water in the drift during the passive test may be explained and modeled as vapor flux through fractures from within the host rock and condensation in cooler rock spots (Salve and Kneafsey, 2005aa) rather than by a capillarity-based seepage model of liquid water dripping into drifts.

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

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

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

FEP 1.1.03.01.0A, Error in Waste Emplacement

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

NRC Staff's Review

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

FEP 1.2.04.07.0B, Ash Redistribution in Groundwater

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

The DOE volcanic eruption model considers the mass and types of waste impacted by erupted magma (a maximum of seven damaged waste packages), the fraction of waste-containing magma that is incorporated into a tephra plume, and the fraction of the tephra plume that is deposited near the eruptive vent (i.e., in or near the repository footprint) (SNL, 2008ag). In contrast, the DOE igneous intrusion model assumes that (i) all waste packages in the repository are compromised by an igneous intrusion and (ii) the subsequent release of waste is not reduced by the amount which could be transported to the surface in an accompanying eruption (SNL, 2007ab). In excluding ash redistribution in groundwater as a potential FEP, DOE reasoned that because eruptive events are always associated with intrusive events in the DOE model, the potential dose consequences from radionuclides leached into groundwater from volcanic ash would be small compared with the consequences of exposing the entire inventory of radionuclides to seepage and transport in groundwater from the repository (SNL, 2008ab).

In the DOE volcanic eruption modeling case, short-lived, high-activity radionuclides, such as Cs-137, Sr-90, Am-241, and Pu-238, which have half-lives on the order of decades or hundreds of years, are important contributors to dose (e.g., SAR Figure 2.4-32). In DOE (2009ab, Enclosure 1), the applicant supplemented its technical basis for exclusion of ash redistribution in groundwater with a supporting calculation to assess the effect of leaching and shallow groundwater transport of contaminated ash deposited near the accessible environment boundary. The applicant's supporting calculation addressed differences in travel times depending on where the contaminated ash was deposited within the drainage basin of Fortymile Wash (i.e., very short flow paths for leaching of ash deposited near the accessible environment boundary and longer flow paths for ash deposited upstream). The transport calculation included the effects of radioactive decay and retardation of radionuclides. The calculation results indicated that leaching of contaminated ash would not contribute significantly to mean annual dose compared to the volcanic eruption modeling case, as detailed in DOE (2009ab, Enclosure 1, Figure 1).

NRC Staff's Review

The NRC staff examined the DOE supporting calculations for leaching and transport from contaminated ash deposited in Fortymile Wash, as described in DOE (2009ab, Enclosure 1). The NRC staff's evaluation focused on short-lived radionuclides because of their high radioactivity levels that could dominate dose estimates for this fast pathway scenario. The NRC staff contrasted the applicant's computations for this fast transport pathway scenario with the contribution to dose from the same short-lived radionuclides in the applicant's volcanic eruption modeling case (e.g., SAR Figure 2.4-32) and determined that the applicant adequately demonstrated the technical basis for excluding FEP 1.2.04.07.0B with respect to leaching and transport in short groundwater flow pathways for the following reasons. First, the applicant's supporting calculations demonstrated that the transport of the short-lived radionuclides was delayed sufficiently, even in the relatively short groundwater transport pathways in Fortymile Wash, to allow radioactive decay to significantly diminish their potential contribution to dose. Second, the models the applicant used for the supporting calculations are consistent with those models that the NRC staff reviewed for water flow paths and radionuclide transport in SER Sections 2.2.1.3.8 and 2.2.1.3.9, which the NRC staff found to be adequate in the context of the applicant's performance assessments. Third, the potential dose consequences near the eruptive vent (i.e., farther from the accessible environment boundary) are bounded by the dose consequences of igneous intrusion because the groundwater transport pathways are similar for both the volcanic and igneous examples. Therefore, the NRC staff concludes that the model the applicant used in the supporting calculations to evaluate potential fast pathways is adequate. Therefore, the NRC staff finds acceptable the applicant's technical basis for excluding FEP 1.2.04.07.0B on the basis of low consequence.

FEP 1.2.07.01.0A, Erosion/Denudation

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

The applicant cited site characterization studies concluding erosion would range from 0.4 to 2.7 cm [0.16 to 1.06 in] in 10,000 years for bedrock outcrops and 0.2 to 6 cm [0.08 to 2.4 in] in 10,000 years for unconsolidated material in hillslopes. The applicant stated that the maximum expected erosion of 6 cm [2.4 in] in 10,000 years is consistent with existing surface irregularities and that erosion would be negligible compared with the minimum distance of 200 m [656.2 ft] from the ground surface to the repository emplacement areas (SNL, 2008ab). The applicant considered the effect of erosion on the extent of net infiltration and determined that any shortening of the flow path length due to erosion would be negligible. The applicant described that the homogenizing action of the Paintbrush nonwelded hydrogeologic unit would buffer any localized change in net infiltration (SNL, 2008ab). Moreover, the applicant stated that bedrock weathering in some cases could increase the soil thickness, which would decrease, rather than increase, net infiltration. The applicant also cited site characterization studies that determined processes such as landslides and debris flows do not play a significant role in the erosional regime at Yucca Mountain.

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

NRC Staff's Review

The NRC staff reviewed the supporting information and the analysis the applicant developed for exclusion of FEP 1.2.07.01.0A. The NRC staff concludes that the technical basis for excluding

FEP 1.2.07.01.0A, Erosion/Denudation, is acceptable because erosion rates the applicant cited are consistent with the site description data at BSC (2004bi) and are expected to cause negligible amounts of erosion in 10,000 years. The NRC staff finds acceptable the applicant's conclusion that neglecting the effects of erosion in the performance assessment would not significantly affect the timing or magnitude of radionuclide releases into the accessible environment. Therefore, the NRC staff finds acceptable the exclusion of FEP 1.2.07.01.0A, Erosion/Denudation, on the basis of low consequence.

FEP 1.2.10.01.0A Hydrologic Response to Seismic Activity

The applicant excluded Hydrologic Response to Seismic Activity on the basis of low consequence (SNL, 2008ab). The technical basis for the exclusion of this FEP was supplemented as described in DOE (2009ab, Enclosure 19; 2009by, Enclosures 1–6; 2009bz, Enclosure 1; 2009ca, Enclosures 1–2; 2009cb, Enclosure 6). In supplementing the technical basis for this FEP, DOE also addressed compliance with the water table rise requirement due to seismic activity beyond the 10,000-year post-disposal period through the period of geologic stability [10 CFR 63.342(c)(1)(i)] and included information on potential permanent changes in hydrologic properties (DOE, 2009cb, Enclosure 6). According to the applicant's definition of FEP 1.2.10.01.0A, seismic activity associated with fault movement may enhance existing or create new flow pathways or connections and barriers between stratigraphic units, or it may change the stress (and therefore fluid pressure) within the rock. These physical changes have the potential to alter groundwater flow directions, water level, water chemistry, and temperature. Seismically induced changes to the local stress fields can cause a transient change in the water table elevations and lead to seismic pumping—a phenomenon the applicant defined as the temporary change in water table elevation resulting from fault movement and the opening and closing of fractures during an earthquake.

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

In SNL (2008ab), the applicant cited modeling investigations that have been conducted to estimate the hydrologic response (i.e., change in water table elevations), given predicted fault displacements (National Research Council, 1992aa, Chapter 5). As described in SNL (2008ab), the National Research Council study estimated, using two fault displacement modeling approaches (i.e., a dislocation approach and a "changes in the regional stress" approach), that the maximum seismically induced water table rise over a 10,000-year period would be 17 m [56 ft] for the dislocation approach and 50 m [160 ft] for the regional stress approach. In addition, SNL (2008ab) described that the hydrologic effects of three seismic events in 1992 which were observed in groundwater monitoring wells at Yucca Mountain provide estimates of water-level fluctuations occurring in response to earthquakes. The applicant examined the effects of the Landers–Big Bear–Little Skull Mountain earthquake sequence that occurred June 28–29, 1992, and indicated the water table rise observed at several Yucca Mountain vicinity monitoring wells ranged from 0.2 to 0.9 m [0.7 to 3 ft]. On the basis of the earthquake-caused water table change data and analyses in the National Research Council study (1992aa), the applicant concluded the maximum change will be no more than a 50-m [160-ft] water table rise beneath the repository.

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

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

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

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

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

NRC Staff's Review

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

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

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

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

Further, the NRC staff finds that the applicant's screening rationale is also applicable to the period of geologic stability because the applicant considered, in general, seismic events with recurrence rates of at least 10^{-8}/yr, as described in DOE (2009cb, Enclosure 6). The National Research Council performed a study in 1995 and concluded that a probable maximum transient would not exceed 20-m (National Research Council, 1995aa, p. 94). This 20-m rise is less than the estimated water table rises in a 1992 study (National Research Council, 1992a). DOE considered this 1992 study in determining the magnitude of the water table rise from seismic activity beyond the 10,000-year post-disposal period through the period of geologic stability (DOE, 2009cb, Enclosure 6). Because the NRC staff determines that analyses by the applicant and those in National Research Council (1992aa) used to estimate seismically induced water table rise overestimate the extent of seismically induced water table rise, and because the applicant's technical basis supports the exclusion of FEP 1.2.10.01.0A from the performance assessment model for the initial 10,000-year period after disposal, the NRC staff finds acceptable the applicant's technical basis to exclude FEP 1.2.10.01.0A from the performance assessment analysis beyond the 10,000-year post-disposal period through the period of geologic stability.

FEP 1.4.01.00.0A, Human Influences on Climate

The applicant excluded Human Influences on Climate on the basis of the exclusion by regulation criterion. The applicant stated (SNL, 2008ab) that the proposed 10 CFR 63.305 (NRC, 2005af) excludes speculative prediction of changes to human behavior.

DOE identified two types of human influences on climate: past and present, and future. The applicant stated that present and past human influences on climate are implicitly included in estimates of modern climate used in the performance assessment (SNL, 2008ab). The past and present human influences on climate are evaluated by the NRC staff in SER Section 2.2.1.3.5. The DOE defined the scope of FEP 1.4.01.00.0A, Human Influences on Climate, to address only future human influences on climate. The applicant excluded the aspect of human influences on climate that depends on future human behavior on the basis of the exclusion by regulation criterion.

Several changes between the proposed and final rule related to treatment of climate change in the performance assessment. The applicant identified (DOE, 2009cb, Enclosure 6) changes between the proposed and final rule, including the change to 10 CFR 63.305(c). DOE did not identify any changes between the proposed and final rule affecting the excluded FEP 1.4.01.00.0A, Human Influences on Climate.

NRC Staff's Review

The NRC staff reviewed the DOE implementation of 10 CFR 63.305, including its analysis of the changes between the proposed and final rule. The NRC staff concludes that the changes in 10 CFR 63.305(c) do not have an effect on the excluded FEP 1.4.01.00.0A. Because the applicant constrained this FEP to changes in climate caused by future changes in human activity, the applicant used the proposed regulation for its exclusion determination and the NRC staff determined that the changes between the proposed and final rule do not have an effect on the exclusion analysis, the NRC staff determines that the applicant's exclusion of this FEP is consistent with the final rule in 10 CFR 63.305. Therefore, the NRC staff finds acceptable the applicant's exclusion of the FEP from the performance assessment on the basis of the regulation.

FEP 1.4.01.02.0A, Greenhouse Gas Effects

The applicant excluded Greenhouse Gas Effects on the basis of the exclusion by regulation criterion. The applicant constrained the scope of the FEP to future changes in human activities that may influence the concentrations of atmospheric gases. The applicant noted that greenhouse gases affect climate.

DOE identified two types of greenhouse gas effects caused by human behavior: past and present, and future. The applicant defined FEP 1.4.01.02.0A, Greenhouse Gas Effects, to include only the changes to greenhouse emissions that may be caused by future changes in human behavior. The applicant stated (SNL, 2008ab) that the proposed 10 CFR 63.305 (NRC, 2005af) excludes speculative prediction of changes by human behavior, including human influence on greenhouse emissions. Present and past increases in greenhouse gases attributed to human activity are implicitly included in estimates of modern climate used in the performance assessment (SNL, 2008ab) and are evaluated by the NRC staff in SER Section 2.2.1.3.5.

Several changes between the proposed and final rule related to the treatment of climate change and greenhouse gas effects in the performance assessment. The applicant evaluated (DOE, 2009cb, Enclosure 6) changes between the proposed and final rule, including the change to 10 CFR 63.305(c). DOE did not identify any changes between the proposed and final rule that affect the excluded FEP 1.4.01.02.0A, Greenhouse Gas Effects.

NRC Staff's Review

The NRC staff reviewed the screening analysis in SNL (2008ab) and the DOE implementation of 10 CFR 63.305, including its analysis of the changes between the proposed and final rule (DOE, 2009, Enclosure 6). The NRC staff concludes that the changes in 10 CFR 63.305(c) do not have an effect on the excluded status of FEP 1.4.01.02.0A. Because the applicant constrained this FEP to changes in greenhouse gases caused by future changes in human activity, the applicant used the proposed regulation for its exclusion determination and the NRC determined that the changes between the proposed and final rule do not have an effect on the exclusion analysis, the NRC staff determines that the applicant's exclusion of this FEP is consistent with the final rule in 10 CFR 63.305. Therefore, the NRC staff finds acceptable the applicant's exclusion of the FEP from the performance assessment on the basis of the regulation.

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

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

The DOE screening analysis for radionuclide recycling in groundwater is based on a model that assumes a single hypothetical water supply well with an uninterrupted withdrawal rate of 3,000 acre-ft [3.7×10^9 L] per year from the center of a contaminant plume. Capture zone dimensions for this hypothetical well are computed based on the local-groundwater-specific discharge and saturated aquifer thicknesses upgradient and downgradient of the well. The applicant considered three mechanisms by which radionuclides can be lost from the recycling process: (i) irrigation water usage on fields located outside of the capture zone, (ii) residential water usage at locations outside of the capture zone, and (iii) erosion of soil from irrigated fields to locations outside of the recycling system. On the basis of current water usage in Amargosa Valley, about 90 percent of withdrawn water is used for irrigation. The applicant's screening analysis concludes that recycling could increase the total mean annual dose by approximately 7 to 11 percent for the seismic ground motion and igneous intrusion scenarios for the 1-million-year simulation period and by an average of 11 percent for the 10,000-year simulation period (SNL, 2008ab), which is not significant compared with the range of uncertainty simulated by the total system performance assessment model. On the basis of this result, the applicant excluded FEP from the performance assessment.

NRC Staff's Review

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

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

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

To evaluate the case where the well intersects the highest concentration in the plume and irrigated fields are in proximity to the well, the NRC staff considered a hypothetical case where a well was used to irrigate a number of fields. The NRC staff considered the well located within the accessible environment and above the maximum plume concentration. If the irrigated fields were distributed in space at random, half of the fields would be located upstream from the well and half downstream. As an approximation, the NRC staff considered that upstream fields would be within the well capture zone and downstream fields outside the well capture zone. Therefore, in this simplified assessment, if pumping were to continue indefinitely with no soil erosion losses, a maximum of 50 percent of the radionuclides in the irrigated water could be recycled, causing concentrations of radionuclides to double at most. The NRC staff considers that a factor-of-two increase in the concentrations and dose consequences is a relatively moderate effect, because this simplified analysis represents a hypothetical case and the applicant’s mean dose estimates are well below the 10 CFR Part 63 individual protection standard. (Note that in SER Section 2.2.1.2.2.3.1, the NRC staff considered uncertainties affecting dose estimates for the igneous scenario by a factor of two were not risk significant, given the large margin to the regulatory limit.) On the basis of its review of the applicant’s analysis and the NRC staff's independent risk insights, the NRC staff finds acceptable the applicant's technical basis to exclude the FEP from the performance assessment on the basis of low consequence.

FEP 2.1.03.02.0B, Stress Corrosion Cracking of Drip Shields

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

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

NRC Staff's Review

The NRC staff finds that the proposed stress-relieving process conditions are consistent with recommended industry practice (ASM International, 2003aa) to reduce residual stresses. Therefore, stress corrosion cracking of the drip shield is unlikely to occur because of weld-induced residual stresses. The thermal expansion of drip shield joints may cause residual stresses; however, the applicant stated that drip shield connectors are designed to allow for thermal expansion with no effect on drip shield performance up to 300 °C [572 °F]. The thermal expansion coefficient of Titanium Grades 7 and 29 is 9.2×10^{-6} K^{-1} and 9.5×10^{-6} K^{-1}, respectively (ASM International, 1994aa). The NRC staff finds acceptable the applicant's conclusion that thermal expansion will not cause any significant tensile stresses, including stresses to induce stress corrosion cracking, because the drip shield temperature will remain below 300 °C [572 °F]. The NRC staff also finds this conclusion acceptable for the following reasons: (i) DOE quantified the additional number of waste packages that could fail by stress corrosion cracking, as a consequence of seepage infiltrating the failed drip shields, following an approach consistent with the waste package localized corrosion model evaluated in SER Section 2.2.1.3.1 and the seismic consequence abstraction model evaluated in SER Section 2.2.1.3.2; (ii) DOE concluded that the additional number of failed waste packages would be less than the number of failed waste packages for the early failure and seismic fault displacement modeling cases; and (iii) given that the contribution to the total dose of these latter cases is minimal, DOE adequately concluded the dose contribution from the additional failed waste packages by stress corrosion cracking would be negligible. In addition, the applicant pointed out in DOE (2009ab, Enclosure 2, Sections 1.2 and 1.6) that volumetric flow through open (unplugged) cracks is expected to be smaller than the seepage flow approaching drip shields. The NRC staff finds the applicant's conclusion on the flow reduction acceptable, because (i) openings can act as capillary barriers to seepage water under unsaturated conditions and (ii) DOE provided experimental evidence for the flow reduction through cracks in DOE (2009ab, Enclosure 2, Figure 1). The applicant provided the same justifications to also exclude FEP 2.1.03.10.0B, Advection of Liquids and Solids Through Cracks in the Drip Shield, and the NRC staff also finds the technical basis to be adequate for this other FEP as described later in this SER Section. In summary, the NRC staff finds acceptable the applicant's technical basis to exclude both FEPs 2.1.03.02.0B, Stress Corrosion Cracking of Drip Shields, and 2.1.03.10.0B, Advection of Liquids and Solids Through Cracks in the Drip Shield, on the basis of low consequence.

FEP 2.1.03.03.0B, Localized Corrosion of Drip Shields

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

NRC Staff's Review

The NRC staff reviewed the technical basis (SNL, 2008ab) and supporting information provided by the applicant (BSC, 2004as; DOE, 2009ab, Enclosures 2−6). The NRC staff examined the applicant's model assumptions and model support in the area related to localized corrosion of the drip shield. The NRC staff finds that the 2.5- and 5-year testing the applicant conducted indicates the possibility for localized corrosion of the drip shield is small in the potential repository environment. The NRC staff performed independent analyses that indicate the concentration of fluoride, which at higher levels increases the localized corrosion susceptibility, would not likely achieve high levels in the proposed repository (Lin, et al., 2003aa; Pabalan, 2010aa). The independent analyses indicate that fluoride precipitates with common chemicals in the groundwater, limiting the concentration of free fluoride ions in the water. Independent localized corrosion analyses of Titanium Grade 7 support the conclusion that localized corrosion of Titanium Grade 7 is not likely under repository conditions (Brossia and Cragnolino, 2000aa). The NRC staff's evaluation of fluoride effects and long-term immersion tests are provided in SER Section 2.2.1.3.1.3.1.1.

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

The NRC staff, in SER Section 2.2.1.3.2, evaluated the ability of the drip shield to maintain its seepage barrier function if the side framework, made of Titanium Grade 29 and welded to Titanium Grade 7 using Titanium Grade 28 as filler metal, buckled. On the basis of that evaluation, the NRC staff finds acceptable the applicant's conclusions that the drip shield would continue to function and protect the waste package against seepage flowing through the Titanium Grade 7 plates, and that localized corrosion of other drip shield parts would not have a significant effect on dose calculations. Therefore, the NRC staff finds acceptable the applicant's technical basis for exclusion of the FEP, Localized Corrosion of Drip Shields, on the basis of low consequence.

FEP 2.1.03.04.0B, Hydride Cracking of Drip Shields

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

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

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

NRC Staff's Review

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

The applicant excluded Hydride Cracking of Drip Shields from the performance assessment model on the basis of low probability (SNL, 2008ab). The applicant updated the technical basis to show that the probability of hydride cracking of drip shields is less than 10^{-4} in 10,000 years, as detailed in DOE (2009cb, Enclosure 1). The applicant described that even with a high corrosion rate at a probability level of 2.5×10^{-5} (applied for 10,000 years), the hydrogen concentration would be below the critical hydrogen concentration for hydride cracking. The NRC staff finds the exclusion of hydride cracking of the drip shields from the performance assessment on the basis of low probability acceptable because the applicant (i) considered high corrosion rates leading to overestimating the amount of hydrogen produced from the general corrosion process, (ii) mitigated hydrogen diffusion through selection and control of titanium alloy material and weld filler metal, and (iii) demonstrated through analysis that the hydrogen concentration would not be sufficient to induce hydride cracking on the drip shield plate and frame.

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

The applicant excluded the Advection of Liquids and Solids Through Cracks in the Drip Shield from the performance assessment model on the basis of low consequence (SNL, 2008ab) and supplemented its technical basis for exclusion in DOE (2009ab, Enclosure 2). According to the applicant's definition of the FEP, if cracks develop on the drip shield, water could flow through those cracks and contact the waste package. The applicant presented technical reasons for excluding the potential of advective flow of water through cracks in a drip shield. These involved (i) creep/stress relaxation in a drip shield (of Titanium Grade 7) could limit the development and penetration of stress corrosion cracks; (ii) a small damaged area (less than 0.5 percent) on the drip shield surface from seismic-induced rockfall could limit the surface area available for advective flow of seepage water; (iii) a low chance of large rockfall from the lithophysal rock zone above the drip shield could cause sufficient stress corrosion cracks and denting of a drip shield; (iv) a low chance of large rock-block falls from the nonlithophysal rock zone above the drip shield could occur due to low probability of seismic events of sufficient magnitude; (v) potential filling and plugging of stress corrosion cracks by mineral precipitates and corrosion products could potentially limit the advective flow of water through a drip shield; (vi) a low chance of perfect alignment of tight and tortuous cracks on a drip shield surface could occur with impinging seepage drips from the drift wall; (vii) in the absence of drip shields, in less than 10 percent of the waste packages, localized corrosion would be initiated (SNL, 2008ag, Appendix O); and (viii) if leakage through a crack-damaged drip shield caused localized corrosion of the waste package, only a small flux {4 mL/yr [0.244 in^3/yr]} would directly flow into the waste package, which would be insignificant from the repository performance standpoint (SNL, 2008ab). Therefore, DOE excluded the FEP from the performance assessment model on the basis of low consequence.

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

NRC Staff's Review

The NRC staff reviewed the technical basis (SNL, 2008ab) and supporting information provided by the applicant (BSC, 2004as; DOE, 2009ab, Enclosure 2), which are applicable to both this FEP and FEP 2.1.03.02.0B, Stress Corrosion Cracking of Drip Shields. The NRC staff finds the technical basis for exclusion of this FEP on the basis of low consequence to be adequate for the same reasons provided under FEP 2.1.03.02.0B, Stress Corrosion Cracking of Drip Shields. In its review of FEP 2.1.03.02.0B, the NRC staff noted that DOE quantified the additional number of waste packages which could fail by stress corrosion cracking as a consequence of seepage infiltrating the failed drip shields. DOE concluded that the additional number of failed waste packages would be less than the number of failed waste packages for the early failure and seismic fault displacement modeling cases. Given that the contribution to the total dose of these latter cases is minimal, DOE concluded the dose contribution from the additional failed waste packages by stress corrosion cracking would be negligible. In addition, the applicant pointed out in DOE (2009ab, Enclosure 2, Sections 1.2 and 1.6) that volumetric flow through open (unplugged) cracks is expected to be smaller than the seepage flow approaching drip shields. The NRC staff finds the applicant's conclusion on the flow reduction acceptable, because (i) openings can act as capillary barriers to seepage water under unsaturated conditions and (ii) DOE provided experimental evidence for the flow reduction through cracks in DOE (2009ab, Enclosure 2, Figure 1). Therefore, the NRC staff finds acceptable the applicant's technical basis to exclude FEP 2.1.03.10.0B, Advection of Liquids and Solids Through Cracks in the Drip Shield, on the basis of low consequence.

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

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

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

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

NRC Staff's Review

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

FEP 2.1.06.06.0B, Oxygen Embrittlement of Drip Shields

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

NRC Staff's Review

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

FEP 2.1.07.02.0A, Drift Collapse

As defined by the applicant, this FEP considered nonseismic drift collapse; specifically, the degradation of emplacement drifts that may result from the combination of excavation-induced rock stress and thermal loading in the absence of significant seismic events. DOE considered seismically induced drift collapse as a separate FEP that was included in its performance assessment evaluation. Seismically induced drift collapse is reviewed as a model abstraction in SER Section 2.2.1.3.2 and, hence, is not addressed in this subsection.

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

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

The applicant stated that drift degradation could occur rapidly if the stress change is large enough to cause instantaneous rock failure or gradually if the stress change is too small to cause rapid failure but large enough to weaken the rock with time. DOE summarized its basis for excluding drift collapse in SNL (2008ab). The applicant concluded that nonseismic drift degradation would cause only minor, localized rockfall that results in insignificant impact on the thermal and hydrologic conditions of the drift and minimal consequences to the engineered barrier system components.

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

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

NRC Staff's Review

The NRC staff evaluated the applicant's analyses and calculations supporting its screening basis and its use of bounding or representative estimates for the consequences. The NRC staff performed independent calculations using analytical tools and numerical models to scope potential issues and to verify or confirm the applicant's conclusions (Cao, 2010aa). In evaluating the applicant’s technical basis for excluding the FEP, the NRC staff also considered its independent prelicensing analysis (Ofoegbu, et al., 2007aa).

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

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

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

Characterization of Rock Mechanical and Thermal Properties

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

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

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

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

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

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

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

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

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

Model Domain and Boundary Conditions

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

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

The temperatures and stresses calculated by the drift-scale model (NUFT-FLAC results), in which simplified rigid boundary conditions (zero displacement) are assumed for the vertical and bottom boundary planes, were compared with the coupled, three-dimensional, regional- and drift-scale model (FLAC3D results). The comparison demonstrated that the simplified rigid boundary condition used in the two-dimensional drift-scale model resulted in higher horizontal stresses compared to the three-dimensional regional model, especially at the repository edge

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

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

NRC Staff's Review of Model Domain and Boundary Conditions

The NRC staff reviewed the details of the applicant's numerical models and related calculations used to determine boundary conditions. The NRC staff finds that the computer codes (NUFT-FLAC and UDEC) the applicant used in the thermal-mechanical boundary calculations are well tested and widely used in geotechnical industries and research communities.

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

The NRC staff's confirmatory calculation with the rigid boundary condition shows that horizontal stresses are overestimated for drifts near the edge of the repository in DOE's calculations (BSC, 2004al). By using a fixed boundary condition for the UDEC–Voronoi model, DOE does not allow for potential horizontal expansion to reduce the accumulation of horizontal stress from thermal expansion of the rock. The NRC staff finds the use of fixed vertical boundaries in the DOE model acceptable, because this assumption will not underestimate the potential effects of thermal stress on rocks surrounding the heated drifts.

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

Initial Stress State and Temperature Inputs

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

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

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

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

Block Size and Shape in the Voronoi Domain

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

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

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

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

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

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

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

Parameter Uncertainty and Model Calibration

In BSC (2004al, Sections 7.6.3 and 7.6.4), DOE described the approach used to calibrate the Young's modulus and unconfined compressive strength of the rock mass modeled to the expected characteristics of the lithophysal rock. Five rock-strength categories were considered in the calibration to represent the range of values for estimated Young's modulus in the lithophysal rock. For each of the five rock-strength categories considered, DOE used a mean value for unconfined compressive strength as indicated in BSC (2004al, Appendix E,

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

A potentially important uncertainty in the DOE model is the representation of spatial variability in rock properties. DOE addressed this uncertainty by developing calibrated models for five different rock-strength categories, which are distinguished by different values of rock mass modulus. In conducting its calibration, DOE used the mean value of unconfined compressive strength as the calibration target for each selected value of rock mass modulus. DOE data presented in SAR Figure 2.3.4-30, on the other hand, showed a range of potential values of unconfined compressive strength for a given value of rock mass modulus [also discussed in detail in BSC (2004al, Appendix E, Figure E-13)]. DOE stated (SAR p. 2.3.4-73) that a number of parametric studies were conducted in which the Young's modulus and strength parameters were varied to account for the reasonable bounding ranges of lithophysal and nonlithophysal rocks. However, a minimum value of 10 MPa was chosen for the unconfined compressive strength in DOE's parametric studies as explained in DOE (2009cd, Enclosure 1).

NRC Staff's Review of Parameter Uncertainty and Model Calibration

The NRC staff reviewed the results of DOE's analyses using the lower bound strength and Young's modulus for rock mass Categories 2, 3, 4, and 5, as outlined in DOE (2009cd, Enclosure 1). The NRC staff finds that not including Category 1 rocks in the parametric analyses is acceptable because Category 1 rocks only constitute a small percentage of lithophysal rock mass. The NRC staff also finds acceptable that DOE did not consider lower unconfined compressive strengths and chose 10 MPa as the minimum unconfined compressive strength for Categories 2 and 3 in its parametric studies, because the observed large-scale rock behavior in the Exploratory Studies Facility (ESF) and the Enhanced Characterization of Repository Block (ECRB) Cross Drift supports DOE's approach. DOE's statement that ESF and ECRB drifts would have shown extensive damage and significant amounts of rockfall if the unconfined compressive strengths were closer to the lower bound values than the mean values (i.e., much less than 10 MPa) is, therefore, acceptable. Further, the NRC staff compared the results of the analyses using lower bound values with analyses that used mean values and concludes there is no significant difference in the amount of rockfall calculated using either parameter set, as shown in DOE (2009cd, Enclosure 1, Figures 9–12). Therefore, the NRC staff concludes DOE has acceptably accounted for uncertainty in rock properties that are important in the UDEC–Voronoi model.

The NRC staff also reviewed DOE's calibration of the Voronoi block model, which showed that the model was capable of reproducing the axial splitting mode of failure of a typical unconfined compressive strength test conducted in the laboratory (BSC, 2004al). In addition, the model was able to duplicate the stress-strain curves, including postpeak failure behavior and brittle response of the test specimen. The NRC staff finds that the calibration demonstrated in DOE (2009cd, Enclosure 1,Figures 2–8) supports the use of the model for the parametric analyses as well as for calculating the amount of rockfall as a time- and temperature-dependent process.

Model Results

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

- The combined *in-situ* and thermal mechanical stress reached in the drift crown is about 7 MPa for Category 1 and about 37 MPa for Category 5 lithophysal rocks, respectively, as shown in BSC (2004al, Figures 6-142 and 6-144).
- These stress values can, in some conditions, slightly exceed the unconfined compressive strength of lithophysal rock.
- The elastic stress paths cover a time range of 10,000 years' variation of temperature.
- The amount of thermally induced rockfall is small for all five categories of lithophysal rocks.
- Basic rock mechanics principles show that the potential for the thermally induced rockfall process should cease at a short distance from the drift wall into the rock, where the confined strength of the rock is greater than the sum of mechanical and thermal stresses.

NRC Staff's Review of Model Results

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

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

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

NRC staff reviewed these results and concludes the UDEC–Voronoi model is capable of representing block failure in the overstressed zone for average block sizes that range from 4 to 30 cm [1.6 to 11.8 in].

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

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

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

Model Support and Consistency With Available Observations

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

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

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

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

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

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

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

Treatment of Time-Dependent Failure

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

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

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

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

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

Alternative Conceptual Models

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

NRC Staff's Review of Alternative Conceptual Models

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

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

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

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

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

DOE accounted for variability in rock types and a range of mechanical properties and strength characteristics, on the basis of laboratory tests and field investigations. DOE has presented acceptable technical bases for its conclusions that rockfall in lithophysal rocks, with natural fractures and weak planes along which preferential failures occur, can be evaluated by the discontiniuum models. The NRC staff finds the applicant has used acceptable technical approaches for quantifying the amount of rockfall that potentially results from nonseismically induced drift collapse. The NRC staff finds DOE's methods acceptable to quantify the amount of thermally induced rockfall. Therefore, the NRC staff finds DOE's conclusion that combined effects of mechanical, thermal, and time-dependent weakening of rocks can be excluded from its performance assessment is adequately supported. On the basis of the results of this review, the NRC staff finds that the DOE technical basis for excluding FEP 2.1.07.02.0A, Drift Collapse, is acceptable.

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

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

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

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

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

NRC Staff's Review

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

The NRC staff reviewed the applicant's methodology to develop equations to represent the creep of Titanium Grades 7 and 29 at room temperature. With respect to Titanium Grade 7, the NRC staff finds adequate the applicant's approach to consider published empirical creep data

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

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

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

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

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

The NRC staff reviewed the applicant's finite difference structural analyses on creep deformation of the drip shield exposed to six loading scenarios (BSC, 2004al) presented in BSC (2005an). In these analyses, the applicant considered the highest vertical pressure applied to the drip shield crown of 154.81 kPa [22.45 psi]. The NRC staff finds acceptable the range of loads the applicant considered. This is further addressed in SER Section 2.2.1.3.2. In summary, the NRC staff finds that (i) DOE did not underestimate the amount of creep strain in its analysis for the postclosure period; (ii) DOE developed acceptable equations to represent the creep of Titanium Grades 7 and 29 at room temperature; (iii) DOE's methodology to rescale the room-temperature creep equations for Titanium Grades 7 and 29 to 150 °C [302 °F] is acceptable; (iv) DOE's use of 10 percent strain is an acceptable threshold for the onset of tertiary creep because it does not underestimate the threshold strain; and (v) DOE considered an acceptable range of loads. On the basis of the results of this review, the NRC staff finds that the DOE technical basis for excluding FEP 2.1.07.05.0B, Creep of Metallic Materials in the Drip Shield, is acceptable.

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

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

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

NRC Staff's Review

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

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

In the applicant's assessment of the dependence of volume increase of corrosion products on outer container corrosion, the applicant considered information on Alloy 22 general corrosion rates. The NRC staff finds that these corrosion rates are consistent with the applicant's general corrosion model evaluated in SER Section 2.2.1.3.1.3.2. Based on the NRC staff's findings that the laboratory test results and models for long-term prediction were adequate (SER Section 2.2.1.3.1.3.2), the NRC staff concludes the Alloy 22 corrosion rates are adequate for their intended use within the performance assessment model. Therefore, the NRC staff finds that the assessment of the effect of outer container corrosion on the volume increase of corrosion products is acceptable.

With regard to early failure, localized corrosion, or igneous intrusion model cases, the applicant stated in DOE (2009ab, Enclosure 10) that the performance assessment for these model cases does not take credit for the further presence of the waste package; hence, volume increase of corrosion products would not change the estimated consequences. Therefore, the NRC staff finds acceptable that DOE's cases do not take credit for the further presence of the waste package, because this assumption would not underestimate consequences.

On the basis of its review of DOE (2009ab, Enclosures 10–11), the NRC staff finds the applicant assessed a range of corrosion modes for the inner and failed outer containers including crevice corrosion, stress corrosion cracking, and galvanic corrosion. The applicant estimated that the gap between the inner and outer containers would be filled with corrosion products after thousands of years (between 1,400 and 37,000 years) after breaching of the outer container (SNL, 2008ab). The time for stress buildup sufficient to cause stress corrosion cracking on the waste package outer container would exceed 10,000 years after the initial waste package breach, according to DOE (2009ab, Enclosure 10).

The NRC staff notes that uncertainties remain with respect to the magnitude of stainless steel corrosion rates, their environmental dependence, and longer term values (He, et al., 2007ab). Higher stainless steel corrosion rates could fill the gap with corrosion products and cause stress buildup earlier than estimated in the applicant's analyses, increasing the waste package cracked area. However, the applicant described that the extent of the area compromised by cracks is overestimated by the consideration of a crack distribution which fills a two-dimensional space (DOE, 2009ab, Enclosure 10; DOE, 2009cj, Enclosure 5) and consideration of a stress level equal to the yield strength of the material (as opposed to allowing the stress to relax when cracks form or grow) (SNL, 2007bb, Section 6.7.3). On the basis of its review of the applicant's information, and supported by the NRC staff's evaluation of stress corrosion crack size and density in SER Section 2.2.1.3.1.3.2.3 (wherein the NRC staff concluded that DOE appropriately abstracted the dimension of waste package area damaged by stress corrosion cracking for the intent of the performance assessment), the NRC staff finds the applicant provided adequate support for its conclusion that additional stresses from the stainless steel corrosion products are unlikely to significantly increase the extent of the waste package area covered by cracks.

In the case of large weld flaws leading to stress corrosion cracking initiation, the NRC staff finds adequate the applicant's conclusion that the results of the performance assessment would not significantly change; the applicant reached this conclusion by considering stress buildup from stainless steel corrosion products leading to a larger waste package area covered by cracks (larger than the weld cracks alone). This is because, as the applicant stated, (i) large welds flaws leading to stress corrosion cracking are rare and (ii) it could take thousands of years for enough corrosion product buildup to fill inner and outer container gaps and even longer to develop sufficient stress buildup.

On the basis of its review of the applicant analyses considering a complete range of waste package failure modes, computation of the time to fill gaps and produce significant stresses on the waste package outer container, and computations of the area compromised by cracks that are likely to overestimate the waste package damage area, the NRC staff finds acceptable the exclusion of the FEP 2.1.09.03.0B, Volume Increase of Corrosion Products Impacts Waste Package, by low consequence.

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

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

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

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

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

NRC Staff's Review

The NRC staff reviewed the technical basis in the FEP document (SNL, 2008ab), additional information in BSC (2005aa) and SNL (2007al), and the analysis to supplement the screening justification in DOE (2009ab, Enclosures 12–15). The NRC staff reviewed the screening rationale, and on the basis of this review, the NRC staff finds that the applicant provided sufficient technical basis to support the exclusion of this FEP from the performance assessment model.

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

On the basis of these short-term experiments showing that localized corrosion did not initiate under specific conditions enabling deliquescence of salts, the NRC staff finds the applicant provided a technical basis to exclude the FEP from the performance assessment model. The NRC staff finds adequate the applicant's conclusion that there is no evidence localized corrosion could initiate and be sustained for extended periods in deliquescent solutions. The NRC staff finds acceptable the applicant's technical basis to exclude the FEP from the performance assessment.

FEP 2.1.11.06.0A, Thermal Sensitization of Waste Packages

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

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

NRC Staff's Review

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

On the basis of the applicant's thermal aging and phase stability analyses and the applicant's description that abstractions for general and localized corrosion and the mechanical properties of the waste package outer container are bounding, the NRC staff finds acceptable the applicant's conclusion that long-term phase stability of Alloy 22 is of low consequence. Therefore, the NRC staff finds acceptable the exclusion of the FEP Thermal Sensitization of Waste Packages by low consequence (with respect to corrosion resistance and mechanical properties of Alloy 22).

FEP 2.2.07.05.0A, Flow in the UZ [Unsaturated Zone] From Episodic Infiltration

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

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

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

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

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

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

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

NRC Staff's Review

The NRC staff reviewed the applicant's modeling results and field observations and determined that they are consistent with DOE's description of the Flow in the Unsaturated Zone from Episodic Infiltration FEP. Therefore the NRC staff finds the information the applicant provided supports the conclusion that the PTn matrix has a strong potential for dampening large pulses with matrix imbibition. The NRC staff also finds the information provided supports DOE's conclusion that bomb-pulse tritium observations in boreholes below the PTn are likely associated with lateral flow from faults and localized fast flow pathways, which are not necessarily episodic but intermittent preferential flow that is most likely associated with prominent structural features (e.g., faults and intensely fractured zones). The NRC staff notes that episodic flow below the PTn may not be entirely precluded based on a relationship of preferential flow to prominent structural features, because (i) 11 of the 22 tritium observations in the Enhanced Characterization of the Repository Block (ECRB) exhibit a modern signature despite being located more than 100 m [330 ft] from a mapped fault or intensely fractured zone and (ii) the travel times through the PTn that DOE concluded were necessary to explain these observations without invoking episodic flow are more than an order of magnitude faster than those obtained from the calibrated parameters. In its responses to NRC staff RAIs, DOE concluded that bomb-pulse tritium levels from elsewhere at Yucca Mountain (surface boreholes and ESF) are generally observed at or near faults (DOE, 2009cb, Enclosure 8) and other tracers (chlorine-36, temperature, carbon-14) and numerical modeling support the interpretation that episodic flow may occur, but is not prevalent in non-faulted areas of the PTn (DOE, 2009cc, Enclosure 1).

The NRC staff notes that there are uncertainties in interpreting the tritium observations and its use to determine the extent of episodic flow because, in general, (i) the extent of lateral flow is difficult to determine; (ii) faults through the PTn may not extend to observation points (i.e., ground surface and ECRB); and (iii) tritium migrates both in the gas and liquid phases. Therefore, the NRC staff considered the consequence of episodic flow on repository performance, upon which the applicant relied to exclude this FEP.

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

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

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

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

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

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

NRC Staff's Review

The NRC staff reviewed the model assumptions and field and laboratory data the applicant used to support its FEP screening justification, as identified in SNL (2008ab) and DOE (2009ai, Enclosure 1). The NRC staff reviewed site investigation information that suggests temporal variations in key geochemical parameters may influence potential sorption in the regional aquifers around Yucca Mountain (Perfect, et al., 1995aa; Turner and Pabalan, 1999aa), and reviewed additional information in SNL (2007ba, Appendix F). In SER Section 2.2.1.3.9, the NRC staff reviewed the DOE selection of sorption modeling data (i.e., ranges of K_d distributions) for performance assessment modeling under saturated zone conditions at Yucca Mountain. Based on the information the NRC staff has reviewed, and based on NRC staff knowledge and experience involving radionuclide sorption (e.g., Bertetti, et al., 2011aa, Turner, et al., 2002aa), the NRC staff concludes the applicant has adequately addressed the potential evolution of geochemical conditions in the saturated zone because (i) available information about Yucca Mountain groundwater chemistry indicates reducing conditions are limited to localized areas that do not appear to be widespread on a regional scale and (ii) the applicant selected ranges of sorption coefficients for modeling radionuclide transport that adequately addressed spatial and temporal uncertainty about natural variations in pH, redox conditions, temperature, and major ion chemistry. On the basis of the foregoing considerations and its review of information the applicant provided, the NRC staff finds the exclusion of this FEP on the basis of low consequence is acceptable.

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

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

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

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

NRC Staff's Review

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

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

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

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

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

NRC Staff's Review

The applicant's technical basis for exclusion is based on the drip shield performance. As stated in SER Section 2.2.1.3.2, DOE evaluated a range of processes that could compromise the barrier capability of drip shields to protect waste packages against seepage, including drip shield ruptures and plate displacements during seismic events. The NRC staff found (SER Section 2.2.1.3.2) that DOE adequately evaluated the barrier capabilities of the drip shield mechanical disruption due to seismic events and appropriately incorporated the risk-significant aspects of this evaluation into the performance assessment calculations. On the basis of the review documented in SER Section 2.2.1.3.2, the NRC staff finds adequate the DOE conclusion that the drip shield would protect the waste package during a potential re-dissolution period. The NRC staff also considered the DOE screening justifications for FEP 2.1.03.10.0B, Advection of Liquids and Solids Through Cracks in the Drip Shield, and FEP 2.1.03.02.0B, Stress Corrosion Cracking of Drip Shields.

The applicant provided a probabilistic analysis in DOE (2009ab, Enclosure 2) to estimate the additional number of waste packages failed and additional radionuclide releases if drip shields failed in the first 10,000 years due to seismic events. DOE concluded that additional consequences would be too small to change the dose estimates. The NRC staff finds this conclusion acceptable, as explained in the NRC staff's reviews of FEP 2.1.03.10.0B, Advection of Liquids and Solids Through Cracks in the Drip Shield, and FEP 2.1.03.02.0B, Stress Corrosion Cracking of Drip Shields. This finding is also consistent with the review and findings documented in SER Section 2.2.1.3.2. Therefore, the NRC staff finds exclusion of the FEP acceptable on the basis of the drip shield function, which will prevent contact of potentially corrosive fluids with the waste packages during the thermal period when the potential for such conditions would exist.

With respect to drip shield failure by seismic events in the first 10,000 years, the NRC staff finds acceptable the applicant's conclusion that additional consequences would not affect dose estimates. Past the thermal pulse period, the DOE abstraction predicts that there is a low probability for the repository environment (i.e., temperature, pH, and chemical composition of in-drift waters) to support localized corrosion of the waste package even if the drip shield fails and allows seepage water to contact the waste package. The DOE abstractions for the chemistry of water in the drifts and localized corrosion are evaluated in SER Sections 2.2.1.3.3.3.2 and 2.2.1.3.1.3.2.2 and found acceptable by the NRC staff. Hence, as related to this FEP and its technical basis for exclusion, there are no potential consequences from seismic events beyond the 10,000-year post-disposal period through the period of geologic stability. Therefore, NRC staff finds acceptable the exclusion of the FEP Re-dissolution of Precipitates Directs More Corrosive Fluids to Waste Packages from the performance assessment.

FEP 2.2.09.01.0B, Microbial Activity in the Unsaturated Zone

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

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

DOE stated that during the repository's early thermal period, a water vapor condensation-related increase in microbial activity beyond the dryout zone would not significantly affect in-drift seepage chemistry for three main reasons (DOE, 2009ci, Enclosure 1). First, the availability of the already scarce nutrients in the rock would not increase during this period, so nutrient limitation would continue to inhibit microbial development. Second, DOE estimated from modeling calculations that even under present-day unsaturated conditions, the relative humidity in the densely welded Topopah Spring tuff at the repository horizon is near the upper end of conditions which support optimal microbial activity in the rock matrix pore spaces. DOE reasoned, therefore, that any increase in microbial activity in a thermal condensation zone in the rock would be due mostly to an increase in water which condensed in fractures and, compared to the volume of water presently in the matrix, any increase in fracture water would be only a small proportion of the total amount of water that is already accessible to microbes in the rock volume. Third, DOE stated that for the near-field setting as a whole, any increased microbial activity in the condensation zone would be offset during the same timeframe by reduced microbial activity elsewhere, due to elevated temperatures and lack of water in the dryout zone closer to the repository.

NRC Staff's Review

The NRC staff reviewed the screening basis that DOE provided for excluding Microbial Activity in the Unsaturated Zone in FEP 2.2.09.01.0B (SNL, 2008ab) and compared the DOE screening argument with the NRC staff's review of the DOE development of unsaturated zone sorption parameters and colloid concentrations in SER Section 2.2.1.3.7. The NRC staff also consulted (i) other descriptions of microbial activity in the unsaturated rocks at Yucca Mountain (e.g., BSC, 2006aa; Meike and Stroes-Gascoyne, 2000aa) and (ii) investigations of microbial activity in other deep geologic settings (e.g., Sherwood Lollar, 2011aa). Based on its review of the DOE information and the NRC staff knowledge and experience, the NRC staff concludes (i) DOE's methods of measuring sorption coefficients for Yucca Mountain groundwaters and DOE's treatment of uncertainty in the selection of parameter ranges for unsaturated zone K_d values and for naturally occurring colloid concentrations adequately bounded, and implicitly included, the effects of microbial activity in modeling radionuclide transport in the unsaturated zone and (ii) DOE acceptably described the main conditions that influence microbial activity in the unsaturated zone at Yucca Mountain under ambient conditions.

The NRC staff also reviewed the additional screening information DOE provided about the potential effects of unsaturated zone microbial activity in a condensation zone during the repository early thermal period (DOE, 2009ci, Enclosure 1). The NRC staff examined site characterization data for the densely welded Topopah Spring tuff (BSC, 2004bi, Section 7.2); Yucca Mountain field tests and interpretations of fracture flow and transport processes, as described in BSC (2004bi, Section 7.6.3.1; 2006aa, Section 6.2.4); and DOE reports of increased microbial activity (biofilms) in fractures during an unsaturated zone tracer transport experiment at Yucca Mountain (BSC, 2006aa, Sections 6.1.2 and 6.2.4).

Based on the review of these materials, the NRC staff notes the applicant's screening statement that the water in fractures represented only a small fraction of the total water content of the rock does not clearly address the potential effect of increased microbial activity in the condensation zone fractures in the context of (i) the geochemical importance of microbially altered water compositions in fracture-dominated flow pathways as potential seepage into drifts (SAR Section 2.3.3.1) or (ii) the potentially large fraction of water transferred by evaporation from the dryout zone to the fractures of the condensation zone during the thermal period, as detailed in SNL (2008aj, Section 6.1.2). Moreover, the growth of biofilms in unsaturated zone fractures beneath ponded water in the Yucca Mountain tracer test (BSC, 2006aa, Section 6.1) attests that an increased availability of water in fractures can promote microbial activity in the unsaturated zone, even if nutrients are limited. However, the NRC staff also notes that DOE's conceptual thermohydrologic model (SNL, 2008aj, Section 6.1.3) describes water condensed in the fractures above the repository during the thermal period does not remain in place but continually drains downwards and away from the dryout zone along a network of connected fractures between the emplacement drifts.

The NRC staff reviewed the adequacy of the applicant's thermohydrologic models in SER Section 2.2.1.3.6 and found that the models were an acceptable basis for the performance assessment abstraction. On the basis of its review of DOE's modeling results, which provide that condensation water in fractures above the repository would drain around and away from the engineered drifts during the thermal period, the NRC staff finds acceptable DOE's decision to exclude the effects of microbial activity in the unsaturated zone during the thermal period on the basis of low consequence.

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

The applicant excluded Thermo-Chemical Alteration of the Topopah Spring Basal Vitrophyre on the basis of low consequence (SNL, 2008ab). According to the applicant's information, the Topopah Spring basal vitrophyre is a glassy, densely welded tuff that forms the lowermost part of the Topopah Spring tuff hydrogeologic unit. The applicant used the FEP to examine the possibility that temperatures elevated by repository heating might cause the volcanic glass in the vitrophyre to alter to zeolites and clay minerals, potentially changing permeability and flow paths in the basal vitrophyre and increasing the sorptive properties of the unit (SNL, 2008ab).

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

NRC Staff's Review

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

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

The criticality FEPs encompass FEPs 2.1.14.15.0A through 2.1.14.26.0A and 2.2.14.09.0A through 2.2.14.12.0A (SAR Table 2.2-5), a total of 16 FEPs, and are classified as events. The applicant excluded all of the criticality FEPs from the performance assessment on the basis of low probability (SAR Table 2.2-5; SNL, 2008ab). As described in SAR Section 2.2.1.1.2, the potential for criticality events is determined by a number of precursor conditions that must occur for the inventory to achieve a potentially critical configuration. An initiating event must occur which causes a breach of the waste package before any other sequence of events on that waste package could lead to criticality. Additional precursor conditions include (i) presence of a moderator (i.e., water), (ii) separation of fissionable material from the neutron absorber material or an absorber material selection error during the canister fabrication process, and (iii) the accumulation (external) or presence of a critical mass of fissionable material in a critical geometric configuration. As described by the applicant in SNL (2008ab), the probability of developing a configuration with criticality potential is insignificant unless the initiating event and all three of the precursor conditions are realized.

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

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

The applicant's technical basis for screening out the individual criticality FEPs was provided in SNL (2008ab). The applicant supplemented its technical basis for exclusion of individual FEPs and, in some cases, the technical basis for exclusion of the criticality event class in DOE (2009aj, Enclosures 1–18; 2009al, Enclosures 1–11; 2009gp, Enclosures 1–5; 2009gq, Enclosures 1–2; 2009bv, Enclosures 1–10; 2009bw, Enclosures 1–11; 2009bx, Enclosures 1–2; 2009co, Enclosures 1–4; 2010ah, Enclosures 1–2; 2010ad, Enclosures 1–2; 2009cb, Enclosures 3–5). In this section, the NRC staff reviews the applicant's technical basis for excluding the individual criticality FEPs and the technical basis for excluding the criticality event class.

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

As discussed in SAR Section 2.2.1.4.1.3.2.3, rockfall related to nonseismic processes does not initiate a breach of the waste package and the probability of criticality for this initiating event scenario is negligible. Because rockfall from nonseismic processes does not generate rock block sizes sufficient to tear or rupture the drip shield plates (SER Section 2.2.1.3.1), the NRC staff did not evaluate rockfall-related configurations (FEPs 2.1.14.21.0A and 2.1.14.22.0A) in detail, as these configurations would be bounded by the more risk significant configurations discussed next.

For the near- and far-field criticality FEPs (FEP 2.1.14.17.0A, FEP 2.1.14.20.0A, FEP 2.1.14.23.0A, FEP 2.1.14.26.0A, and FEPs 2.2.14.09.0A through FEP 2.2.14.12.0A), the applicant adopted a number of assumptions (SAR Section 2.2.1.4.1.1.2.2) to overestimate the probability of criticality. The NRC staff considered that two of the most significant DOE assumptions unique to near- and far-field criticality were (i) fissile material would accumulate in the optimum geometry for criticality and (ii) neutron absorbers and fission products would not be located nearby. Despite these assumptions, SAR Table 2.2-8 indicates that near- and far-field criticality is a negligible contributor to the overall probability of criticality [the cutoff for including probabilities in this table is two orders of magnitude lower than the probability limit, or 10^{-8} per year, in 10 CFR 63.342(a)]. Therefore, the NRC staff did not perform a detailed review of the near- and far-field criticality FEPs.

The NRC staff reviewed the information provided in the application, including the cited references, and concluded that the remaining three FEPs encompassing in-package intact configurations (FEP 2.1.14.15.0A) and in-package degraded configurations (FEPs 2.1.14.16.0A and 2.1.14.19.0A) are the most risk significant. The in-package intact configurations are the as-designed configurations and initially apply to all waste packages. Any problems with this configuration could potentially impact all waste packages. The in-package degraded conditions, including seismic in-package degraded conditions, were assigned the highest probabilities of a criticality event by the applicant (SAR Table 2.2-8). For these reasons, these three criticality FEPs (i.e., the two waste configurations) were evaluated in greater detail in this SER section.

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

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

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

Moderator Intrusion

The presence of a moderator is necessary for the spent fuel to go critical. Because water is a neutron moderator, its presence results in increased reactivity. Unbreached waste packages do not allow ingress of water (neutron moderator) and hence do not pose a criticality concern. The applicant conservatively assumed that enough water is available to act as a neutron moderator in the criticality calculations whenever any breach of the package is calculated—even for cracks on the waste package which are too small to permit liquid infiltration (SNL, 2008ab). No credit is taken for only partial filling of the waste package or for the drip shield's diversion of liquid from the package (DOE, 2009bx, Enclosure 1); hence the waste packages, once breached, are assumed to be flooded. . In the nominal modeling case, DOE estimates that the waste packages would not be breached in 10,000 years (e.g., SAR Figure 2.1-10). However, other modeling cases (early failure, seismic ground motion, seismic fault displacement, and igneous intrusion) account for failure of the waste package within 10,000 years (e.g., SAR Figure 2.4-18). Because of the low probability for igneous intrusion and fault displacement, DOE concluded that criticality events could only occur within 10,000 years for the early failure and seismic scenarios. In its criticality analysis, the applicant considered various reactivity control mechanisms within a waste package (e.g., neutron absorbers) to ensure that all probable configurations remain subcritical.

NRC Staff's Review of Moderator Intrusion

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

The NRC staff finds that the applicant adequately identified and quantified conditions which could lead to in-package criticality for the following reasons. The NRC staff reviewed the conditions the applicant identified as necessary for in-package criticality, as well as the methodology used to identify these conditions. The total probability of criticality is dependent on the probability to attain those necessary conditions (e.g., waste package failure, moderator intrusion, configuration changes, package/absorber degradation, and fuel characteristics). The applicant stated that a criticality event was possible only after a waste package breach and under the following conditions: (i) accidentally loading a fuel assembly with higher reactivity than permissible into the waste package (a mistake referred to as a misload) or (ii) manufacturing errors resulting in missing neutron absorber material. The NRC staff finds the misloading criteria and manufacturing performance criteria used in the analysis acceptable, because the applicant applied human reliability data developed by the industry and the international community. The data appropriately considered dependencies and human factors in manufacturing and loading procedures. The NRC staff also finds the data and human reliability factors which the applicant used for fuel assembly misload or manufacturing error resulting in missing neutron absorbing material that the applicant used to exclude the criticality FEPs are consistent with those used by the nuclear industry.

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

Misloaded Fuel

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

The waste package configuration used to compute the loading curves was selected to bound degraded configurations that were not explicitly evaluated in the screening argument [e.g., the conversion of UO_2 into schoepite, as described in SNL (2008aa, Sections 6.1.3 and 6.2.5)]. The applicant implemented these bounding analyses to provide confidence that waste package configurations not explicitly analyzed are less reactive than those which were analyzed and used to generate the loading curve.

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

The applicant assumed that misloads will not occur for DOE spent nuclear fuel, because the physical differences in fuel types allow operators to easily distinguish spent fuel types and in some cases physically prevent misloads [SNL, 2008ab; FEP 2.1.14.15.0A, In-package Criticality (Intact Configuration)]. A commercial spent nuclear fuel misload would occur if an underburned assembly is loaded; however, a criticality event would only be possible if the other assemblies were not burned enough to compensate for the underburned assembly.

NRC Staff's Review of Misloaded Fuel

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

The applicant also calculated an additional probability related to misloads: the probability that a criticality event would occur given one assembly has been misloaded into a canister. As described in DOE (2009aj, Enclosure 13), the probability of criticality due to an assembly misload in a breached pressurized water reactor (PWR) waste package is the product of the probability of a misload (1.18×10^{-5}) and the conditional probability of criticality given a misload (0.014). To calculate the conditional probability of criticality given a misload, the applicant assumed that misloaded assemblies would be positioned in the center (i.e., along the axis) of the waste package. The NRC staff finds this assumption to be adequate because this position is the most reactive due to neutron interaction with the surrounding assemblies, maximizing the likelihood of criticality. In other words, this assumption is conservative with regard to estimates of the probability of criticality. However, the applicant's analysis did not account for the probability of criticality if there are two or more misloads. Multiple misloads are likely to result from a common mode failure and cannot be accurately modeled as random misloads. In DOE (2009aj, Enclosure 13), the applicant supplemented its previous analysis by performing a sensitivity study that assumed a criticality event resulted whenever a misload occurred (which could be a single assembly misloaded or the first of several assemblies being misloaded). Hence, this analysis changed the value of the conditional probability of criticality from a misload from 0.014 to 1. The applicant identified that this change in the conditional probability only causes the overall probability of criticality to increase to a value of 5.76×10^{-5} in 10,000 years, which is less than the exclusion criterion. Because in this sensitivity study every misload is assumed to cause a criticality event, the probability of criticality given multiple misloads is bounded. On the basis of this sensitivity study, the NRC staff finds adequate the applicant's conclusion that multiple misload assemblies alone would not cause the probability of criticality to exceed the regulatory criterion or 10^{-8} per year.

Neutron Absorbers

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

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

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

NRC Staff's Review of Neutron Absorbers

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

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

Because liquid water flowing into a waste package is unlikely, the NRC staff concludes that water ingress into the waste packages in 10,000 years will most likely be in the form of water vapor. Accordingly, the NRC staff reviewed literature data on borated stainless steel corrosion under humid air conditions. Beavers and Durr (1991aa) studied corrosion of 304 stainless steel under vapor and aqueous conditions at 90 °C [194 °F] and concluded that corrosion rates were under their detection limit of 200 nm/yr [7.9×10^{-3} mil/yr]. The corrosion rate value the applicant used to support the use of a 6-mm [0.24-in]-thick plate of borated stainless steel in the criticality computations, 27 nm/yr [0.001 mil/yr], is below the upper bound of 200 nm/yr [7.9×10^{-3} mil/yr] estimated by Beavers and Durr (1991aa). The NRC staff conducted independent tests of the corrosion behavior of Types 304B4 and 304B5 borated stainless steel in simulated groundwater and humid air at 60, 75, and 90 °C [140, 167, and 194 °F] (He, et al., 2012). The NRC staff found that a small amount of specimens exposed to humid air at 75 and 90 °C [167 and 194 °F] suffered pitting corrosion, but pitting corrosion was not observed at 60 °C [140 °F] or from the simulated groundwater exposure at 75 and 90 °C [167 and 194 °F]. The general corrosion rates of Type 304B4 were less than 80 nm/yr [0.0032 mil/yr], and those of Type 304B5 were less than 600 nm/yr [0.024 mil/yr]. The NRC staff concludes, however, that even if the corrosion rate under humid air exceeded 27 nm/yr [0.001 mil/yr] and pitting corrosion was observed, stainless steel is not expected to continuously corrode from the time of closure to 10,000 years after closure {as assumed in the applicant's derivation of the 6-mm [0.24-in]-thick value}, because the waste packages could fail at different times in the 10,000-year period and the spent fuel is a

heat source that could mobilize water away. Also, boron present in corrosion products could still be an effective neutron absorber. Accordingly, the NRC staff concludes that use of a 6-mm [0.24-in]-thick plate of borated stainless steel in the criticality analysis is adequate. Therefore, the NRC staff finds acceptable the applicant's incorporation of a degraded neutron absorber in its criticality analyses. Furthermore, the NRC staff finds the neutron absorbers in the transportation, aging, and disposal canister, with the assumption that neutron absorber plates are at least 6 mm [0.24 in] thick for 10,000 years, provide adequate reactivity control.
The NRC staff concludes the applicant appropriately calculated probability that is below the regulatory exclusion probability of 1.0×10^{-4} over 10,000 years for the following reasons: (i) the applicant adequately used a surrogate analysis to estimate the probability of errors in neutron absorber manufacturing and installation (i.e., probability with a mean value of 1.25×10^{-7} per canister), because the applicant considered representative reliability data and prototype manufacturing procedures to estimate the probability of misloading the neutron absorber plates and (ii) the applicant used conservative probability values for conditions that may lead to in-package criticality, including waste package damage, fully flooded waste packages after breach, and significant fuel degradation and/or fuel reconfiguration, in combination with the previously mentioned probability of neutron absorber misload, to obtain a mean in-package criticality probability of 4.4×10^{-5} over 10,000 years (DOE, 2009aj, Enclosure 5).

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

The NRC staff reviewed the applicant's proposed use of a gadolinium absorber shot, which prevents criticality for some of the most reactive DOE spent nuclear fuel types. The NRC staff finds that filling the DOE spent nuclear fuel canisters with enough absorber shot to keep the most reactive intact or degraded configuration subcritical, with an adequate margin, is an acceptable use of neutron absorbers. This is because the applicant modeled both degraded and undegraded absorber shots, and the model results provide sufficient technical basis to support the applicant's conclusion that the absorber shot will perform its intended function.

Burnup Credit

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

The applicant used reactor records to determine the burnup of the fuel (SAR Section 2.2.1.4.1.1.4.1) and a computer model to generate the isotopic composition for a given burnup and enrichment. The applicant accounted for the uncertainties in the reactor records by using a burnup that is 5 percent less than reported. This adjustment bounded the highest reactor record uncertainty identified in AREVA (2004, Table 10B), which was 4.2 percent. The reactor record uncertainty was determined from the difference between the calculated and measured values of burnup. Measured burnup was determined with calibrated in-core instrumentation. Calculated burnup was determined using analytic methods of the reactor core power distribution.
The applicant used a decay time for the isotopic composition of 5 years to bound the increase in reactivity caused by Am-241 and Pu-240 decay. Uncertainties in the isotopic composition were accounted for by using modeling parameters that would conservatively overestimate the reactivity of the fuel and overestimate the probability of criticality (SAR Section 2.2.1.4.1.1.2.4.1). In DOE (2009bx, Enclosure 2), DOE also stated that it used a maximum burnup of 50 GWd/MTU with respect to the burnup credit loading curves in criticality analyses for commercial pressurized water reactor fuels.

For boiling water reactor (BWR) spent nuclear fuel, the applicant performed two different criticality analyses. One analysis applies to BWR spent nuclear fuel with initial enrichments up to 4.5 wt% U-235; this analysis does not incorporate burnup credit in the analysis. For the second analysis, some BWR burnup credit is necessary to accommodate fuel assemblies with higher initial enrichments (i.e., > 4.5wt% U-235) than the projected waste stream inventory in SNL (2008aa, Figure 6-33). As discussed in DOE (2010ah, Enclosure 2), only a small amount of burnup credit is required for disposal; in other words, only a fraction of the burnup needs to be credited. While this analysis was performed with a “simple” BWR assembly design as compared to the “modern” and more advanced fuel designs, the more advanced designs ultimately result in a net decrease in the total fissile mass available in the fuel assembly at discharge. The applicant stated that assembly designs which have not been explicitly analyzed (like the advanced BWR designs) would be evaluated by DOE prior to waste receipt to ensure acceptability from a postclosure criticality perspective in accordance with SAR Section 5.10.2 and SAR Table 5.10-3. As stated by the applicant in DOE (2010ah, Enclosure 2), SAR Table 5.10-3 describes a number of administrative controls to be included in the license specifications required by 10 CFR 63.43. DOE stated that these administrative controls will be compiled in the Technical Requirements Manual and maintained by DOE in accordance with the requirements of the license specifications.

NRC Staff’s Review of Burnup Credit

The NRC staff reviewed the applicant’s use and justification of BWR fuel burnup credit. For BWR fuel assemblies with enrichment below 4.5 wt% U-235, the applicant’s analytic methodology that does not incorporate burnup credit is acceptable. However, a very small percentage of BWR fuel assemblies may contain enrichment greater than 4.5wt% U-235. For these assemblies, the applicant requested burnup credit. While the applicant did not have the requisite data to support the isotopic composition and associated reactivity of these fuel assemblies (DOE, 2004ab), the NRC staff does not consider the applicant’s inability to accurately predict the isotopic composition and reactivity of modern BWR spent nuclear fuel to be risk significant, because the applicant uses conservative input parameters to the irradiation and depletion calculations when generating the BWR isotopic database of spent fuel isotopic compositions with a small amount of burnup being credited. As a result, the applicant only takes credit for a portion of the burnup credit from the principal isotopes that are expected to be available. Additionally, NRC staff does not consider granting BWR burnup credit to be risk significant, because the administrative controls described in SAR Section 5.10.2 and

SAR Table 5.10-3 require that similar analyses (similar to the analysis performed for the simple BWR design) be completed prior to receiving individual waste forms or canisters/waste package design configurations which are not explicitly analyzed in the SAR. Therefore, the NRC staff considers it acceptable for the applicant to take burnup credit for BWR spent nuclear fuel assemblies with greater than 4.5 wt% enrichment. The NRC staff reviewed the applicant's use and justification of pressurized water reactor burnup credit. The applicant used conservative input parameters (such as fuel types and operating histories) in the irradiation and depletion calculations when generating the isotopic composition of the pressurized water reactor fuel. The NRC staff determines that taking full credit for the neutron absorptive properties of Mo-95, Tc-99, Ru-101, Rh-103, and Ag-109 is technically unjustified due to inadequate radiochemical assay data, which were primarily based on the 11 TMI-1 radiochemical assay samples. However, in DOE (2010ah, Enclosure 1), the applicant demonstrated that the isotopic bias and uncertainty incorporated into the critical limit should make up for the errors and uncertainties in the predictions of these five isotopes. Furthermore, the NRC staff finds the following factors provide additional technical basis to support the conclusion that the overall uncertainties in the isotopic predictions of the 29 principal isotopes for pressurized water reactor burnup credit are not sufficient to cause a significant increase in the probability of criticality: (i) the presence of some neutron-absorbing fission products that the applicant did not credit in the analysis, (ii) the radiochemical assay data from a variety of reactors, and (iii) conservative input parameters .

Criticality Code Validation

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

An important aspect in assessing criticality code validation is the applicability of the selected benchmark experiments, which must be similar in form and composition to the systems to be evaluated. The applicant described the benchmarks used for the criticality code validation, and the determination of the lower bound tolerance limits for commercial spent nuclear fuel and DOE spent nuclear fuel, in DOE (2004aa) and Radulescu, et al. (2007aa). The applicant included in the criticality model document (DOE, 2004aa) analyses for the various types of commercial and DOE spent nuclear fuel. The applicant updated its criticality validation methodology and benchmarks for commercial spent nuclear fuel in Radulescu, et al. (2007aa), as explained in DOE (2009al, Enclosure 7). The applicant did not update the benchmark selection and validation for DOE spent nuclear fuel with the new methodology used in Radulescu, et al. (2007aa). In the note to SAR Table 2.2-11, DOE stated that DOE spent nuclear fuel interim critical limits have not been rigorously established for all fuel groups, but will be confirmed by DOE prior to waste acceptance as identified in SAR Table 5.10-3. In DOE (2010ad, Enclosure 1) the applicant committed to revise SAR Section 2.2.1.4.1.1.2.4.2 with text that includes the statement: "Prior to waste receipt, DOE will demonstrate that the bias used in establishing loading limits for DOE spent nuclear fuel canisters conservatively envelopes any uncertainty associated with the limited availability of applicable benchmarks."

An important consideration in the development of critical limits is the determination and implementation of an administrative margin. As described in SAR Section 2.2.1.4.1.1.2.4.2, the administrative margin is an arbitrary margin ensuring subcriticality and turning the criticality limit function into an upper subcritical limit function. The applicant described that this term is not applicable for use in postclosure analyses, because there is no risk associated with a subcritical event. The applicant described that in contrast to "traditional" nuclear criticality safety analyses and associated governing regulations, in which the purpose is to ensure prevention of criticality and corresponding protection of personnel and facilities, the purpose of the postclosure criticality evaluation is to determine the probability of a criticality event in the postclosure time period. The probability of criticality is then compared to the probability of the regulatory screening criterion, or 10^{-8} per year, to reach a decision relative to the inclusion or exclusion of a criticality event in the evaluation of the total system performance for the facility. Therefore, the applicant used a zero administrative margin and provided additional justification for its use of a zero administrative margin in DOE (2009bv, Enclosure 10).

NRC Staff's Review of Criticality Code Validation

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

The NRC staff finds that the upper subcritical limit was calculated properly and the calculation was performed in accord with NRC-endorsed methods (NRC, 2005ac). The validation and validation methodology in Radulescu, et al. (2007aa) is acceptable to the NRC staff because the predictor variables used are sufficient to fully characterize the bias across the range of applicability. Therefore, the NRC staff finds that the applicant's validation of the criticality model for commercial spent nuclear fuel loading limits is appropriate.

The NRC staff notes that the applicant's validation methodology and selection of benchmarks for DOE spent nuclear fuel was not evaluated using the applicant's updated methodology in Radulescu, et al. (2007aa). The NRC staff further notes that, as described previously, the updated methodology has been applied to commercial spent nuclear fuels (Radulescu, et al., 2007aa) and the updated results for commercial spent nuclear fuels show results from DOE's previous methodology is not affected by the updated methodology. In response to an NRC staff RAI, the applicant stated in DOE (2010ad, Enclosure 1) that prior to waste receipt, DOE will apply its updated methodology to DOE spent nuclear fuel and will demonstrate the bias used in establishing loading limits for DOE spent nuclear fuel canisters conservatively envelopes any uncertainty associated with the limited availability of applicable benchmarks. The applicant also stated in SAR Table 2.2-11 that, for all DOE spent nuclear fuel groups for which the applicant has established bounding interim critical limits, the limits will be confirmed prior to waste acceptance. The NRC staff found that DOE's RAI response addressed the differences in the updated methodology used for commercial spent nuclear fuels and the previous methodology used for DOE spent nuclear fuels. The NRC staff notes that evaluation of the critical limits would be one element of an NRC review of an application to receive and possess waste.

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

Summary of NRC Staff's Review of Criticality FEPs

The NRC staff reviewed the models, calculations, and results DOE used for excluding criticality from the performance assessment using risk-informed, performance-based review methods described in the YMRP. Important areas of the review included determining the acceptability of the inputs to and validation of the isotopic irradiation and depletion model, the criticality models, the probability models, and the methodologies used in the models. Based on the results of its review, the NRC staff finds that the DOE's technical basis for excluding the criticality FEPs (FEPs 2.1.14.15.0A through 2.1.14.26.0A and 2.2.14.09.0A through 2.2.14.12.0A) from the performance assessment on the basis of low probability is acceptable. The NRC staff finds that the applicant submitted an adequate technical basis for excluding the criticality event class on the basis of low probability because the overall probability of criticality (4.4×10^{-5}; SAR Table 2.2-8) is less than the limit in 10 CFR 63.342(a).

Summary of NRC Staff's Review of Screening of the List of Features, Events, and Processes

As described above the NRC staff makes the following findings. First, the NRC staff finds that the applicant has identified all FEPs related to either the geologic setting or to the degradation, deterioration, or alteration of engineered barriers (including those processes that would affect the performance of natural barriers) which have been excluded (in accordance with YMRP Section 2.2.1.2.1.3, Acceptance Criterion 2, Subcriterion 1). Second, the NRC staff finds that the applicant has provided a justification for exclusion of each FEP and finds acceptable the applicant's criteria for exclusion (i.e., justification) on the basis of low probability, low consequence, or exclusion by regulation, because these criteria are consistent with regulatory requirements for scenario analysis discussed in SER Section 2.2.1.2.1.2 (following YMRP Section 2.2.1.2.1.3, Acceptance Criterion 2, Subcriterion 2). Third, the NRC staff finds DOE has provided an adequate technical basis for each FEP excluded from the performance assessment to support the conclusion that either (i) the FEP is specifically excluded by regulation, (ii) the probability of the FEP falls below the regulatory criterion, or (iii) omission of the FEP does not significantly change the magnitude and time of the resulting radiological exposures to the RMEI or radionuclide releases to the accessible environment (following YMRP Section 2.2.1.2.1.3, Acceptance Criterion 2, Subcriterion 3). Fourth, the NRC staff finds that the technical basis for FEP 1.2.10.01.0A, Hydrologic Response to Seismic Activity, and DOE's supplemental analyses, comprise an analysis of seismic effects on the elevation of the water table beyond the

10,000-year postdisposal period through the period of geologic stability and satisfy the requirement in 10 CFR 63.342(c)(1)(i) for the analysis of seismic effects on the elevation of the water table. The NRC staff finds that the applicant properly excluded FEPs from the performance assessments beyond the 10,000-year postdisposal period through the period of geologic stability and properly identified particular FEPs to be included in these performance assessments as required in 10 CFR 63.342(c). The NRC staff finds that DOE considered only FEPs consistent with the limits on performance assessment specified at 10 CFR 63.342.

2.2.1.2.1.3.3 Technical Review for Formation of Scenario Classes

Summary of the DOE License Application on Formation of Scenario Classes

The applicant described the approach to the definition of event class and scenario class formation in SAR Section 2.2.1.3. The applicant cited 10 CFR 63.102(j) as a basis to define an event class as consisting of all possible initiating events that are caused by a common natural process. The applicant extended the definition to allow event classes to represent the aggregation of initiating events with a common characteristic, either natural or manmade. The applicant stated that the objective of scenario class development is to define a limited set of scenario classes that could be analyzed quantitatively while still covering the range of possible future states of the repository system (SAR Section 2.2.1.3, p. 2.2-22).

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

The applicant discussed scenario class formation for the human protection standard in SAR Section 2.2.1.3.2 and referred to 10 CFR 63.322 to define assumptions to support the development of the scenario.

NRC Staff's Evaluation of Formation of Scenario Classes

The NRC staff evaluated the formation of scenario classes for performance assessments to demonstrate compliance with the individual protection standard and separate standards for protection of groundwater by evaluating whether they cover the full range of potential future states of the repository system. The NRC staff finds the classifications adequate and comprehensive as they encompass all possibilities (aside from human intrusion) leading to radionuclide release consistent with the set of included FEPs. These four scenarios are not mutually exclusive (e.g., an initially failed waste package could also be affected by a seismic event). However, to assess the effect of combined scenarios, the applicant considered the total of eight possible combinations from the three event and the nominal scenarios (see the Venn diagram in SAR Figure 2.2-3). The NRC staff concludes that these eight scenario class combinations cover all of the possibilities of damaging events or processes which could affect a waste package during the timeframe of the analysis. Therefore, the applicant left no situation that could lead to radionuclide release without consideration and evaluation. The applicant considered the effect of these eight combinations in determining aggregated dose estimates.

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

With respect to the scenario class formation for human intrusion, DOE referred to 10 CFR 63.322 to define assumptions for the analysis to demonstrate compliance with individual protection standards for human intrusion (10 CFR 63.321). The NRC staff finds acceptable the reference to 10 CFR 63.322 to set assumptions for the human intrusion scenario analysis, on the basis of the NRC staff's evaluation of the human intrusion scenario analysis performed in SER Section 2.2.1.4.2.3, where the NRC staff found that the performance assessment used to estimate the annual dose curve is consistent with the requirements for the postulated human intrusion event.

2.2.1.2.1.3.4 Screening of Scenario Classes

Summary of DOE's Approach for Screening of Scenario Classes

In SAR Section 2.2.1.4, DOE described Step 4 (screening for scenario classes) of the scenario analysis (SAR Section 2.2.1). DOE provided justifications for excluding scenario classes from the performance assessment analyses on the basis of probability or consistency with the regulations (10 CFR 63.311, 10 CFR 63.321, and 10 CFR 63.331).

The applicant included the following four scenario classes in the performance assessment to demonstrate compliance with the individual protection standard (10 CFR 63.311): nominal, early failure, seismic, and igneous scenario classes. The applicant asserted that all of these scenario classes have a probability greater than 10^{-4} in 10,000 years.

For the performance assessment to demonstrate compliance with separate groundwater protection standards (10 CFR 63.331), the applicant excluded the igneous scenario class on the basis that its probability is less than 0.1 in 10,000 years.
The applicant excluded the human intrusion scenario class from the performance assessments to demonstrate compliance with individual protection and separate groundwater protection standards on the basis that it is explicitly ruled out by the regulation.

NRC Staff's Evaluation of Screening of Scenario Classes

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

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

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

DOE included the nominal, early failure, seismic, and igneous scenarios in the performance assessment to demonstrate compliance with individual protection standards. The human intrusion scenario was analyzed separately in conformance with 10 CFR 63.321. The NRC staff finds acceptable the inclusion of the four scenario classes. As stated in the previous section on formation of scenario classes, the four scenario classes incorporate all events (human intrusion aside) that could lead to significant radionuclide release. The NRC staff finds DOE's exclusion of human intrusion from the performance assessment acceptable on the basis of consistency with regulations at 10 CFR 63.321 and 63.322 and the definition of Undisturbed Yucca Mountain Disposal System in 10 CFR 63.302.

DOE included the nominal, early failure, and seismic scenarios in the performance assessment to demonstrate compliance with separate groundwater protection standards and excluded the igneous and human intrusion scenarios. The NRC staff finds acceptable the exclusion of the igneous scenario on the basis that igneous events are unlikely. In 10 CFR 63.342(b), unlikely events are defined as those estimated to have a chance less than 1 in 100,000 per year of occurring and at least 1 chance in 100 million per year. On the basis of the NRC staff's evaluation in SER Section 2.2.1.2.2, the probability of igneous events is below 1 in 100,000 per year of occurring; therefore, the NRC staff finds the applicant provided sufficient technical basis to exclude the igneous scenario. The igneous scenario class is sufficiently broad to encompass a range of events. Therefore, the NRC staff determines that this scenario class was not prematurely excluded by a narrow definition. Finally, the NRC staff finds exclusion of human intrusion from the performance assessment consistent with 10 CFR 63.331 and the definition of Undisturbed Yucca Mountain Disposal System in 10 CFR 63.302.

The NRC staff finds the DOE's screening of scenario classes acceptable on the basis of the evaluations in the preceding paragraphs. DOE's screening of scenario class was comprehensive, clearly documented, and technically acceptable. Scenario classes DOE excluded from the performance assessment on the basis that they are specifically ruled out by regulation were identified, and sufficient justifications were provided to exclude those scenario classes. Those scenario classes DOE excluded from the performance assessment on the basis that their probabilities fall below the regulatory criterion were identified, and adequate bases were provided to exclude those scenario classes.

For compliance with individual protection standards (10 CFR 63.311), DOE included nominal, early failure, seismic, and igneous scenario classes from the performance assessment, and the NRC staff finds this to be acceptable on the basis that the probabilities of these scenario classes exceed 10^{-8} per year (probabilities evaluated in SER Section 2.2.1.2.2) and that these scenario classes incorporate all included events which could lead to significant radionuclide release. For compliance with the separate standards for groundwater protection (10 CFR 63.331), DOE excluded the igneous scenario class from the performance assessment model, and the NRC staff finds this exclusion acceptable because the applicant provided sufficient justification to show that igneous events are unlikely. For compliance with 10 CFR 63.311 and 63.331, DOE excluded the human intrusion scenario from the performance assessments, and the NRC staff finds this exclusion acceptable because the applicant provided sufficient justification. The NRC staff finds the DOE's screening of scenario classes to be acceptable for performance assessments used in determining compliance with 10 CFR 63.311 and 10 CFR 63.331. Evaluation of event classes in regard to computations to address compliance with human intrusion standards (10 CFR 63.321) is included in SER Section 2.2.1.4.2.

2.2.1.2.1.4 Evaluation Findings

The NRC staff has reviewed the SAR and other information submitted in support of the license application, which includes information required by 10 CFR 63.21(c)(1) and (9), and finds, with reasonable expectation, that relevant requirements of 10 CFR 63.114 and 10 CFR 63.342 are satisfied. In particular, the NRC staff finds

- The SAR provides an adequate initial list of FEPs related to the geologic setting or the degradation, deterioration, or alteration of engineered barriers (including those processes that would affect the performance of natural barriers) that have the potential to influence repository performance

- The list of initial FEPs has been appropriately screened

- Scenario classes formed from the screened list of FEPs are adequate

- Scenario classes have been appropriately screened

The NRC staff reviewed the screening argument for FEP 1.2.10.01.0A, Hydrologic Response to Seismic Activity, and supplemental analyses and finds, with reasonable expectation, that DOE satisfied 10 CFR 63.342(c)(1)(i) requirements for the analysis of seismic effects on the elevation of the water table beyond the 10,000-year postdisposal period throughout the period of geologic stability.

The NRC staff finds, with reasonable expectation, that the applicant properly excluded FEPs from the performance assessments beyond the 10,000-year postdisposal period throughout the period of geologic stability and properly identified particular FEPs to be included in these performance assessment as required in 10 CFR 63.342(c).

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CHAPTER 3

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

2.2.1.2.2.1 Introduction

Safety Evaluation Report (SER) Section 2.2.1.2.2 evaluates the U.S. Department of Energy's (DOE, or the applicant) information on event probability used in its performance assessment evaluations and in support of DOE's Total System Performance Assessment (TSPA) model calculations. The U.S. Nuclear Regulatory Commission (NRC) staff evaluated information provided in the Safety Analysis Report (SAR) (DOE, 2008ab, 2009av), as supplemented by DOE's responses to the NRC staff's requests for additional information (RAIs) (DOE, 2009aa–ad,aq,bd), and in the references cited therein.

A performance assessment is a systematic analysis that answers the following triplet risk questions: What can happen? How likely is it to happen? What are the consequences? Scenario analysis answers the first question: What can happen? The NRC staff's evaluation of DOE's scenario analysis is documented in SER Section 2.2.1.2.1. One result from the scenario analysis is the identification of events to be included in the performance assessment calculation used to demonstrate compliance with the postclosure performance objectives. SER Section 2.2.1.2.2 addresses the second question: How likely is it that these events will happen?

A performance assessment evaluation is required, per 10 CFR 63.113, to demonstrate compliance with the postclosure individual protection standard, the human intrusion standard, and the groundwater protection standard. Sections 63.114, 63.303, 63.305, 63.312, and 63.342 identify the regulatory standards for the performance assessment used to demonstrate compliance with the individual protection standard (10 CFR 63.311). For demonstrating compliance with the human intrusion standard (10 CFR 63.321), the requirements for the performance assessment are provided in 10 CFR 63.114, 63.303, 63.305, 63.312, and 63.342, and the requirements for the human intrusion scenario are provided in 10 CFR 63.322. For the purpose of the human intrusion analysis, DOE must make the assumptions identified in 10 CFR 63.322(a)−(g). The requirements for the performance assessment used for demonstrating compliance with the groundwater protection standard (10 CFR 63.331) are identified in 10 CFR 63.114, 63.303, 63.332, and 63.342.

A performance assessment evaluation that is used to demonstrate compliance with the individual protection standard for the proposed Yucca Mountain repository must consider events that have at least 1 chance in 100 million per year of occurring. To address this requirement, DOE identified and described those events that exceeded this probability threshold (10^{-8} per year). Performance assessments are also used to demonstrate compliance with the human intrusion and groundwater protection standards. These performance assessments have different considerations for event probabilities than those required for the individual protection standard and are evaluated in SER Sections 2.2.1.4.2 and 2.2.1.4.3, respectively. DOE's approach for quantifying the event probabilities and the technical basis for determining the probability estimates assigned to each event type with a probability of 10^{-8} per year or higher are evaluated in SER Section 2.2.1.2.2.

2.2.1.2.2.2 Regulatory Requirements

SER Section 2.2.1.2.2 documents the NRC staff's review and findings on whether DOE complied with the requirements of 10 CFR 63.21(c)(1) and (9), 10 CFR 63.114(a)(1), (2), (4), and (7), 10 CFR 63.114(b), and 10 CFR 63.342.

Section 63.21(c)(1) requires that the SAR must include a description of the Yucca Mountain site, with appropriate attention to those features, events, and processes (FEPs) of the site that might affect performance of the geologic repository. The description of the site must include information regarding FEPs outside of the site to the extent the information is relevant and material to performance of the geologic repository.

Section 63.21(c)(9) requires that the SAR must include an assessment to determine the degree to which those FEPs of the site that are expected to materially affect compliance with 10 CFR 63.113, either beneficial or potentially adverse to performance of the geologic repository, have been characterized, and the extent to which they affect waste isolation. For disruptive events, in particular, specific FEPs of the geologic setting must be investigated outside of the site if they affect performance of the geologic repository.

Section 63.114(a)(1) requires that any performance assessment used for demonstrating compliance with 10 CFR 63.113 for 10,000 years after disposal must include data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the Yucca Mountain site and the surrounding region, to the extent necessary, and information on the design of the engineered barrier system used to define, for 10,000 years after disposal, parameters and conceptual models used in the assessment.

Section 63.114(a)(2) requires that any performance assessment used for demonstrating compliance with 10 CFR 63.113 for 10,000 years after disposal must account for uncertainties and variability in parameter values for 10,000 years after disposal, and provide for the technical basis for parameter ranges, probability distributions, or bounding values used in the performance assessment.

Section 63.114(a)(4) requires that any performance assessment used for demonstrating compliance with 10 CFR 63.113 for 10,000 years after disposal must consider only FEPs consistent with the limits on performance assessment specified at 10 CFR 63.342. Section 63.342(a) requires that a performance assessment used to show compliance with 10 CFR 63.311(a)(1), 10 CFR 63.321(b)(1), and 10 CFR 63.331 not consider very unlikely events (i.e., those that are estimated to have a less than 1 chance in 100 million per year of occurring). Further, 10 CFR 63.342(b), applicable only to the human intrusion and groundwater protection standards, establishes that compliance with 10 CFR 63.321(b)(1) and 10 CFR 63.331 shall exclude unlikely events (i.e., those events that are estimated to have an annual probability of occurring between 1 in 100,000 and 1 in 100 million).

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

Section 63.114(b) states that the performance assessment methods used to satisfy the requirements of 10 CFR 63.114(a) are considered sufficient for the performance assessment for the period of time after 10,000 years and through the period of geologic stability. As defined in

10 CFR 63.302, the period of geologic stability means the time during which the variability of geologic characteristics and their future behavior in and around the Yucca Mountain site can be bounded; that is, they can be projected within a reasonable range of possibilities. This period is defined to end at 1 million years after disposal. Section 63.342(c) specifies how to project the continued effects of FEPs beyond 10,000 years in the performance assessment models to show compliance with 10 CFR 63.311(a)(2) and 10 CFR 63.321(b)(2). Section 63.342(c) requires that DOE's performance assessment shall project the continued effects of the FEPs included in 10 CFR 63.342(a) beyond the 10,000-year post-disposal period through the period of geologic stability. Section 63.342(c)(1) requires that DOE must assess the effects of seismic and igneous activity scenarios, subject to the probability limits in 10 CFR 63.342(a) for very unlikely FEPs, or sequences of events and processes. Section 63.342(c)(1)(i) specifies that the seismic analysis may be limited to the effects caused by damage to the drifts in the repository, failure of the waste packages, and changes in the elevation of the water table under Yucca Mountain. Section 63.342(c)(1)(ii) specifies that the igneous activity analysis may be limited to the effects of a volcanic event directly intersecting the repository and that the igneous event may be limited to that causing damage to the waste packages directly, causing releases of radionuclides to the biosphere, atmosphere, or groundwater.

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

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

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

2.2.1.2.2.3 Technical Review

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

As described in SAR Section 2.2.2, DOE analyzed the human intrusion event in its performance assessment evaluation to demonstrate compliance with 10 CFR 63.321, consistent with the regulatory requirements of 10 CFR 63.322. Section 63.102(l) does not specify a probability for the human intrusion event, but it indicates that the consequences of an assumed intrusion event would be a separate analysis [similar to the performance assessment required by 10 CFR 63.113(b) but subject to specific requirements for evaluation of human intrusion specified in 10 CFR 63.321, 63.322, and 63.342]. Because no probability value is required, the YMRP Acceptance Criteria 2 through 5 listed previously in SER Section 2.2.1.2.2.2 are not applicable. DOE defined the human intrusion event in SAR Section 2.2.2.4.1 and in SAR Table 2.2-6.

The NRC staff reviewed DOE's definition of the human intrusion event to determine whether the event definition is unambiguous and satisfies the regulatory requirements. The NRC staff concludes that the three human intrusion-related FEPs described in SAR Table 2.2-6, which defines the human intrusion event, are acceptable because the FEPs unambiguously describe the event and are consistent with each of the specific requirements for human intrusion. For example, DOE defined the FEP 1.4.02.02.0A "Inadvertent Human Intrusion" in accordance with the regulatory definition of human intrusion (10 CFR 63.302) and following the required regulatory assumptions (10 CFR 63.322). For instance, in DOE's summary of technical basis/approach for FEP inclusion in SAR Table 2.2-6, DOE assumed that no particulate waste material falls into the borehole, in accordance with 10 CFR 63.322(e). DOE also incorporated, as part of the technical basis and approach for including the FEP, the regulatory requirement (10 CFR 63.321) for determining the earliest time after disposal that the waste package would degrade sufficiently that human intrusion could occur without recognition by the drillers. Therefore, the NRC staff finds that DOE has adequately defined the human intrusion event. The NRC staff's evaluation of the performance assessment DOE used to demonstrate compliance with the human intrusion standard (10 CFR 63.321 and 63.322) is documented in SER Section 2.2.1.4.2.

DOE retained igneous activity, seismic activity, and early failure events in its performance assessment to demonstrate compliance with the individual protection standard at 10 CFR 63.311. The NRC staff's evaluations of DOE information for igneous events, seismic events, and early failure events are documented in the following SER Sections 2.2.1.2.2.3.1, 2.2.1.2.2.3.2, and 2.2.1.2.2.3.3 respectively.

2.2.1.2.2.3.1 Igneous Event Probabilities

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

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

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

Summary of DOE's Approach on Igneous Event Probability

DOE evaluated the risk of future igneous activity, in part, by considering the probability that a future igneous event could intersect the repository. To quantify the probability of future igneous activity at the proposed Yucca Mountain repository, DOE conducted an expert elicitation review (probabilistic volcanic hazard assessment, PVHA) (CRWMS M&O, 1996aa). This expert elicitation review resulted in a quantification of the mean annual probability of intersection of the repository by a future basaltic dike (an igneous intrusion) and its associated uncertainty distribution. The NRC staff's evaluation of DOE's expert elicitation on probabilistic volcanic hazard assessment is documented in SER Section 2.5.4.

In SAR Sections 2.2.2.2.1 to 2.2.2.2.5, DOE described the probability of a future igneous event intersecting the repository, the technical basis for the probability estimate, the probability model support including alternative estimates of the intersection probability, the probability model parameters, and the uncertainties associated with the probability estimate, respectively.

Subsequent to the PVHA evaluation, DOE conducted an aeromagnetic survey and drilling program to increase confidence in site characterization results related to igneous activity. As described in SER Section 2.5.4, DOE updated the PVHA evaluation in a study known as the probabilistic volcanic hazard assessment-update (PVHA-U) (SNL, 2008ah). In SAR Section 5.4.1 (DOE, 2009av), DOE stated that the PVHA-U was conducted in a manner that is consistent with NUREG–1563 (NRC, 1996aa) and DOE past practices at Yucca Mountain. The average annual probability for an intrusive igneous event calculated in the PVHA-U is approximately twice as high as calculated in the original PVHA evaluation (Boyle, 2008aa).

DOE stated that the difference between these two event probability distributions would not significantly affect the estimates of repository performance over either 10,000 years or 1 million years, and in SAR Section 5.4.1, DOE further stated that the PVHA-U results are confirmatory of the original PVHA technical basis (DOE, 2009av; also see Boyle, 2008aa).

NRC Staff's Review of Igneous Event Probability

The NRC staff reviewed DOE's igneous event probability presented in SAR Sections 2.2.2.2.2, 2.2.2.2.4, and 2.2.2.2.5. These three SAR sections address the YMRP, Section 2.2.1.2.2 acceptance criteria on technical bases of probability estimates, probability model parameters, and uncertainty in event probability, respectively. The NRC staff's review focused on DOE's information that addressed the event definition (SAR Section 2.2.2.2.1) and probability model support acceptance criteria (SAR Section 2.2.2.2.3), as guided by the YMRP (NRC, 2003aa). The NRC staff's findings on the acceptability of DOE's igneous intrusive event probability are provided next, following the NRC staff's evaluation of event definition and probability model support. This technical review of igneous event probabilities follows the review guidance provided in the YMRP Sections 2.2.1 and 2.2.1.2.2, as supplemented by additional guidance for the period beyond 10,000 years after permanent closure (NRC, 2009ab).

Event Definition

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

The applicant also calculated the proportion of the intersections that include development of a conduit (i.e., an eruption through the repository). The calculation incorporated information from the PVHA and model calculations that are supported by information obtained from studies of analog volcanoes with exposed intrusive rocks from depths of 200–300 m [656–984 ft] (SNL, 2007ae). DOE subsequently calculated that an initial conditional probability of 28 percent of intrusive events would develop a volcanic conduit within the repository footprint (SAR Sections 2.2.2.2.1.3 and 2.3.11.4.2.1). As described in SAR Section 2.3.11.4.2.1.3, this fraction was determined by considering that a conduit can form at any location on a dike that intersects the repository and hence may not necessarily form within the footprint, and by considering several other factors, including the number of dikes in a swarm, consistent with analog basaltic volcanic events that included multiple dikes and in which conduit(s) formed on the widest dike. DOE then applied a second conditional probability of 29.7 percent (SAR Section 2.3.11.4.2.1) to represent the fraction of conduits that may intersect a drift containing waste packages and eject the waste contents through a volcanic vent. Hence, the mean recurrence frequency for DOE's volcanic eruption modeling case is 1.4×10^{-9} per year (SAR Section 2.3.11.4.2.1; see also SER Section 2.2.1.4.1.3.2.1).

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

The probability of an intrusive disruption of the repository differed between the original PVHA and its update by a factor of 1.8 (1.7×10^{-8} versus 3.1×10^{-8} per year, respectively). The event definition for the extrusive (volcanic) case in the PVHA-U was the formation of a conduit within the repository that would support an (explosive) eruption column; hence, it was different and more specific than the original PVHA conditional probability of a conduit forming. The probability of an eruptive conduit event developing within the repository differed between the PVHA and its update by a factor of 2.5 (4.8×10^{-9} versus 1.2×10^{-8} per year, respectively). However, the two values are not directly comparable for the reason stated previously (Boyle, 2008aa). Therefore, any potential effects of inconsistencies related to event definitions within the PVHA and the PVHA-U evolutions have less than a factor of two effect on the uncertainty associated with the mean annual probability. Given the breadth of new information considered in the PHVA-U, the NRC staff concludes that the potential effect of uncertainty in event definitions would have less than a factor of two effect on the estimated mean annual igneous event probability. Further, because the doses DOE calculated for the intrusive and extrusive (volcanic) igneous cases (SAR Section 2.4) are on the order of 0.01 mSv/yr [1 mrem/yr] or less, two or more orders of magnitude below the dose limits, the NRC staff finds that an approximate doubling of the igneous probabilities would have no significance to risk (SNL, 2008ah; Boyle, 2008aa). Further information and review of the intrusive and extrusive (volcanic) igneous scenarios are provided by DOE in SAR Section 2.3.11 and by the NRC staff's evaluation in SER Section 2.2.1.3.10.

On the bases of (1) the information provided in SAR Section 2.2.2.2; (2) DOE's statement that the updated PVHA results confirmed the conclusions of the original PVHA technical basis, as stated in the SAR Revision 1, Section 5.4.1 (DOE, 2009av); and (3) NRC staff's independent evaluation, the NRC staff finds that DOE has acceptably defined an igneous intrusive event and an extrusive (volcanic eruptive) event that may affect the proposed repository. The NRC staff finds that these definitions are consistent with their use in the performance assessment for the igneous abstraction.

Probability Model Support

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

DOE has also indicated that past volcanic activity has occurred in the tectonically active Yucca Mountain region and could continue into the future with a very small probability of occurrence. During the period from 14 to 10 million years ago, major explosive eruptions involving rhyolite (silicic) magma from volcanic centers lying roughly 20–40 km [12–25 mi] north of the repository site formed large caldera volcanoes and deposited the volcanic ash-flow tuffs of the region, some of which are the host rocks for the proposed repository. DOE concluded that the annual chance of a recurrence within the repository lifetime is less than 1 in 10,000 during 10,000 years (SAR Sections 2.2.2.2.2.1 and 2.3.11.2.1.1; see also Detournay, et al., 2003aa) because there has been a sufficient time lapse since caldera-forming volcanic activity ended about 8 million years ago (BSC, 2004bi). The NRC staff notes that a caldera-forming volcanic activity typically lasts 2–4 million years and does not recur in the same location, and therefore, it is not likely that a caldera-forming volcanic activity will happen again in the Yucca Mountain region in the next million years. On the basis of available literature on the age and duration of this major phase of caldera-forming volcanism in the Yucca Mountain region (Sawyer, et al., 1994aa; Fleck, et al., 1996aa) and the NRC staff's knowledge of the lifespans of caldera volcanoes in other parts of the world (e.g., Costa, 2008aa), the NRC staff finds that DOE's conclusion has an adequate technical basis.

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

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

To support the PVHA estimates developed prior to 2002, DOE characterized known basaltic igneous features within approximately 80.5 km [50 mi] of the Yucca Mountain site (BSC, 2004af). These investigations provided the location, age, and basic characteristics necessary to support probability estimates in the probabilistic volcanic hazard assessment evaluation. DOE also conducted geophysical investigations and borehole drilling to further characterize buried igneous features in the Yucca Mountain region (O'Leary, et al., 2002aa;

Perry, et al., 2005aa). This new information was considered in the PVHA-U evaluation (SNL, 2008ah). The NRC staff reviewed the information in these documents and determined that DOE provided sufficient support for the probability model development, because the information provided an appropriate level of detail on the location, age, and basic characteristics of igneous features in the Yucca Mountain region.

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

DOE discussed alternative estimates of the annual probability of an intrusive event intersecting the repository footprint (SAR Section 2.2.2.2.3.2). Both the NRC staff and the State of Nevada independently sponsored the development of these published models (SAR Table 2.2-18). Some of these models use methods and data developed after the 1996 probabilistic volcanic hazard assessment elicitation. Annual probability estimates for these published models range from 3×10^{-10} to 3×10^{-7}. DOE stated that these values cluster at slightly greater than 10^{-8} per year (SAR Section 2.2.2.2.3.2) and concluded that the apparent clustering near 10^{-8} per year provides confidence that the PVHA estimate is robust. The NRC staff concludes that the published values encompass a similar range to that of the PVHA, and that the published probability estimates for igneous events generally fall within the range of 10^{-8} to 10^{-7} per year. This range coincides with the PVHA mean annual probability estimate.

In the discussion of probability model support (SAR Section 2.2.2.2.3.2), DOE did not address published models by Ho and Smith (1997aa) and Ho, et al. (2006aa). In response to the NRC staff's RAI (DOE, 2009bd), DOE concluded that the calculations in Ho and Smith (1997aa) were performed as sensitivity analyses, which included parameter ranges selected from either expert knowledge or for "mathematical convenience," as stated in Ho and Smith (1997aa, p. 621). DOE (2009bd) stated that the probability model approach developed in Ho and Smith (1997aa) was captured in the range of probability estimates that Ho and Smith (1998aa) presented subsequently, and which DOE considered in the SAR. The NRC staff reviewed the information in DOE's response and concludes that the probability estimates in Ho and Smith (1998aa) reasonably represent the probability models developed in Ho and Smith (1997aa) using expert knowledge. Therefore, the NRC staff finds that DOE has appropriately considered the results from Ho and Smith (1997aa) in establishing confidence that DOE's probability estimate is robust. Although, DOE stated that Ho, et al. (2006aa) did not present disruption probability results (DOE, 2009bd), a probability estimate of 10^{-7} is contained in Ho, et al., p. 121 (2006aa) on the basis of those author's interpretation of recurrence rates given in Smith, et al. (2002aa). The NRC staff finds that the probability estimate in Ho, et al. (2006aa) is consistent with the information DOE presented in SAR Section 2.2.2.2.3.2 and would not significantly affect the rationale DOE presented to support confidence in DOE's probability estimate.

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

NRC Staff's Evaluation of DOE's Approach on Igneous Event Probability

On the basis of its review of the information DOE provided, the NRC staff's evaluation of the PVHA expert elicitation process in SER Section 2.5.4 and the preceding review, the NRC staff makes the following findings.

The NRC staff finds that DOE has acceptably evaluated separate probabilities, including uncertainties, for the igneous intrusion and volcanic eruptive events.

The NRC staff finds that the uncertainties in DOE's probability estimates (i.e., probabilistic volcanic hazard assessment as provided by the PVHA and supported by the PVHA-U results) are not risk significant for the following reasons:

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

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

- DOE’s performance assessment calculations [SAR Figures 2.4-18(a) and 2.4-18(b)] show that the expected annual dose from igneous events is at least two orders of magnitude lower than (i.e., less than 1 percent) the dose standard in the regulation.

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

- Model or data uncertainties represented by an increase in mean annual probability to 10^{-7} would increase the expected annual dose from igneous events to less than 6 percent of the dose standard.

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

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

2.2.1.2.2.3.2 Seismic Event Probabilities

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

Summary of DOE's Approach on Seismic Event Probability

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

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

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

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

NRC Staff's Review of Seismic Event Probability

DOE applied the above three steps to the preclosure seismic design and safety analyses as well as to its postclosure performance assessment. The NRC staff documented its evaluation of Step 1 in SER Section 2.5.4. The NRC staff's review and evaluation of those aspects of DOE's seismic hazard assessment (Steps 2 and 3) that are pertinent to postclosure performance assessment, including evaluations needed to address the five event probability acceptance criteria in the YMRP Section 2.2.1.2.2, as supplemented by additional guidance for the period beyond 10,000 years after permanent closure (NRC, 2009ab), are documented in this SER Section.

DOE's Probabilistic Seismic Hazard Assessment Expert Elicitation

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

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

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

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

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

Probabilistic Fault Displacement Hazard Assessment

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

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

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

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

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

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

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

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

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

The applicant’s expert elicitation process is reviewed and evaluated in SER Section 2.5.4, in which the NRC staff found the process acceptable. On the basis of the expert elicitation process performed to support the fault displacement hazard estimate in the license application, the NRC staff finds that results of the probabilistic fault displacement hazard assessment are appropriate and acceptable.

Conditioning of Low Probability Ground Motions

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

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

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

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

The unconditioned hazard curve, which is the annual probability of exceedance as a function of ground motion, is convolved with the distribution of extreme ground motion for the reference rock outcrop to produce the conditioned ground motion hazard of the same rock outcrop. SAR

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

NRC Staff's Evaluation of DOE's Approach on Seismic Event Probability

The findings and conclusions discussed next are based on the information DOE provided, the NRC staff's evaluation of the probabilistic seismic hazard assessment expert elicitation process in SER Section 2.5.4, and the preceding review.

The NRC staff reviewed the information in the SAR with regard to the applicant's probabilistic seismic hazard assessment and probabilistic fault displacement hazard assessment and finds that these assessments are acceptable. The NRC staff's conclusion regarding the applicant's probabilistic seismic hazard assessment and probabilistic fault displacement hazard assessment is based on the following findings.

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

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

Therefore, because the applicant relied on the collective judgment of established experts, followed an acceptable procedure to elicit and document the experts' conclusions, and supported the experts' elicitation with sufficient technical and scientific information, the results of the elicitations are adequate for use in the other portions of the license application. The NRC staff's conclusion is supported by the following detailed evaluation of the applicant's information against the five acceptance criteria described in YMRP Section 2.2.1.2.2, as supplemented by additional guidance for the period beyond 10,000 years after permanent closure (NRC, 2009ab).

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

The NRC staff finds that DOE's probabilistic seismic hazard assessment, the conditioning of the probabilistic seismic hazard assessment ground motions, and probabilistic fault displacement hazard assessment are supported by appropriate technical bases that include the expert elicitation program in which the experts considered the full range of information available when the probabilistic seismic hazard assessment was developed. During the expert elicitation process, the seismic source teams considered a range of information provided by DOE, the U.S. Geological Survey, other project-specific Yucca Mountain studies, and information published in the scientific literature. This information included data and models for the geologic and seismotectonic setting, seismic sources, historical and instrumented seismicity, earthquake recurrence, maximum magnitude, and ground motion attenuation. Detailed evaluations of this information are provided in NUREG–1762 (NRC, 2005aa). The NRC staff finds that DOE's elicitation process (see the NRC staff's evaluation in SER Section 2.5.4) for the probabilistic seismic hazard assessment was implemented in accordance with NRC guidance in NUREG–1563 (NRC, 1996aa). On the basis of the adequacy of the probabilistic seismic hazard assessment expert elicitation process and supporting information, the NRC staff finds that DOE's probabilistic seismic hazard assessment and probabilistic fault displacement hazard assessment programs adequately characterized the seismic and fault displacement hazards at Yucca Mountain. Additional geological, geophysical, and seismological information discovered since the elicitation was performed (e.g., Hanks, et al., 2013) is consistent with the probabilistic seismic hazard assessment results (except for the overly conservative result on large ground motions at low annual exceedance, as previously described, regarding the conditioning of low probability ground motions).

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

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

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

In addition, the NRC staff finds that the applicant appropriately considered new information acquired since the development of the probabilistic seismic hazard assessment in 1998. In particular, the NRC staff finds that the applicant's conditioned hazard curves, which reflect geological and seismological information that suggests limits on the low probability ground motions in the probabilistic seismic hazard assessment, are acceptable. The NRC staff finds that the applicant's conditioning approach follows appropriate basic mechanical and material behaviors consistent with the current understanding of seismological phenomena. The applicant's assumption that the tectonic setting and therefore the stress drops of earthquakes from the existing faults at Yucca Mountain are not going to change significantly in the next million years is also reasonable on the basis of the NRC staff's understanding of the seismotectonic history of the Yucca Mountain region. In SER Section 2.2.1.3.2, the NRC staff further determined that, on the basis of physical constraints and improved ground motion attenuation relations, a recent study on the extreme high ground motion (Hanks, et al., 2013) indicates that the seismic hazards at Yucca Mountain are substantially lower than the seismic hazard values developed by DOE in the license application for postclosure performance assessment, thereby rendering the seismic inputs currently used in the TSPA conservative.

2.2.1.2.2.3.3 Early Waste Package and Drip Shield Failures Event Probabilities

This section reviews and evaluates information DOE presented to estimate the probability of early failure of waste packages and the drip shield at the proposed repository site. This technical review of early failure event probabilities follows the review guidance provided in YMRP Sections 2.2.1 and 2.2.1.2.2. The NRC staff reviewed SAR Sections 1.3.4, 1.5.2, 2.2.2.3, 2.3.6.6, and 2.3.6.8.4, and additional information provided in response to the NRC staff's RAIs and the references cited therein.

Summary of DOE's Approach on Early Waste Package and Drip Shield Failures Event Probabilities

In SAR Section 2.2.2.3.1, DOE defined early failure of a waste package or drip shield as through-wall penetration of the barrier caused by the presence of manufacturing- or handling-induced defects at a time earlier than would be predicted by mechanistic degradation models for a defect-free waste package or drip shield.

In SAR Section 2.2.2.3.2, DOE summarized the early waste package failure probability and stated that the probability values are based on the waste package fabrication and handling processes described in SAR Section 1.5.2. DOE further stated that details of technical bases for the probability estimates, including the parameters and data used and their associated uncertainties, are described in SAR Section 2.3.6.6. In SAR Section 2.2.2.3.3, DOE summarized the early drip shield failure probability and stated that the probability values are based on the drip shield fabrication and handling processes described in SAR Section 1.3.4. DOE further stated that details of technical bases for the probability estimates, including the parameters and data used and their associated uncertainties, are described in SAR Section 2.3.6.8.4.

DOE's approach for early failure probability calculations is to quantify errors in manufacturing or handling of waste packages or drip shields, and to quantify the potential that the errors go undetected prior to emplacement. In such instances, the defective waste package or drip shield is assumed to experience early failure.

DOE systematically identified the types of errors or defects that could lead to early failures of the waste package and drip shield and reviewed the technical literature for empirical data of similar systems and components (i.e., industrial analogues). DOE identified five industrial analogues, which can generally be described as welded metallic containers: (i) boilers and pressure vessels, (ii) nuclear fuel rods, (iii) underground storage tanks, (iv) radioactive cesium capsules, and (v) dry storage casks for spent nuclear fuel (SNF). For these analogues, DOE obtained qualitative and quantitative information on the types of manufacturing and handling errors that may occur and their associated frequency of occurrence, as identified in SNL (2007aa, Section 6.1).

SAR Table 2.3.6-21 identifies the specific types of defects and their occurrence rates for these analogues. From these industrial analogues, DOE developed a list of 13 generic errors or defects that could lead to early failure of welded metallic containers (SAR Section 2.3.6.6.2.1).

Given that the industrial analogues are only partly analogous to the waste package and drip shield in terms of manufacturing techniques, intended safety function, and operating environment, DOE determined that only some of the generic defects applicable to welded metallic containers are applicable to the waste package and drip shield, as identified in SAR Sections 2.3.6.6.3.1 and 2.3.6.8.4.3.1, and described in SNL (2007aa, Section 6.1.6). DOE evaluated the defect types and eliminated from further consideration those defects not applicable to the waste package and drip shield (SAR Sections 2.3.6.6.3.1 and 2.3.6.8.4.3.1). DOE considered that weld flaws, particularly in the waste package closure weld, could affect the performance of the waste package, but would not necessarily lead to early failure. It considered weld flaws as potential initiation sites for stress corrosion cracking. SAR Section 2.3.6.5 addressed weld flaws and is evaluated in SER Section 2.2.1.3.1.3.2.3.

For the waste package, DOE identified six types of defects or errors that could lead to early failure (SAR Section 2.3.6.6.3.2). These waste package errors are: (i) improper base metal selection; (ii) improper weld filler material selection; (iii) improper heat treatment of the outer corrosion barrier; (iv) improper heat treatment of the outer lid; (v) improper low-plasticity burnishing; and (vi) improper handling. For the drip shield, DOE identified four types of defects or errors that could lead to early failure (SAR Section 2.3.6.8.4.3.2). These drip shield errors are: (i) base metal flaws; (ii) improper weld filler material; (iii) improper heat treatment; and (iv) improper handling and installation. Those defects that could lead to early failure of the waste package or failure of the drip shield were further analyzed. DOE developed event trees to identify event sequences that could lead to undetected defects or errors in the waste package and drip shield as identified in SNL Sections 6.3 and 6.4 (2007aa), respectively. The event sequences generally consist of an equipment or process failure event followed by human error event(s), where the equipment or process failure is undetected or uncorrected. To quantify the probabilities for the respective event sequences, each basic event in the sequences was assigned a probability distribution. For the equipment or process failure events, the probability distribution was based on data that were generated from similar components or processes at nuclear power plants (e.g., Blanton and Eide, 1993aa). For human reliability data, the probability distributions were taken from data for nuclear power plant activities (Swain and Guttmann, 1983aa; Benhardt, et al., 1994aa).

DOE used Monte Carlo simulations to analyze the event trees and calculate the probability distributions for event sequences that could lead to undetected errors or defects in the waste package or drip shield. DOE described end state probabilities for event sequences as lognormal distributions. DOE calculated the overall mean probability that waste packages contain at least one undetected defect (i.e., the waste package early failure probability) by summing the mean probabilities of independent event sequences that could lead to the presence of an undetected defect. DOE followed the same process for the drip shield to calculate the overall mean probability that the drip shields contain at least one undetected defect (i.e., the drip shield early failure probability). In response to the NRC staff's RAI (DOE, 2009ac), DOE clarified that the mean probabilities for the individual event sequences leading to early failure of the waste package, as reported in SAR Sections 2.3.6.6.3.2.1 to 2.3.6.6.3.2.6, were incorrect, and that the mean probabilities for the individual event sequences leading to early failure of the drip shield, as reported in SAR Sections 2.3.6.8.4.3.2.1 to 2.3.6.8.4.3.2.4, were also incorrect. DOE's response also provided corrected values for these mean probabilities for SAR Sections 2.3.6.6.3.2.1 to 2.3.6.6.3.2.6 and SAR Sections 2.3.6.8.4.3.2.1 to 2.3.6.8.4.3.2.4. Furthermore, DOE stated that the early failure probabilities for the waste package and drip shield listed in SAR Sections 2.3.6.6.3.2.7 and 2.3.6.8.4.3.2.5, respectively, are correct. DOE described the early failure probability for the waste package as a lognormal distribution with a mean of 1.13×10^{-4} per waste package and an error factor of 8.17 (SAR Sections 2.2.2.3.2 and 2.3.6.6.3.2.7). DOE described the early failure probability for the drip shield as a lognormal distribution with a mean of 2.21×10^{-6} per drip shield and an error factor of 14 (SAR Sections 2.2.2.3.3 and 2.3.6.8.4.3.2.5).

DOE compared its probability estimates for early failure of the waste package and drip shield, respectively, with the defect-related failure rates for the industrial analogues. The failure rates for the industrial analogues for pressure vessels, nuclear fuel rods, underground storage tanks, and radioactive cesium capsules DOE cited ranged from 10^{-6} to 10^{-4} per component (SAR Table 2.3.6-21). DOE did not identify any cases of SNF casks that failed due to undetected defects after entering service.

The probability estimates for early failure of the waste package and drip shield are implemented in the TSPA model in the Early Failure Scenario Class, as described in SAR Section 2.4.1.2.2. This implementation is reviewed by the NRC staff in SER Section 2.2.1.4.1. The NRC staff's review of the implementation of the model abstraction for early failure is documented in SER Sections 2.2.1.3.1.3.1.2 and 2.2.1.3.1.3.2.4.

NRC Staff's Review and Evaluation of DOE's Approach on Early Waste Package and Drip Shield Failures Event Probabilities

The NRC staff finds that DOE defined the early failure probability event without ambiguity. Early failure refers to through-wall penetration of the waste package or drip shield at a time earlier than the design life because of undetected manufacturing- or handling-induced defects. Early failure is distinguished from other events and processes that could lead to through-wall penetration (e.g., corrosion, tensile rupture). Further, the event definition is consistent with the barrier functions of the waste package and drip shield, respectively, as stated in SAR Table 1.9-8. Therefore, the events are adequately defined.

The NRC staff reviewed DOE's assumption that the early failure probabilities for the waste package and drip shield are equivalent to the probabilities that there are undetected manufacturing- or handling-induced defects in the respective barriers. The NRC staff finds that this assumption is appropriate for the following reasons. NRC staff notes, from empirical observation of industrial analogues, that a waste package or drip shield with a manufacturing- or handling-induced defect will likely maintain some barrier capability [as stated by DOE in SNL (2007aa, Section 6.1)], and that the presence of a defect, in itself, is unlikely to cause through-wall penetration of an engineered barrier without an additional degradation mechanism. Therefore, the NRC staff finds that DOE's assumption that the probability of undetected manufacturing- or handling-induced defects in waste packages and drip shields is equivalent to the probability of through-wall defects is conservative.

The NRC staff reviewed DOE's use of industrial analogues to identify the generic types of defects or errors that, if undetected, could lead to early failure. The NRC staff finds acceptable that DOE identified industrial analogues to substantiate the probability of defects potentially leading to early failure because those analogues to waste packages and drip shields have similar construction material, geometry (generally cylindrical or spherical in shape), and fabrication approach (e.g., welded, heat treated), and because the analogues are closed and designed to act as a barrier or container.

Defects and Errors Eliminated From the Early Failure Probability Analysis

The NRC staff reviewed DOE's decision to eliminate from further consideration some of the generic errors and defects from the early failure probability models and reached the following conclusions:

- The NRC staff finds DOE's decision to eliminate improper weld-flux material from the early failure analyses for the waste package and drip shield acceptable because the welding method to be employed for waste packages (SAR Section 1.9.2) and the welding method to be employed for drip shield (SNL, 2007aa, Section 6.2.3) do not use weld-flux material.

- The NRC staff finds DOE's decision to eliminate weld flaws from the drip shield and waste package early failure analyses acceptable because (i) SAR Section 1.3.4.7 stated that the drip shield will be fully stress relieved, hence preventing stress corrosion crack initiation from weld flaws; and (ii) the waste package is solution annealed to remove welding stresses, meaning that only waste package closure weld flaws will act as possible stress corrosion cracking locations. Waste package closure weld flaws are modeled in SAR Section 2.3.6.5 as part of the stress corrosion cracking model (not part of the early failure) and is evaluated in SER Section 2.2.1.3.1.3.2. The NRC staff confirmed that Postclosure Design Control Parameter 07-13 in SAR Table 1.9-9 stated that the drip shield will be stress relieved. The NRC staff determined that without external stress, weld flaw propagation is unlikely in a stress-relieved drip shield. Therefore, the NRC staff finds that use of the Postclosure Design Control Parameters provides an acceptable basis for DOE to eliminate weld flaws from the drip shield early failure analysis.

- The NRC staff also finds DOE's decision to eliminate poor weld-joint design from further early failure analyses for the waste package and drip shield acceptable because controls specified in SAR Section 1.9.2 provide for extensive examination of waste package and drip shield joints. The NRC staff confirmed that Postclosure Design Control Parameters in SAR Table 1.9-9 (03-12 and 03-14 for the waste package, and 07-09 and 7-10 for the drip shield) provide that fabrication welds are conducted in accordance with standard nuclear industry practice, including inspection and nondestructive examination. Therefore, the NRC staff finds that use of the Postclosure Design Control Parameters provides an acceptable basis for DOE to eliminate poor weld-joint design from the drip shield and waste package early failure analyses.

- The NRC staff finds DOE's decision to eliminate missing welds from further early failure analyses for the waste package and drip shield acceptable because controls specified in SAR Section 1.9.2 provide for extensive inspection and nondestructive examination of the welds. The NRC staff confirmed that Postclosure Design Control Parameters in SAR Table 1.9-9 (03-13 and 03-14 for the waste package, and 07-09 and 7-10 for the drip shield) provide that fabrication welds be conducted in accordance with standard nuclear industry practice, including inspection and nondestructive examination. Because Postclosure Design Control Parameters 03-13, 03-14, 07-09 and 07-10 in SAR Table 1.9-9 provide for extensive inspection and nondestructive evaluation of waste packages and drip shields, the NRC staff finds that use of the Postclosure Design Control Parameters provides an acceptable basis for DOE to eliminate missing welds from the drip shield and waste package early failure analyses.

- The NRC staff finds DOE's decision to eliminate mislocated welds from further early failure analyses for the waste package and drip shield acceptable because controls specified in SAR Section 1.9.2 provide for extensive inspection and nondestructive examination of the welds. The NRC staff confirmed that Postclosure Design Control Parameters in SAR Table 1.9-9 (03-13 and 03-14 for the waste package, and 07-09 and 07-10 for the drip shield) provide that fabrication welds be conducted in accordance with standard nuclear industry practice, including inspection and nondestructive examination. Therefore, because the Postclosure Design Control Parameters 03-13, 03-14, 07-09 and 07-10 in SAR Table 1.9-9 provide for extensive inspection and nondestructive evaluation of waste packages and drip shields, the NRC staff finds that use of the Postclosure Design Control Parameters provides an acceptable basis for DOE to eliminate mislocated welds from the drip shield and waste package early failure analyses.

- The NRC staff finds DOE's decision to eliminate surface contaminants (e.g., material that could enhance the corrosion rate) from further early failure analyses for the waste package and drip shield acceptable because controls specified in SAR Section 1.9.2 provide that fabrication and handling processes will limit the type and amount of surface contamination. The NRC staff confirmed that Postclosure Design Control Parameters in SAR Table 1.9-9 (03-21 for the waste package and 07-14 for the drip shield) provide for stringent controls on waste package and drip shield fabrication and handling. Therefore, the NRC staff finds that use of the Postclosure Design Control Parameters provides an acceptable basis for DOE to eliminate surface contaminants from the drip shield and waste package early failure analyses.

- The NRC staff finds DOE's decision to eliminate improper low-plasticity burnishing from the drip shield early failure analysis acceptable because the drip shield is not low-plasticity burnished. The NRC staff reviewed the description of the drip shield design in SAR Section 1.3.4.7 and confirmed that drip shield welds will not be low-plasticity burnished.

- The NRC staff finds DOE's decision to eliminate handling damage from early failure analysis for the drip shield acceptable because the high strength-to-weight ratio of titanium makes it resilient to scratches and denting from handling-induced impacts and because DOE will control drip shield handling. The NRC staff confirmed that Postclosure Design Control Parameter 07-14 in SAR Table 1.9-9 provides for controls on drip shield handling that will minimize damage and impacts to the drip shield, and drip shield emplacement will be monitored by equipment that can detect damage. Therefore, the NRC staff finds that use of Postclosure Design Control Parameters provides an acceptable basis for DOE to eliminate handling damage from early failure analysis for the drip shield.

- The NRC staff finds DOE's decision to eliminate administrative or operational errors as distinct errors in the waste package and drip shield early failure analyses acceptable because DOE implicitly incorporated such errors (e.g., failure to follow a written procedure) into the analyses of the defects that were not screened out.

Defects and Errors Included in the Early Failure Probability Analysis

The NRC staff reviewed the event trees and event sequences DOE used to calculate the probabilities for the errors that could cause early failure. The NRC staff reviewed the extent to which DOE identified key processes involved with waste package and drip shield handling and manufacturing, and whether the event sequences were appropriate and realistic to estimate the undetected defect (i.e., early failure) probabilities. The NRC staff makes the following specific conclusions on DOE's event trees and event sequences used to calculate probabilities.

<u>Waste Package</u>

- The NRC staff reviewed DOE's event tree for waste package fabrication with improper base metal selection, which is shown in SNL (2007aa, Figure 6-9). In response to the NRC staff's RAI (DOE, 2009ac), DOE stated that the composition of the base metal will be certified by the supplier and independently checked upon receipt at the fabrication facility. The NRC staff finds that DOE identified the key processes for this event sequence and that the event sequence is appropriate and realistic because DOE acceptably described the processes and stated that fabrication will be accomplished under stringent controls and in accordance with standard nuclear industry practices. In this regard, the NRC staff further finds acceptable DOE's reliance on Postclosure Design Control Parameter 03-19 in SAR Table 1.9-9, which specifies the waste package outer corrosion barrier material specifications, and DOE's use of Postclosure Design Control Parameter 03-02, which provides that the waste package material be controlled by the configuration management system. Therefore, the NRC staff concludes that the event sequence for this defect is acceptable because DOE identified the key processes for this event sequence and adequately described the processes and their controls.

- The NRC staff reviewed DOE's event sequence for waste package fabrication with improper weld filler material selection, which is shown in SNL (2007aa, Figure 6-14). In response to the NRC staff's RAI (DOE, 2009ac), DOE stated that the composition of the weld filler metal will be certified by the supplier and independently checked upon receipt at the fabrication facility. The NRC staff finds that DOE identified the key processes for this event sequence. The NRC staff further finds that the event sequence is appropriate and realistic because DOE acceptably described the processes and stated that fabrication will be accomplished under stringent controls and in accordance with standard nuclear industry practices. In this regard, the NRC staff finds acceptable DOE's reliance on Postclosure Design Control Parameter 03-14 in SAR Table 1.9-9, which specifies that the waste package fabrication welds shall be conducted in accordance with standard nuclear industry requirements, and the NRC staff finds acceptable DOE's use of Postclosure Design Control Parameter 03-02, which provides that the waste package material be controlled by the configuration management system. Therefore, the NRC staff concludes that the event sequence for this defect is acceptable because DOE identified the key processes for this event sequence and adequately described the processes and their controls.

- The NRC staff reviewed DOE's event sequence for waste package fabrication with improper heat treatment for the waste package outer shell, which is shown in SNL (2007aa, Figure 6-10). The NRC staff finds that DOE identified the key processes for this event sequence. For example, in SNL (2007aa, Section 6.3.3), DOE stated that the critical steps during heat treatment are moving the heated shell from the furnace to the quench tank and the subsequent quench. The NRC staff finds that the event sequence is appropriate and realistic because DOE acceptably described the processes and stated that fabrication will be accomplished under stringent controls. The NRC staff finds acceptable DOE's reliance on Postclosure Design Control Parameter 03-16 in SAR Table 1.9-9, which specifies the waste package heat treatment conditions. Therefore, the NRC staff concludes that the event sequence for this defect is acceptable because DOE identified the key processes for this event sequence and adequately described the processes and their controls.

- The NRC staff reviewed DOE's event sequence for waste package fabrication with an improperly heat-treated outer lid, which is shown in SNL (2007aa, Figure 6-11). The NRC staff finds that DOE identified the key processes for this event sequence. For example, in SNL (2007aa, Section 6.3.4), DOE stated that the critical steps during heat treatment are moving the heated lid from the furnace to the quench tank and the subsequent quench. The NRC staff finds that the event sequence is appropriate and realistic because DOE acceptably described the processes and stated that fabrication will be accomplished under stringent controls. The NRC staff finds acceptable DOE's reliance on Postclosure Design Control Parameter 03-16 in SAR Table 1.9-9, which specifies the waste package heat treatment conditions. Therefore, the NRC staff concludes that the event sequence for this defect is acceptable because DOE identified the key processes for this event sequence and adequately described the processes and their controls.

- The NRC staff reviewed DOE's event sequence for waste package fabrication with improper low-plasticity burnishing of the closure weld, which is shown in SNL (2007aa, Figure 6-12). The NRC staff finds that DOE identified the key processes for this event sequence. For example, in SNL (2007aa, Section 6.3.5), DOE stated that burnishing will be performed by a dedicated, automated system, with subsequent inspection to assure that the appropriate procedures were followed. The NRC staff finds that the event sequence is appropriate and realistic because DOE acceptably described the processes and stated that fabrication will be accomplished under stringent controls. The NRC staff finds acceptable DOE's reliance on Postclosure Design Control Parameter 03-17 in SAR Table 1.9-9, which provides for process controls to ensure adequate stress relief, along with subsequent nondestructive examination. Therefore, the NRC staff concludes that the event sequence for this defect is acceptable because

DOE identified the key processes for this event sequence and adequately described the processes and their controls.

- The NRC staff reviewed DOE's event sequence for improper handling of the waste package, which is shown in SNL (2007aa, Figure 6-13). The NRC staff finds that DOE identified the key processes for this event sequence. For example, in SAR Section 2.3.6.6.3.2.5 and DOE (2009ad), DOE defined damage as any visible gouging or denting of the waste package surface that occurs between receipt and drip shield installation. Damage would be any such gouging or denting that could jeopardize the performance of the outer barrier. Because handling procedures have not been fully developed, DOE assumed that the waste package could be damaged by any one of eight generic events, each of which is analogous to fuel assembly handling events at nuclear power plants. In response to the NRC staff's RAI (DOE, 2009ac), DOE stated that this comparison is appropriate because fuel assemblies are handled in tightly controlled conditions similar to those expected at the repository. The NRC staff finds that the event sequence is appropriate and realistic because DOE acceptably described the processes and stated that handling will be accomplished under stringent controls. In this regard, the NRC staff finds acceptable DOE reliance on Postclosure Design Control Parameters 03-18, 03-21, and 03-22 in SAR Table 1.9-9 of the SAR. These Postclosure Design Control Parameters provide for the waste package to be handled in a controlled manner to minimize damage, including inspection for surface damage prior to emplacement and monitoring during emplacement activities. Therefore, the NRC staff concludes that the event sequence for this defect is acceptable because DOE identified the key processes for this event sequence and adequately described the processes and their controls.

<u>Drip Shield</u>

- The NRC staff reviewed DOE's event sequence for drip shield fabrication with out-of-specification base metal, which is shown in SNL (2007aa, Figure 6-16). In response to the NRC staff's RAI (DOE, 2009ac), DOE stated that the composition of the base metal will be certified by the supplier and independently checked upon receipt at the fabrication facility. The NRC staff finds that DOE identified the key processes for this event sequence and that the event sequence is appropriate and realistic because DOE acceptably described the processes and stated that fabrication will be accomplished under stringent controls and in accordance with standard nuclear industry practices. The NRC staff finds acceptable DOE's reliance on Postclosure Design Control Parameter 07-09 in SAR Table 1.9-9, which specifies that the drip shield shall be fabricated in accordance with standard nuclear industry practices, including those for material control. The NRC staff also finds acceptable DOE's use of Postclosure Design Control Parameter 07-01 in SAR Table 1.9-9, which provides that the drip shield materials be controlled by the configuration management system. Therefore, the NRC staff concludes that the event sequence for this defect is acceptable because DOE identified the key processes for this event sequence and adequately described the processes and their controls.

- The NRC staff reviewed DOE's event sequence for drip shield fabrication with out-of-specification weld filler metal, which is shown in SNL (2007aa, Figure 6-18). In response to the NRC staff's RAI (DOE, 2009ac), DOE stated that the composition of the base metal will be certified by the supplier and independently checked upon receipt at the fabrication facility. The NRC staff finds that DOE identified the key processes for this event sequence and that the event sequence is appropriate and realistic because DOE acceptably described the processes and stated that fabrication will be accomplished under stringent controls and in accordance with standard nuclear industry practices. In this regard, the NRC staff finds acceptable DOE's reliance on Postclosure Design Control Parameter 07-09 in SAR Table 1.9-9, which specifies that the drip shield shall be fabricated in accordance with standard nuclear industry practices, including those for material control and welding. The NRC staff also finds acceptable DOE's use of Postclosure Design Control Parameter 07-01 in SAR Table 1.9-9, which provides that the drip shield materials be controlled by the configuration management system. Therefore, the NRC staff concludes that the event sequence for this defect is acceptable because DOE identified the key processes for this event sequence and adequately described the processes and their controls.

- The NRC staff reviewed DOE's event sequence for drip shield fabrication with improper heat treatment, which is shown in SNL (2007aa, Figure 6-17). The NRC staff finds that DOE identified the key processes for this event sequence. For example, in SNL (2007aa, Section 6.4.2), DOE stated that the drip shield temperature during heat treatment will be monitored by calibrated thermocouples in contact with the material and that the drip shield will be subject to a post-heat-treatment inspection to ensure that the heat-treatment procedure was properly followed. The NRC staff finds that the event sequence is appropriate and realistic because DOE acceptably described the processes and stated that fabrication will be accomplished under stringent controls. In this regard, the NRC staff finds acceptable DOE's reliance on Postclosure Design Control Parameters 07-09 and 07-13 in SAR Table 1.9-9. The Postclosure Design Control Parameters (i) provide that the drip shield heat treatment be performed in a manner consistent with standard nuclear industry practice and (ii) specify the drip shield heat treatment conditions. Therefore, the NRC staff concludes that the event sequence for this defect is acceptable because DOE identified the key processes for this event sequence and adequately described the processes and their controls.

- The NRC staff reviewed DOE's event sequence for improper drip shield installation, which is shown in SNL (2007aa, Figure 6-19). The NRC staff finds that DOE identified the key processes for this event sequence. For example, in SNL (2007aa, Section 6.4.4), DOE stated that drip shields will be visually inspected at the surface facilities, that emplacement activities will be monitored by camera, and that the inspections will be independently checked and documented. In response to the NRC staff's RAI, DOE (2009ad) provided additional justification for the probability value of a camera not detecting improper interlocking between adjacent drip shields and a demonstration that mechanical or equipment reliability is not a significant component of the drip shield emplacement failure analysis. The NRC staff finds that the event sequence is appropriate and realistic because DOE acceptably described the processes and stated that installation will be accomplished under stringent controls. The NRC staff finds acceptable DOE's reliance on Postclosure Design Control Parameters 07-02 and 07-14 in SAR Table 1.9-9. The Postclosure Design Control Parameters provide that drip shield handling and emplacement be monitored by appropriate equipment, including an alarm, with an operator and independent inspector verifying proper installation. Therefore, the NRC staff concludes that the event sequence for this defect is acceptable because DOE identified the key processes for this event sequence and adequately described the processes and their controls.

Event Tree Models for Determining Early Failure Event Probabilities

The NRC staff has reviewed in detail the event trees DOE used to evaluate the probabilities that could lead to damage of the waste package and the drip shield. The NRC staff finds that DOE has identified the key processes involved and has implemented event sequences correctly. Therefore, the event trees are adequately constructed.

In SAR Sections 2.3.6.6.4.2 and 2.3.6.8.4.4.2, DOE compared its probability estimates for early failure of the waste package and drip shield, respectively, with the defect-related failure rates for the industrial analogues. For pressure vessels, nuclear fuel rods, underground storage tanks, and radioactive cesium capsules, the failure rates DOE cited are in the range of 10^{-6} to 10^{-4} per component (SAR Table 2.3.6-21), which is consistent with the calculated early failure rates for the waste package and drip shield. DOE did not identify any cases of SNF casks that failed after entering service.

The NRC staff finds that the waste package and drip shield are sufficiently similar to the industrial analogues to support a general comparison of the manufacturing- and handling-induced failure rates. In particular, the waste package, drip shield, and industrial analogues are (i) metallic, (ii) cut from sheet and formed into a cylindrical-type shape, (iii) welded, (iv) heat treated, and (v) closed/sealed (i.e., intended to act as a container or barrier).

The NRC staff finds that DOE has identified key processes involved with manufacturing and handling of the respective components and that it has developed realistic and appropriate event sequences to calculate the early failure probabilities. The NRC staff further finds that DOE's model support for estimating waste package and drip shield early failure probabilities is acceptable. Predicted early failure rates for the waste package and drip shield are close to those of the industrial analogues, as discussed previously.

Early Failure Event Probability Model Parameters

DOE developed event sequences to calculate the probabilities for undetected errors or defects (i.e., the early failure probabilities) in the waste package and drip shield, respectively. The event sequences generally consist of an equipment or process failure (e.g., probability that a motorized valve fails to open on demand), followed by human error(s), where the equipment or process failure is undetected (e.g., probability that the responsible technician does not detect the failure of the valve to open) or uncorrected. As described in SAR Sections 2.3.6.8.4.2 and 2.3.6.6.2, each event in the event sequences was assigned a probability distribution that was obtained from external data sources, as identified in SNL (2007aa, Section 4.1).

The external data used to establish the probability distributions for key processes and events in waste package and drip shield manufacturing and handling come from nuclear power plant activities. For the equipment or process failure events, DOE used reliability data that were generated from similar components or activities at nuclear power plants (e.g., Blanton and Eide, 1993aa). For human error events, the probability distributions for these events were taken from nuclear power plant human reliability analyses (Swain and Guttmann, 1983aa; Benhardt, et al., 1994aa).

The NRC staff reviewed the external data to determine whether it is reasonable and appropriate to use such data to quantify the reliability of events and processes associated with manufacturing and handling of the waste package and drip shield. The external data come from reliable, reputable sources that are widely accepted in the nuclear industry, as identified in SNL (2007aa, Section 4.1). Further, the NRC staff noted that SAR Section 1.9.2 specified rigorous controls on the manufacturing and handling of the waste package and drip shield, including use of nuclear industry standards and codes (e.g., American Society of Mechanical Engineers Boiler and Pressure Vessel Code). On the basis of the reasonable and appropriate nature of the data DOE used and DOE's adherence to industry codes and standards, the NRC staff finds that DOE has appropriately used data from nuclear power plants to establish the probability distributions for key processes and events in waste package and drip shield manufacturing and handling. Therefore, the NRC staff finds that the probability model parameters have been adequately established.

Uncertainty in Early Failure Event Probability Models and Parameters

DOE represented each basic event in an event sequence that can lead to an undetected defect (i.e., early failure) by a lognormal distribution. For the human error events, the external human reliability data DOE cited specify lognormal distributions with particular mean values and error factors (presented in SAR Table 2.3.6-22). For equipment or process failure events, the reliability data DOE cited typically specify only point (mean) values. As a result, DOE assigned an error factor to the probability data given in the literature as point (mean) values, as identified in SNL (2007aa, Section 5.3). DOE assumed that this point (mean) value is the mean of an unspecified probability distribution and that it is, therefore, appropriate to characterize the reliability with any reasonable, probability distribution. DOE used the lognormal distribution to be consistent with the human reliability data.

DOE used Monte Carlo simulations to calculate the probability distributions for the end states of the event sequences that could lead to early failure. Because the probability distributions for the basic events in the sequences may have different error factors, DOE stated that the mean value of the probability distribution for the end state of the sequence is not just a simple product of the mean of each basic event in the sequence, as identified in SNL (2007aa, Section 6.5.1). As described in DOE (2009ac), the probability distributions for all of the event sequences that could lead to an undetected defect in the waste package were combined to give the overall probability that the waste package has at least one undetected defect, which is assumed to be equivalent to the waste package early failure probability. The same was done for the drip shield. DOE ran thousands of realizations to obtain the probability distributions for early failure of the waste package and drip shield, as identified in SNL (2007aa, Section 6.5.1).

The NRC staff reviewed the treatment of uncertainty in the early failure probability calculations. The NRC staff finds that DOE has established appropriate, reasonable uncertainty distributions for the events in the event sequences that can lead to undetected defects (i.e., early failure). The lognormal distributions used for human reliability events are consistent with common practice (Swain and Guttmann, 1983aa). The NRC staff finds that, for those events given in the literature as a mean failure rate, DOE's assumption of uncertainty range that is consistent with human reliability events does not overestimate the reliability of components and processes. Further, the NRC staff finds that DOE has adequately propagated uncertainty through the early failure probability calculations for the waste package and drip shield. Use of Monte Carlo simulation is appropriate to ensure that the output is unbiased. DOE ran a sufficient number of realizations with Monte Carlo sampling to support its probability estimates. In summary, the NRC staff finds that uncertainty in event probability has been properly evaluated because DOE used reasonable uncertainty distributions; the assumptions that DOE used do not overestimate the reliability of components and processes; and DOE adequately propagated uncertainty.

The NRC staff notes that the probability distributions and values DOE provided for the probabilities of waste package and drip shield early failure are lognormal distributions. There is a mean of 1.13×10^{-4} failures per waste package and an error factor of 8.17 (SAR Section 2.3.6.6.3.2.7). There is a mean of 2.21×10^{-6} failures per drip shield and an error factor of 14 (SAR Section 2.3.6.8.4.3.2.5). These distributions and values are acceptable for use in DOE's Yucca Mountain repository performance assessment.

2.2.1.2.2.4 Evaluation Findings

The NRC staff reviewed the SAR and other information submitted to support the license application, which includes information required by 10 CFR 63.21(c)(1) and (9), and finds, with reasonable expectation, that the relevant requirements of 10 CFR 63.114 and 10 CFR 63.342 are satisfied.

For igneous events, NRC staff finds that the applicant identified and characterized specific FEPs of the geologic setting related to intrusive and extrusive (volcanic) igneous events.

To support the expert elicitation process, the applicant investigated and characterized known igneous features for the entire Yucca Mountain region and provided the location, age, and basic characteristics necessary to support probability estimates in the probabilistic volcanic hazard assessment evaluation. The NRC staff finds that DOE's specification of separate igneous event probabilities for intrusive and extrusive (volcanic) events is consistent with the guidance in the YMRP. The NRC staff finds that the license application (i) included geologic data and used that data to adequately define the igneous event and adequately establish probability model parameters; (ii) accounted for, and adequately evaluated, uncertainties in the igneous event probability model; (iii) provided appropriate technical bases supporting the models used and the estimated probability; and (iv) included an igneous activity analysis that was consistent with the limits on performance assessment specified at 10 CFR 63.342.

For seismic events, the NRC staff finds that the applicant identified and characterized specific FEPs of the geologic setting related to seismic fault displacement and ground motion events. Through the expert elicitation process, the applicant developed distributions of locations and activity rates for fault and background seismic sources for the entire Yucca Mountain region to assess the probabilistic seismic and fault displacement hazard. The NRC staff finds that the license application (i) included geologic data and used that data to adequately define the faulting and seismic events and adequately establish probability model parameters; (ii) accounted for, and adequately evaluated, uncertainties in the faulting and seismic event probability models; (iii) provided appropriate technical bases supporting the models used and the estimated probabilities; and (iv) included a seismic activity analysis that was consistent with the limits on performance assessment specified at 10 CFR 63.342.

For early waste package and drip shield failure events, the NRC staff finds that the license application (i) included information on the design of the engineered barrier system to adequately define the waste package and drip shield early failure events and adequately establish probability model parameters; (ii) accounted for, and adequately evaluated, uncertainties in waste package and drip shield early failure analyses; (iii) provided appropriate technical bases supporting the analyses used and the estimated probabilities; and (iv) included an event sequence analysis for early waste package and drip shield failures that was consistent with the limits on performance assessment specified at 10 CFR 63.342.

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CHAPTER 4

2.2.1.3.1 Degradation of Engineered Barriers

2.2.1.3.1.1 Introduction

Safety Evaluation Report (SER) Section 2.2.1.3.1 addresses the chemical degradation of the drip shield and waste packages stored in the repository drifts. The drip shield and the waste packages are engineered barriers, a subset of the Engineered Barrier System. The general functions of the Engineered Barrier System are to (i) prevent or significantly reduce the amount of water that contacts the waste, (ii) prevent or significantly reduce the rate at which radionuclides are released from the waste, and (iii) prevent or significantly reduce the rate at which radionuclides are released from the engineered barrier system to the Lower Natural Barrier [Safety Analysis Report (SAR) Section 2.1.1.2 (DOE, 2008ab)]. The engineered barrier system consists of the emplacement drift, the drip shield, the waste package, the naval spent nuclear fuel structure, the waste form and waste package internals (e.g., transportation, aging, and disposal canisters), the waste package pallet, and invert features (SAR Figure 2.1-7).

In the postclosure performance assessment, the U.S. Department of Energy ("DOE" or "applicant") evaluated whether the ability of the engineered barrier system components to perform their barrier functions could be compromised by features, events, and processes (FEPs) that degrade their physical structure. In particular, DOE considered that the engineered barrier system components were subject to mechanical degradation caused by seismic ground motion (SAR Section 2.3.4). The U.S. Nuclear Regulatory Commission (NRC) staff's review of DOE's Total System Performance Assessment (TSPA) models for mechanical degradation of the engineered barrier system is found in SER Section 2.2.1.3.2. DOE also considered chemical degradation, or corrosion, caused by reactions between the engineered barrier system materials and the environment in its postclosure performance assessment of engineered barrier system degradation. In SAR Section 2.3.6, DOE described the TSPA model abstractions for chemical degradation of the drip shield and the waste package outer barrier. SER Section 2.2.1.3.1 reviews DOE's TSPA model abstractions for chemical degradation of the drip shield and the waste package outer barrier.

2.2.1.3.1.2 Regulatory Requirements

The applicant is required to provide information specified in 10 CFR 63.21(c)(3),(9),(10), and (15) that relate to the degradation of engineered barriers. The requirements in 10 CFR 63.114 (Requirements for Performance Assessments) and 63.342 (Limits on Performance Assessments) include postclosure performance assessment to demonstrate compliance with 10 CFR 63.113 (Performance Objectives for the Geologic Repository after Permanent Closure). Compliance with 10 CFR 63.113 is reviewed in SER Section 2.2.1.4.1.

The regulations for performance assessment in 10 CFR 63.114 require, in part, that a performance assessment

- Include data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the surface and subsurface from the site and the region surrounding Yucca Mountain [10 CFR 63.114(a)(1)]

- Account for uncertainty and variability in the parameter values [10 CFR 63.114(a)(2)]

- Consider and evaluate alternative conceptual models [10 CFR 63.114(a)(3)]

- Provide technical bases for either the inclusion or exclusion of FEPs, including effects of degradation, deterioration, or alteration processes of engineered barriers that would adversely affect performance of the natural barriers, consistent with the limits on performance assessment in 10 CFR 63.342, and evaluate in sufficient detail those processes that would significantly affect repository performance [10 CFR 63.114(a)(4-6)]

- Provide technical basis for the models used in the performance assessment to represent the 10,000 years after disposal [10 CFR 63.114(a)(7)]

The NRC staff's evaluation of inclusion or exclusion of FEPs is given in SER Section 2.2.1.2.1. Requirements for performance assessment for the initial 10,000 years following disposal are specified in 10 CFR 63.114(a). Requirements for the performance assessment methods for the time from 10,000 years through the period of geologic stability, defined in 10 CFR 63.302 as 1 million years following disposal are specified in 10 CFR 63.114(b) and 63.342. These sections require that through the period of geologic stability, with specific limitations, the applicant

- Use performance assessment methods consistent with the performance assessment methods used to demonstrate compliance for the initial 10,000 years following permanent closure[10 CFR 63.114(b)]

- Include in the performance assessment those FEPs used in the performance assessment for the initial 10,000-year period (10 CFR 63.342)

For the period beyond 10,000 years following permanent closure, the applicant has chosen to assess the effects of general corrosion on engineered barriers in its performance assessment by using a distribution of corrosion rates correlated to other repository parameters [10 CFR 63.342(c)(3)].

The NRC staff's review of the SAR and supporting information follows the guidance in Yucca Mountain Review Plan (YMRP) (NRC, 2003aa), Sections 2.2.1.3.1, Degradation of Engineered Barriers, as supplemented by additional guidance for the period beyond 10,000 years after permanent closure (NRC, 2009ab). The YMRP acceptance criteria for model abstractions that provide guidance for the NRC staff's review of the applicant's abstraction of degradation of engineered barriers are

1. System description and model integration are adequate
2. Data are sufficient for model justification
3. Data uncertainty is characterized and propagated through the abstraction
4. Model uncertainty is characterized and propagated through the abstraction
5. Model abstraction output is supported by objective comparisons

In its review of SAR and supporting information, the NRC staff used a risk-informed approach and the guidance in the YMRP, as supplemented by NRC (2009ab), for aspects of degradation of engineered barriers important to repository performance. The NRC staff considered all five YMRP criteria in its review of information provided by DOE. In the context of these criteria, only those aspects of the model abstraction that substantively affect the performance assessment results, as determined by the NRC staff, are discussed in detail in this SER section. The NRC staff's determination is based both on risk information provided by DOE, and on the NRC staff's knowledge gained through experience and independent analyses.

2.2.1.3.1.3 Technical Review

DOE's models for chemical degradation of the engineered barrier systems focus on both the drip shield and the waste package outer barrier. Consistent with the YMRP guidance, the NRC staff performed a risk-informed, performance-based review, focusing on those aspects of DOE's models for chemical degradation of the drip shield and the waste packages that are most important to the assessment of barrier capability. DOE concluded that seepage flux is the primary source of water that may react with the engineered barrier system components (SAR Section 2.3.7.12.1). In DOE's model for flow of seepage water through the engineered barrier system, the water must first pass through the drip shield and then through the waste package before contacting and mobilizing the waste form. As such, SER Section 2.2.1.3.1 first concentrates on DOE's models for chemical degradation of the drip shield and then addresses DOE's models for chemical degradation of the waste package.

2.2.1.3.1.3.1 Drip Shield Degradation

The drip shield, which DOE described in SAR Section 1.3.4.7, is an engineered metal barrier designed to divert water that enters the drift and prevent it from contacting the waste package. DOE stated that the drip shield will be fabricated from Titanium Grade 7 (UNS R52400). Titanium Grade 7 is a commercially pure titanium alloy with the addition of a small amount of palladium (approximately 0.2 weight percent) to enhance its corrosion resistance. The drip shield structural supports will be fabricated from Titanium Grade 29 (UNS R56404), which is a titanium alloy composed of approximately 6 weight percent aluminum and 4 weight percent vanadium for strength, plus approximately 0.1 weight percent ruthenium for corrosion resistance.

In developing the postclosure performance assessment analysis, DOE evaluated a number of FEPs (in SAR Table 2.2-5) related to chemical degradation of the drip shield, including

- General corrosion of the drip shields (FEP 2.1.03.01.0B)
- Stress corrosion cracking of the drip shields (FEP 2.1.03.02.0B)
- Localized corrosion of the drip shields (FEP 2.1.03.03.0B)
- Hydride cracking of the drip shields (FEP 2.1.03.04.0B)
- Microbially influenced corrosion (MIC) of the drip shields (FEP 2.1.03.05.0B)
- Early failure of the drip shields (FEP 2.1.03.08.0B)
- Oxygen embrittlement of the drip shields (FEP 2.1.06.06.0B)
- Creep of metallic materials in the drip shield (FEP 2.1.07.05.0B)
- Localized corrosion on drip shield surfaces due to deliquescence (FEP 2.1.09.28.0B)
- Thermal sensitization of the drip shields (FEP 2.1.11.06.0B)

With the exception of general corrosion and early failure of drip shields, these features, events, and processes were screened out from the performance assessment on the basis of low consequence or low probability (SAR Table 2.2-5). The NRC staff's evaluation of DOE's bases for excluding these FEPs from the performance assessment is addressed in SER Section 2.2.1.2.1.

With respect to the FEPs included in the performance assessment, DOE described general corrosion of the drip shield as the uniform thinning of both the Titanium Grade 7 drip shield plates and the Titanium Grade 29 structural supports (SAR Section 2.3.6.8.1.1). In SAR Section 2.2.2.3, DOE defined drip shield early failure as through-wall penetration caused by manufacturing- and handling-induced defects at a time earlier than would be expected for a nondefective drip shield.

In the TSPA analysis, DOE calculated that conditions in the drift (e.g., temperature, pH, seepage water chemistry) may support localized corrosion of the waste package if the drip shield fails and allows seepage water to contact the waste package within approximately 12,000 years after repository closure, as detailed in DOE (2009dg, Enclosure 1). Following 12,000 years after repository closure, DOE calculated that there is a low probability for aqueous chemical conditions in the drift to support localized corrosion of the waste package even if the drip shield fails and allows seepage water to contact the waste package, as shown in DOE (2009dg, Enclosure 1, Figure 1). The TSPA analysis calculates that few drip shields will fail within 12,000 years after repository closure. Therefore, the probability of waste package breach by localized corrosion is low in DOE's model.

Other than for localized corrosion, the integrity of the drip shield does not have a significant effect on DOE's model abstractions for chemical degradation of the waste package. In the TSPA Nominal Modeling Case, DOE's models for general corrosion and stress corrosion cracking of the waste package, DOE assumed aqueous degradation conditions in these models, even for the intact drip shield. In the Seismic Ground Motion Modeling Case in the TSPA analysis, the presence of the drip shield does have some effect on stress corrosion cracking of the waste package because DOE calculated that the waste package under an intact drip shield will have a greater likelihood of being damaged under low-probability seismic ground motion events than it would under the assumption of a failed drip shield condition. This is because an intact drip shield permits unobstructed free movement of the waste package, thereby potentially causing damage as waste packages strike one another (SAR Section 2.3.4.5). Under a collapsed drip shield event, waste packages are constrained from significant movement and unable to strike or bump into each other. Consequently, DOE concluded that a waste package under an intact drip shield is more susceptible to stress corrosion cracking if a low probability seismic event occurs that imparts the required energy for the waste packages to strike each other. Nevertheless, DOE calculated that the probability of a seismic ground motion with sufficient magnitude to damage the waste package is so low, even in the Seismic Ground Motion Modeling Case, that the presence of the drip shield has an insignificant effect on the postclosure performance assessment beyond 12,000 years after repository closure, as described in DOE (2009cn, Enclosure 5).

The NRC staff's reviews of DOE's model abstractions for general corrosion and early failure of the drip shield are presented in the following sections. Because the presence of the drip shield is important for DOE's calculations that localized corrosion of the waste package is unlikely within 12,000 years after repository closure, the NRC staff focused on those aspects of the models that were most important to DOE's calculations of the drip shield lifetime.

2.2.1.3.1.3.1.1 Drip Shield General Corrosion

In SAR Section 2.3.6.8.1, DOE described the model for general corrosion of the drip shield that was implemented in the TSPA. The drip shield is constructed of titanium alloys that are assumed to be highly corrosion resistant because of their passivity. Passivity refers to a state in which metals and alloys are not chemically reactive under certain environmental conditions. The passive state is generally attributed to the presence of a thin, protective oxide film on the metal surface. Because the maintenance of the passive state is important to the corrosion performance of the drip shield, the NRC staff first reviewed the drip shield's long-term passive film stability in the repository conditions. The NRC staff then reviewed the model abstraction used to calculate the drip shield general corrosion rate in the TSPA model.

Drip Shield's Long-Term Passive Film Stability

In BSC (2004as, Section 1.1), DOE presented literature references (Pourbaix, 1974aa; Schutz and Thomas, 1987aa) that indicated the passive films on titanium alloys are stable over wide ranges of chemical potential and pH, and that, should the passive film rupture, titanium has a strong tendency for repassivation in the type of oxidizing conditions that are expected in the repository. DOE, however, also cited literature references (e.g., Lorenzo de Mele and Cortizo, 2000aa; Brossia, et al., 2001aa; Brossia and Cragnolino, 2000aa, 2001ab, 2004aa; Pulvirenti, et al., 2002aa, 2003aa) that indicated dissolved fluoride in brine solutions can increase the general corrosion rates for titanium alloys and possibly compromise the stability of the passive film. Therefore, DOE evaluated the uncertainty in long-term drip shield passive film persistence associated with possible passive film degradation by fluoride-bearing seepage water brines, as described in BSC (2004as, Section 6.5.7).

In BSC (2004as, Section 6.5.7.2), DOE reviewed and analyzed passive film instability. The applicant cited literature references that described the onset of localized corrosion on titanium specimens that were exposed to fluoride shortly after the passive film was manually removed by polishing (e.g., Brossia and Cragnolino, 2000aa, 2001ab; Brossia, et al., 2001aa). When the specimens were in an oxidizing environment for as little as 4 days prior to fluoride exposure, the specimens exhibited resistance to fluoride attack (Lorenzo de Mele and Cortizo, 2000aa). DOE stated that it expects the drip shield to have an extended period of dry thermal oxidation between the time the repository is closed and the time at which seepage water may fall onto the drip shield, as described in BSC (2004as, Section 6.5.7). Even for thermally oxidized Titanium Grade 7 specimens, however, passive film instability in a fluoride-rich solution with low pH (~4) has been observed (Lian, et al., 2005aa). However, DOE concluded that such conditions are not representative of the environment expected in the repository, as outlined in SNL (2008ac, p. 6-408). DOE stated that even if seepage water brines in the repository contain fluoride, high concentrations of other species will also be present that will suppress or neutralize any fluoride attack. DOE identified studies of alloys with similar composition to Titanium Grade 7 in environments with temperatures up to 177 °C [351 °F]; pH as low as 1; and fluoride, along with other species such as calcium, magnesium chloride, and silicate (Thomas and Bomberger, 1983aa; Schutz and Grauman, 1986aa). The studies showed that the titanium had a high passive film persistence, which was attributed to calcium reducing the fluoride ion solubility by precipitation of calcium fluoride, as well as the displacement of fluoride from absorption on the passive film by other species. Moreover, DOE presented its own test results in BSC (2004as, Section 6.5) in which Titanium Grade 7 specimens showed no evidence of passive film instability after 5 years of exposure to simulated concentrated water, which contained fluoride, as well as chloride, silica, sulfate, nitrate, and bicarbonate (composition given in SAR Table 2.3.6-1). Therefore, DOE concluded that the drip shield passive film will be stable during the postclosure period given the expected composition of seepage water brines, as described in SNL (2008ac, pp. 6-410).

NRC Staff's Review

The NRC staff reviewed DOE's assessment of drip shield passivity. On the basis of its review of the information DOE provided in BSC (2004as, Section 6.5.7), the NRC staff finds that uncertainty in the long-term persistence of the titanium passive film is primarily related to potential passive film degradation by fluoride-bearing brines. The NRC staff concludes that there is no evidence of localized corrosion of Titanium Grade 7 exposed to fluoride-bearing simulated concentrated water for 5 years and thus that the passive film is stable when in contact with a brine having this composition. The NRC staff finds that, based on NRC staff's independent tests, the extent of titanium passive film degradation will generally decrease with decreasing fluoride concentration in the brines (Brossia, et al., 2001aa). Analyses by the NRC staff (Pabalan, 2010aa) indicate that the fluoride concentration in simulated concentrated water is greater than would be expected in water that would contact the drip shield in repository conditions because other species in the waters, such as calcium, can precipitate fluoride ions out of solution, thus limiting the free-chloride concentration. Therefore, the NRC staff finds that DOE's 5-year testing results of the Titanium Grade 7 specimens in fluoride-bearing simulated concentrated water is consistent with DOE's assumption of passive film stability in repository conditions. The NRC staff also concludes that literature references DOE cited (Thomas and Bomberger, 1983aa; Schutz and Grauman, 1986aa) further show that other species in the waters, such as calcium, can protect the passive film by causing fluoride ions to precipitate out of solution. Finally, independent analyses conducted by the NRC staff (Lin, et al., 2003aa; Codell and Leslie, 2006aa) also support that seepage water brines that may contact the drip shield will have insufficient fluoride concentration to significantly affect passive film stability on the drip shield titanium alloys. On the basis of this information, the NRC staff finds DOE's assumption that the drip shield is protected by a passive oxide film during the postclosure period acceptable.

Drip Shield General Corrosion Conceptual Model

In SAR Section 2.3.6.8.1, DOE described the conceptual model for general corrosion of the drip shield that was implemented in the TSPA. In DOE's model, corrosion begins at the time of repository closure and progresses at a constant rate over time. DOE assumed aqueous conditions in the drift and also that the general corrosion rate is independent of in-drift environmental conditions (e.g., temperature, relative humidity).

The NRC staff requested DOE's technical basis for assuming that the general corrosion rate of the drip shield is independent of temperature. In DOE (2009cn, Enclosure 3), DOE stated that at the start of the general corrosion process the corrosion rates of titanium alloys are temperature dependent. However, over time, the corrosion rates at different temperatures tend to converge. DOE showed a noticeable trend of increasing corrosion rate with increasing temperature for Titanium Grade 7 specimens tested in the range of 50 to 110 °C [122 to 230 °F] after 4 weeks exposure, but DOE also showed that the corrosion rate was less temperature dependent after 8 weeks (Hua and Gordon, 2004aa). Further, DOE referenced 3-year corrosion tests of titanium plus 0.2 weight percent palladium, which has nearly the same composition as Titanium Grade 7 in the temperature range of 90 to 200 °C [194 to 392 °F] in a pH 4.9 chloride-sulfate brine (Smailos and Köster, 1987aa). DOE concluded that the corrosion rates initially showed some temperature dependence, but were effectively identical within 3 years as shown in Smailos and Köster (1987aa, Figure 1).

NRC Staff's Review

The NRC staff reviewed DOE's conceptual model for general corrosion of the drip shield. The NRC staff finds DOE's assumed aqueous conditions acceptable because titanium general corrosion proceeds more rapidly in aqueous conditions than in dry conditions. Further, the NRC staff concludes that data from DOE testing (SAR Figure 2.3.6-49) and independent corrosion data DOE referenced for material similar to the drip shield titanium alloys (Smailos and Köster, 1987aa) indicate that the general corrosion rates decrease for titanium alloys over time. The technical literature indicates that the decreasing corrosion rate may correspond to thickening of the passive oxide film (Jones, 1996aa). Thus, the NRC staff finds DOE's use of a constant corrosion rate over time acceptable because this assumption will not underestimate the corrosion rate.

In addition, the NRC staff reviewed DOE's assumption that the general corrosion rate is independent of temperature. The NRC staff concludes that the studies DOE cited (e.g., Smailos and Köster, 1987aa; Hua and Gordon, 2004aa) considered materials analogous to the drip shield titanium alloys and environmental conditions that are similar to or more aggressive than those expected in the repository. On the basis of information provided in these studies, the NRC staff concludes that, although the corrosion rates of titanium alloys may have some temperature dependence during an initial period of exposure to corrosive brines, there is little temperature dependence after this period. The NRC staff finds that the information DOE provided is also consistent with independent analyses by NRC (Mintz and He, 2009aa), which confirmed that corrosion rates for titanium alloys do not show significant temperature dependence for temperatures that are representative of the repository conditions. Therefore, the NRC staff finds DOE's assumption that the corrosion rates of the drip shield titanium alloys are independent of temperature acceptable because this assumption will not underestimate the corrosion rate.

Long-Term Corrosion Test Data

The corrosion rates for Titanium Grades 7 and 29 that were sampled in the TSPA were based on data from weight-loss corrosion tests at the Long-Term Corrosion Test Facility (SAR Section 2.3.6.8.1.2.1). The following summarizes the NRC staff's review of DOE's data used in the TSPA analysis.

Titanium Grade 7

The corrosion rate for Titanium Grade 7 that was sampled in the TSPA was based on 2.5-year tests of Titanium Grade 7 crevice and weight-loss specimens with wrought (base metal-type) and as-welded metallurgical conditions (SAR Section 2.3.6.8.1.2.1). Some specimens were fully immersed in solution (i.e., aqueous phase), whereas others were in the saturated vapor above the aqueous phase. DOE exposed the test specimens to different solutions, including simulated acidified water, simulated dilute water, and simulated concentrated water, the compositions of which are given in SAR Table 2.3.6-1. The tests were performed at temperatures of 60 and 90 °C [140 and 194 °F]. DOE measured the material weight loss during the test period and used these data to calculate the general corrosion rates, following American Society of Testing and Materials (ASTM) G1-90 (ASTM International, 1999aa). DOE observed lower corrosion rates of crevice specimens than those of weight-loss specimens (SAR Figure 2.3.6-44). Therefore, DOE chose to use only the data from the weight-loss specimens in the model abstraction because it will calculate a higher corrosion rate in the TSPA analysis. For the weight-loss specimens, DOE did not observe a significant difference in corrosion rates between wrought and as-welded materials, but did observe that the corrosion rates depended upon the chemistry of the test solution. In particular, the corrosion rates for specimens tested in the simulated concentrated water aqueous phase were as high as 50 nm/yr [1.97×10^{-6} in/yr], whereas the corrosion rates for the specimens tested in the aqueous and vapor phases of simulated acidified water and simulated dilute water, as well as for specimens tested in the simulated concentrated water vapor phase, were below 20 nm/yr [7.87×10^{-7} in/yr] as shown in BSC (2004as, Figures 6.6[a] and 6.7[a]).

In the TSPA analysis, DOE assumed that corrosion occurs simultaneously on the inner surface and the outer surface of the Titanium Grade 7 drip shield plates, with different corrosion rates for the respective surfaces. DOE assumed that the outer surface of the plate corroded faster than the inner surface because the outer surface is expected to be exposed to a more aggressive environment, including dust and dripping seepage water, as detailed in BSC (2004as, Section 6.1.6[a]). DOE used the data from the most aggressive test condition, obtained from the simulated concentrated water aqueous phase, to derive the distribution from which the outer surface corrosion rate was sampled in the TSPA model. In aqueous simulated concentrated water, DOE measured higher corrosion rates for Titanium Grade 7 at 90 °C [194 °F] than at 60 °C [140 °F] as shown in BSC (2004as, Figure 6.6[a]). DOE did not, however, consider temperature dependence for the titanium general corrosion rate. Instead, DOE elected to use only the data from the 90 °C [194 °F] tests because these gave a higher corrosion rate. These data (“Aggressive Condition” in SAR Figure 2.3.6-46) have a mean corrosion rate of 46.1 nm/yr [1.81×10^{-6} in/yr]. For the general corrosion rate on the underside of the drip shield plates, DOE used the data from specimens tested at 60 and 90 °C [140 and 194 °F] in the aqueous and vapor phases of the simulated acidified water and the simulated dilute water, respectively, as well as specimens tested at 60 and 90 °C [140 and 194 °F] in the simulated concentrated water vapor phase, as detailed in BSC (2004as, Section 6.1.7[a]). These data (“Benign Condition” in SAR Figure 2.3.6-46) have a mean corrosion rate of 5.1 nm/yr [2.01×10^{-7} in/yr].

DOE considered uncertainty in the measured corrosion rates, which is attributed to difficulties in cleaning and weighing corrosion specimens, particularly given the very small weight losses associated with low corrosion rates, as well as randomness in the general corrosion processes, as described in BSC (2004as, Section 6.1.6.1[a]). DOE determined that the corrosion rate for the outside of the drip shield plates is best represented by a normal distribution, the mean of which is sampled from a *t*-distribution, described in SNL (2008ag, Table 6.3.5-3). The *t*-distribution is a broader normal distribution DOE used given that this set of corrosion rate data only has six data points. The mean of the *t*-distribution is approximately 46.1 nm/yr [1.81×10^{-6} in/yr], with 2.5th and 97.5th percentile values of approximately 43.0 and 49.1 nm/yr [1.69×10^{-6} in/yr and 1.93×10^{-6} in/yr], respectively, as detailed in BSC (2004as, Section 6.1.6.2[a]). The variability of distributions for the general corrosion rate on the outside of the drip shield plates were shown in BSC (2004as, Figure 6-11[a]). For the inside of the drip shield plates, DOE determined that the general corrosion rate is best represented by a gamma distribution, the mean of which is sampled from a normal distribution, described in SNL (2008ag, Table 6.3.5-3). The gamma distribution is a continuous skewed distribution function used to describe the distribution of variables that are positive and unbound. The mean of the normal distribution is approximately 5.1 nm/yr [2.01×10^{-7} in/yr], with 2.5th and 97.5th percentile values of approximately 3.5 nm/yr and 6.8 nm/yr [1.38×10^{-7} in/yr and 2.68×10^{-7} in/yr], respectively, as outlined in BSC (2004as, Section 6.1.7.2[a]). The variability distributions for the general corrosion rate on the inside of the drip shield plates were shown in BSC (2004as, Figure 6-19[a]).

DOE compared the corrosion rate sampled in the TSPA code to independently reported corrosion rates for analogous alloys in environments similar to or more aggressive than those expected in Yucca Mountain, as detailed in SAR Section 2.3.6.8.1.5 and BSC (2004as, Section 7.2.1[a]). DOE concluded that the TSPA-calculated corrosion rates are consistent with corrosion rates measured by Smailos and Köster (1987aa) for titanium plus 0.2 weight percent palladium in the temperature range of 90 to 200 °C [194 to 392 °F] in a pH 4.9 chloride-sulfate brine.

In response to the NRC staff's request for additional information (RAI) on how the experimental uncertainties associated with sample cleaning, weighing, and measuring were incorporated into the sampled corrosion rate distributions, DOE (2009cn) stated that subsequent examination of corrosion test specimens revealed that posttest specimen cleaning did not adequately remove a residual oxide film. This resulted in under-measurements of specimen weight loss and, in turn, an underestimation of the general corrosion rates for the inside and outside of the drip shield plates. To assess the effect of the incomplete specimen-cleaning procedure on corrosion rate uncertainties, DOE conducted cross section analyses of the chemically cleaned posttest specimens. DOE estimated that the general corrosion rates for Titanium Grade 7, presented in SAR Section 2.3.6.8.1, were underestimated by, at most, a factor of two. Consequently, DOE conducted a sensitivity analysis in which it considered corrosion rates up to four times those given in SAR Section 2.3.6.8.1. This shortened the drip shield framework and plate lifetime compared to those calculated in the TSPA model. DOE stated that this sensitivity analysis showed that corrosion rates of up to four times higher than those given in SAR Section 2.3.6.8.1 resulted in negligible differences in the expected dose curves, as shown in DOE (2009cn, Enclosure 5, Figure 2). Therefore, DOE concluded that the data presented in SAR Section 2.3.6.8.1 were acceptable to use in the TSPA model because unquantified experimental uncertainties had negligible impact on the postclosure performance assessment.

In DOE (2009cn, Enclosure 4), the applicant responded to the NRC staff's RAI on the adequacy of immersion test conditions in simulated brines to determine general corrosion rates used to model the corrosion behavior of the drip shield, considering that some passive alloys may be more susceptible to corrosion in dripping conditions than in immersion conditions (e.g., Lee and Solomon, 2006aa). The applicant stated that the temperatures at which dripping effects on corrosion behavior have been observed in other passive alloys are greater than the temperatures expected for the drip shield in dripping conditions. Moreover, DOE stated that data in the technical literature indicate that titanium alloys are highly resistant to dripping effects because of their tenacious passive film (Schutz, 2005aa). Therefore, DOE concluded that the immersion tests in the simulated brines were adequate to model the corrosion behavior of the drip shield in the repository because the simulated brines accounted for potential dripping conditions.

NRC Staff's Review

The NRC staff reviewed the information DOE provided in SAR Section 2.3.6.8.1 and DOE (2009cn) and arrived at the following findings:

- With regard to the material conditions, the NRC staff finds DOE's testing of Titanium Grade 7 specimens with wrought and as-welded microstructures acceptable. DOE's tests on the material microstructures are acceptable based on the fabrication procedures set forth in BSC (2007bu) because they are representative of the microstructures expected for Titanium Grade 7 drip shield plates in the base metal and in the region of the weld.

- With respect to the corrosion test solutions, the NRC staff finds that DOE tested the Titanium Grade 7 specimens in a range of corrosion test solutions, including simulated acidified water, simulated dilute water, and simulated concentrated water. The NRC staff concludes that the corrosion rate for the drip shield in the repository may depend on such factors as the pH and concentration of ionic species in water that contacts the drip shield. Therefore, the NRC staff reviewed the corrosion test solutions to determine whether they are adequate to model the drip shield corrosion rate for the conditions expected in the repository. The NRC staff finds that DOE's corrosion test solutions are more chemically aggressive than waters expected to occur within the repository drifts, including starting water compositions in DOE's near-field chemistry model described in SAR Section 2.3.5.5, and waters considered in NRC staff's independent analysis of in-drift water evolution, described in SER Section 2.2.1.3.3.3.2. Therefore, the NRC staff finds acceptable DOE's model of the drip shield corrosion rate based on tests in simulated acidified water, simulated concentrated water, and simulated dilute water.

- With regard to the testing conditions, DOE used immersion corrosion tests to represent corrosion behavior, including potential dripping conditions in the drift. The NRC staff finds that literature DOE cited (Schutz, 2005aa) and other independent literature (He, et al., 2007aa) show that titanium alloys are highly resistant to dripping-induced corrosion effects at temperatures expected in the repository because of their strong tendency to passivity. NRC staff's confirmatory tests showed, under dripping of seepage water, general corrosion rates of titanium Grades 7 and 29 are less than the mean general corrosion rate that DOE obtained from immersion conditions (He, 2011aa; Jung, et al., 2011aa). Therefore, the NRC staff finds that DOE acceptably performed corrosion tests in immersion because this will not underestimate the corrosion rate in dripping conditions.

In addition, the NRC staff evaluated DOE's experimental procedures for cleaning, weighing, and measuring the corrosion rates of the test specimens. The NRC staff finds that DOE adequately identified deficiencies with its specimen preparation and cleaning that initially led to unquantified experimental uncertainties in the general corrosion rate for Titanium Grade 7. reported in SAR Section 2.3.6.8.1, by demonstrating that the actual corrosion rates for the Titanium Grade 7 drip shield plates would not exceed a factor of two to four times the corrosion rates sampled in the TSPA analysis and that the higher corrosion rates resulted in negligible changes in calculated results. In the TSPA, the sum of the mean general corrosion rates for the inside and outside of the Titanium Grade 7 drip shield plates is approximately 51.2 nm/yr [2.06×10^{-6} in/yr]. Therefore, the NRC staff observed that increasing the corrosion rate by a factor of two to four would increase the mean corrosion rate to approximately 100 to 200 nm/yr [3.93×10^{-6} to 7.87×10^{-6} in/yr]. Hence, the NRC staff concludes that at this range of corrosion rates, breach of the 15 mm- [0.59 in]-thick Titanium Grade 7 plate by general corrosion would generally occur between 75,000 and 150,000 years after repository closure. This is well beyond 12,000 years after repository closure. Prior to 12,000 years after repository closure, DOE calculated that the waste package outer barrier may undergo localized corrosion if contacted by seepage water (SAR Section 2.3.6.4). As described in SER Section 2.2.1.3.1.3.1, DOE calculated that conditions in the drift (e.g., temperature, pH, seepage water chemistry) may support localized corrosion of the waste package if the drip shield fails and allows seepage water to contact the waste package within approximately 12,000 years after repository closure

(DOE, 2009dg). The probability that waste packages will undergo localized corrosion decreases with time. Therefore, the NRC staff finds acceptable DOE's demonstration that unquantified experimental uncertainties in the general corrosion rate for Titanium Grade 7, reported in SAR Section 2.3.6.8.1, will have a negligible effect on the postclosure performance assessment results.

Finally, the NRC staff compared the Titanium Grade 7 corrosion rates sampled in the TSPA code to independently reported corrosion rates for the same or similar material. The NRC staff finds that the corrosion rates sampled in the TSPA code are similar to those measured in independent testing that DOE cited (Smailos and Köster, 1987aa; Mattsson and Olefjord, 1990aa). Further, the NRC staff considered the report by Schutz (2005aa), which stated that the corrosion rates of titanium alloys are negligible in a variety of solutions, including seawater up to 260 °C [500 °F], 62 percent $CaCl_2$ at 150 °C [302 °F], boiling solutions of 10 percent and 30 percent $FeCl_3$, and boiling saturated $MgCl_2$ solution. Additionally, the NRC staff performed independent corrosion tests of Titanium Grade 7 in 1 M NaCl solution at 95 °C [203 °F] and measured a corrosion rate of approximately 87 nm/yr [3.43×10^{-6} in/yr] (Brossia, et al., 2001aa; Brossia and Cragnolino, 2004aa). The NRC staff determined that the titanium alloys tested in these studies are analogous to the drip shield titanium alloys and that the environmental conditions (e.g., temperature and brine chemistry) considered in these studies are chemically and thermally more aggressive than those expected for the postclosure period in the repository. The NRC staff finds that the corrosion rates measured in these studies are similar to the Titanium Grade 7 drip shield plate corrosion rates the TSPA code calculated. Therefore, the NRC staff finds that independent reports in the technical literature provide support for the corrosion rate DOE calculated in the TSPA code.

Titanium Grade 29

The corrosion rate for Titanium Grade 29 that was applied in the TSPA analysis was based on 42-day weight-loss measurements of Titanium Grades 7 and 29 specimens in solutions that DOE stated were representative of seepage water and deliquescent brines expected in the repository [SAR Section 2.3.6.8.1.3 and BSC (2004as, Section 6.2.1[a]). The compositions of the brines are given in BSC (2004as, Table 6-7[a]). For tests at 120 and 150 °C [248 and 302 °F], DOE calculated the ratios of the corrosion rates of Titanium Grade 29 to those of Titanium Grade 7. For a given test environment, DOE calculated that the corrosion rate of Titanium Grade 29 could be a factor of one to seven times higher than that of Titanium Grade 7 (SAR Figure 2.3.6-48). From these data, DOE developed a discrete probability distribution function, summarized in BSC (2004as, Table 6-8[a]), which gave the ratio for the corrosion rate of Titanium Grade 29 to that of Titanium Grade 7. To calculate the corrosion rate for the Titanium Grade 29 structural supports in the TSPA model, DOE sampled the ratio from this probability distribution function and multiplied the sampled ratio by the corrosion rate on the outside of the Titanium Grade 7 plate (i.e., under aggressive conditions).

In BSC (2004as, Section 6.2[a]), DOE acknowledged that it did not have long-term general corrosion data for Titanium Grade 29. DOE stated, however, that the passive films for both Titanium Grade 7 and Grade 29 are likely to be predominantly titanium oxide. DOE also stated that data show that the passive behavior for the respective alloys is the same for the range of brines expected in the repository (Andresen and Kim, 2006aa). Therefore, DOE concluded that the corrosion processes for Titanium Grades 7 and 29 are similar and that comparing the corrosion rates of the respective alloys in short-term tests is an adequate basis for calculating the long-term corrosion rate for Titanium Grade 29.

In response to the NRC staff's RAI that requested DOE to assess additional uncertainties associated with the comparative corrosion tests, DOE reanalyzed the comparative corrosion data (DOE, 2009cn). DOE determined that the weight loss for the respective alloys was measured by a weighing balance that had uncertainty larger than most of the measured weight-change values. DOE concluded that it was unable to make a meaningful distinction between actual material weight loss and measurement uncertainty. Further, DOE stated that for the same tests, corrosion rates were also measured by electrochemical impedance spectroscopy and linear polarization resistance (Andresen and Kim, 2006aa), with negligible difference for the respective alloys. On the basis of this reanalysis, DOE determined that, because there was no measurable difference between the corrosion rates for the respective alloys in the 42-day tests, the corrosion rate ratio described in SAR Section 2.3.6.8.1.3 was not needed. DOE decided to follow an alternative approach in which the corrosion rates for the Titanium Grade 29 structural supports are the same as the corrosion rate for the outer surface of the Titanium Grade 7 plate. DOE stated that this approach is justified because the corrosion rates measured by electrochemical impedance spectroscopy and linear polarization resistance for the respective alloys in the 42-day tests were nominally identical (Andresen and Kim, 2006aa). DOE also referenced Schutz (2005aa), which showed that the corrosion rates of Titanium Grades 7 and 29 are similar when exposed in a chloride solution with pH greater than 1, as shown in BSC (2004as, Figure 6-22[a]).

In addition, DOE (2009cn) performed a sensitivity analysis using the TSPA model that compared the approach described in SAR Section 2.3.6.8.1.3 (in which the ratio for the corrosion rate of Titanium Grade 29 to that of Titanium Grade 7 was sampled from a probability distribution function with a value in the range of approximately one to seven) to the new approach, in which the corrosion rate of Titanium Grade 29 is assumed to be equivalent to that of Titanium Grade 7. The analysis revealed that the drip shield structural framework failure time occurred later for the new approach, as shown in DOE (2009cn, Enclosure 2, Figure 1). The analysis also showed that, in the event of a seismic ground motion, the new approach gives a median dose that is about 25 percent higher between 80,000 and 300,000 years after repository closure, due to increased probability of waste package damage, as shown in DOE (2009cn, Enclosure 2, Figure 3). DOE stated that the mean expected dose was nearly the same for the respective approaches because the contribution of the seismic ground motion modeling case to the total mean annual dose is small during this time period. Therefore, DOE concluded that the data presented in SAR Section 2.3.6.8.1 were acceptable to use in the TSPA calculation because unquantified experimental uncertainties had a negligible effect on the results from the postclosure performance assessment calculation.

NRC Staff's Review

The NRC staff reviewed DOE's approach for calculating the general corrosion rate of Titanium Grade 29. The NRC staff finds that DOE's assumption of equivalent corrosion rates for Titanium Grades 7 and 29 is primarily based on corrosion testing of the alloys in simulated seepage water and deliquescent brines, including fluoride-bearing solutions (Andresen and Kim, 2006aa). The NRC staff notes that the corrosion rate for Titanium Grade 29 in the repository may depend on such factors as the pH and concentration of ionic species in water that contact the drip shield. Therefore, the NRC staff reviewed the corrosion test solutions to determine whether they are adequate to model the Titanium Grade 29 corrosion rate in repository conditions. The NRC staff finds that DOE's corrosion test solutions are more chemically aggressive than waters expected to occur within repository drifts, including starting water compositions in DOE's near-field chemistry model described in SAR Section 2.3.5.5, and waters considered in the NRC staff's independent analysis of in-drift water evolution, described in SER Section 2.2.1.3.3.3.2. As such, the NRC staff concludes that the solutions given in BSC (2004as, Table 6-7[a]) are acceptable to calculate the general corrosion rate of Titanium Grade 29 because they are more chemically aggressive than waters expected to occur within the repository drifts. The NRC staff finds additional support for the assumption of equivalent corrosion rates in Schutz (2005aa), which showed the equivalent corrosion rate is observed even in acidic solutions. Finally, the NRC staff also observed nearly equivalent corrosion rates during testing of Titanium Grades 7 and 29 in 1 M NaCl and 4 M $MgCl_2$ solutions at elevated temperatures (Mintz and He, 2009aa). The NRC staff notes that the assumption of equivalent corrosion rates would extend the drip shield framework lifetime compared to that calculated in the TSPA model, where the corrosion rate of Titanium Grade 29 was up to a factor of seven times faster than that for Titanium Grade 7. The NRC staff finds that DOE's sensitivity analysis demonstrated that the TSPA calculations would not significantly differ when implementing the assumption of equivalent corrosion rates.

Abstraction and Integration

For the Nominal Modeling Case in the TSPA analysis, DOE implemented the model abstraction for general corrosion of the drip shield in the Waste Package and Drip Shield Degradation Submodel, as described in SNL (2008ag, Section 6.3.5.1), in which the drip shields were distributed in the five percolation subregions. The Waste Package and Drip Shield Degradation Submodel considers only general corrosion breach of the Titanium Grade 7 plates. DOE concluded that the drip shield would protect the waste package against seepage if the drip shield plates are intact, even if the drip shield supports collapsed and the sidewall buckled (SAR Section 2.3.4.5.3.1). For each realization, DOE sampled one general corrosion rate for the outside of the drip shield plates (under the aggressive condition) and one for the inside of the drip shield plates (under the benign condition) from the respective distributions given in SNL (2008ag, Table 6.3.5-3). The corrosion rates were applied to all drip shields, regardless of the percolation subregion, such that all drip shields in a given realization failed at the same time. The output of the Waste Package and Drip Shield Degradation Submodel was the fraction of that for drip shields in each percolation subregion breached by general corrosion as a function of time. This output was provided to the Waste Form Degradation and Mobilization Model Component and the Engineered Barrier System Flow and Engineered Barrier System Transport Submodels.

SAR Figures 2.1-8 and 2.4-24 showed the distribution of calculated failure times for the Titanium Grade 7 drip shield plates in the Nominal Modeling Case, on the basis of the model described in SAR Section 2.3.6.8.1. DOE's analyses calculated that most drip shield failures occur between 260,000 and 340,000 years after repository closure. DOE (2009cn, Enclosure 5, Figure 1) showed a modified distribution of failure times considering both a higher corrosion rate (based on additional uncertainties associated with specimen cleaning) and lower corrosion rate (based on potential decrease in corrosion rate over time). The modified distribution shows that most drip shield plate failures occur between 80,000 and 500,000 years after repository closure. In either case, DOE concluded that there is negligible probability of drip shield plate breach by nominal processes within 12,000 years after repository closure, the time period during which DOE calculates that the waste package is susceptible to localized corrosion if contacted by seepage water.

For the Seismic Ground Motion Modeling Case in the TSPA analysis, DOE also implemented the Waste Package and Drip Shield Degradation Submodel to calculate the timing and magnitude of drip shield plate breach by general corrosion, as outlined in SNL (2008ag, Section 6.6.1). Both the Titanium Grades 7 and 29 corrosion rates are sampled in this modeling case. The Titanium Grade 7 corrosion rate was sampled in the same manner as in the Nominal Modeling Case. For Titanium Grade 29 structural supports, DOE sampled the ratio of the corrosion rate of Titanium Grade 29 to that of Titanium Grade 7, once per realization, from the discrete probability distribution function summarized in BSC (2004as, Table 6-8[a]). The ratio was applied to all drip shields in a realization. SAR Figures 2.1-11 and 2.4-24 showed the distribution of failure times for the Titanium Grade 7 drip shield plates in the Seismic Ground Motion Modeling Case. Most plate failures occur between 100,000 and 300,000 years after repository closure. DOE stated that there is negligible probability of drip shield breach within 12,000 years after repository closure because the general corrosion rate of the Titanium Grade 7 drip shield plates is low, and the likelihood of plate failure by a seismic event is negligible before that time period. For the Titanium Grade 29 structural supports, DOE calculated that most drip shield frameworks failed between 20,000 and 170,000 years after repository closure, using the model described in SAR Section 2.3.6.8.1 and DOE (2009cn, Enclosure 2, Figure 1). For the alternative approach, in which DOE assumed equivalent corrosion rates for the structural supports and the plate, DOE calculated that most frameworks failed between about 80,000 and 170,000 years after repository closure, as shown in DOE (2009cn, Enclosure 2, Figure 1).

NRC Staff's Review

The NRC staff reviewed the implementation and integration of the model abstraction for general corrosion of the drip shield used in the postclosure performance assessment calculation. The NRC staff finds that DOE has provided sufficient information for the NRC staff to understand how the conceptual model is implemented in the TSPA code and how the model inputs and outputs are integrated with other model components. The NRC staff determines that DOE's use of the model abstraction is consistent with the design features of the drip shield, including materials of construction and dimensions given in SAR Section 1.3.4.7. Further, the NRC staff finds that, with respect to the general corrosion rates, DOE adequately justified the data and model used because DOE showed the ranges of these parameters and accounted for uncertainty in the model abstraction, as summarized in SER Section 2.2.1.3.1.3.1.1. Therefore, the NRC staff finds DOE's implementation of the drip shield general corrosion model abstraction in the TSPA code acceptable because it would not overestimate the timing or underestimate the magnitude of radionuclide release to the accessible environment.

NRC Staff's Summary of Evaluation Findings for General Corrosion of the Drip Shield

The NRC staff reviewed DOE's model abstraction for general corrosion of the drip shield that was implemented in the TSPA code. On the basis of its review, the NRC staff finds that DOE used appropriate experimental tests and other independent technical literature to provide adequate support for the model abstraction. In addition, DOE appropriately identified and adequately considered processes and features such as the general corrosion of the drip shield that affect barrier capabilities for the initial 10,000 year period, and projected these processes and features beyond the 10,000 year post-disposal period through the period of geologic stability. Therefore, the NRC staff finds that DOE acceptably accounted for general corrosion of the drip shield in the TSPA model.

2.2.1.3.1.3.1.2 Drip Shield Early Failure

In SAR Section 2.3.6.8.4, DOE described how it developed the probability distribution for early failure of the drip shield that was sampled in the TSPA code. DOE assumed that a drip shield experienced early failure if it was emplaced in the repository with an undetected manufacturing- or handling-induced defect. On the basis of the processes associated with drip shield manufacturing and handling, DOE concluded that the probability of a drip shield early failure is best represented in the TSPA by a lognormal distribution with a mean of 2.21×10^{-6} per drip shield and an error factor of 14, as shown in SNL (2007aa, Table 6-7). The NRC staff reviewed the adequacy of this probability distribution in SER Section 2.2.1.2.2.4. The implementation of this probability distribution is addressed in this section.

Drip Shield Early Failure Conceptual Model

In DOE's conceptual model for early drip shield failure, a drip shield with an undetected manufacturing- or handling-induced defect completely fails (i.e., is removed as a barrier to the flow of water) at the time of repository closure (SAR Section 2.3.6.8.4.4.1). DOE selected this representation because there are uncertainties associated with the timing and extent of breach for defective drip shields and a completely degraded drip shield at the time of repository closure will not overestimate the timing and underestimate the magnitude of radionuclide releases, as described in SNL (2007aa, Section 6.5.2). DOE concluded that this is a conservative representation of the early failed drip shield because the most likely consequence of improper drip shield manufacturing or handling would be stress corrosion cracking. DOE excluded drip shield stress corrosion cracking from the performance assessment because, even if cracking occurred, the cracks would not affect the drip shield performance because advective flow through cracks in the drip shield is also excluded from the performance assessment (SAR Section 2.3.6.8.3).

NRC Staff's Review

The NRC staff reviewed DOE's conceptual model for drip shield early failure, as described in SAR Section 2.3.6.8.4. The NRC staff finds that DOE attributed no barrier capability to the early failed drip shield. Based on metal manufacturing knowledge, the NRC staff notes that consequences related to manufacturing- or handling-induced defects would likely allow the drip shield to maintain some barrier capability, which limits radionuclide releases. Because early failed drip shields in DOE's model are assumed to have no barrier capability, the NRC staff finds that the model will not cause DOE to overestimate the timing or underestimate the magnitude of radionuclide releases. Therefore, the NRC staff finds acceptable DOE's conceptual model for drip shield early failure.

Abstraction and Integration

The model abstraction for early failure of the drip shield was implemented in the TSPA calculation in the Drip Shield Early Failure Modeling Case, as described in SAR Section 2.4.2.1.5.2 and SNL (2008ag, Section 6.4.1). This modeling case uses most of the same modeling components and submodels as were implemented in the Nominal Modeling Case. In the Nominal Modeling Case, however, the Waste Package and Drip Shield Degradation Submodel calculates the waste package and drip shield breached areas as a function of time and passes this to the Engineered Barrier System Flow and Transport Submodels and the Waste Form Degradation and Mobilization Model Components. In the Drip Shield Early Failure Modeling Case, the Waste Package and Drip Shield Degradation Submodel was replaced with the drip shield early failure model, which simulated early failure by removing a selected drip shield as a barrier to seepage at the time of repository closure.

In the Drip Shield Early Failure Modeling Case, the underlying waste package immediately experienced initiation of localized corrosion if the early failed drip shield was exposed to seepage conditions. If the early failed drip shield was not exposed to seepage conditions, the underlying waste package did not experience initiation of localized corrosion. In the TSPA model, DOE calculated the dose consequence of a drip shield early failure in each of the five percolation subregions for both commercial spent nuclear fuel (CSNF)-type and codisposal-type waste packages. DOE then calculated the expected dose using the early failure probability [sampled from the distribution given in SNL (2007aa, Table 7-1)], the distribution for the waste package type, and the seepage fraction for each percolation bin.

DOE calculated that there is approximately 98.3 percent probability of no drip shield early failures, approximately 1.6 percent probability of one drip shield early failure, and approximately 0.1 percent probability of two or more drip shield early failures, as shown in SNL (2008ag, Table 6.4-1). Using the TSPA model, DOE calculated that drip shield early failure has a negligibly small contribution to the calculated mean annual dose during the first 10,000 years following closure {less than 10^{-8} Sv [10^{-3} mrem]}, with a declining contribution thereafter (SAR, Figure 2.4-18).

NRC Staff's Review

The NRC staff reviewed the implementation of the drip shield early failure model in the TSPA calculation, as described in SNL (2008ag, Section 6.4.1). The NRC staff finds that DOE has provided sufficient information for the NRC staff to understand how the conceptual model is implemented in the TSPA code and how the model inputs and outputs are integrated with other model components. The NRC staff determines that the model abstraction is consistent with the design features of the drip shield, including materials of construction and dimensions given in SAR Section 1.3.4.7. Further, the NRC staff concludes that DOE adequately justified the data and model used because DOE showed the ranges of these parameters and accounted for uncertainty in the model abstraction as summarized in SER Section 2.2.1.3.1.3.1.2. Therefore, the NRC staff finds DOE's implementation of the drip shield early failure model abstraction in the TSPA analysis acceptable because it will not overestimate the timing or underestimate the magnitude of radionuclide release to the accessible environment.

NRC Summary of Evaluation Findings for Drip Shield Early Failure

The NRC staff reviewed DOE's model abstraction for early failure of the drip shield that was implemented in the TSPA code. On the basis of its review, the NRC staff finds that DOE provided adequate support for its model abstraction. In addition, DOE appropriately identified and adequately considered features and events such as drip shield early failure that affect barrier capabilities for the initial 10,000 year period, and projected these features and events beyond the 10,000 year post-disposal period through the period of geologic stability. Therefore, the NRC staff finds that DOE acceptably accounted for drip shield early failure in the TSPA model.

2.2.1.3.1.3.2 Waste Package Degradation

In SAR Section 1.5.2, DOE stated that waste packages are relied upon to limit water contacting the waste form and to prevent the mobilization of radionuclides. The waste package will have an outer barrier that is fabricated from a material that is expected to be corrosion resistant in the range of environmental conditions expected in the repository (SAR Section 2.3.6.1). In particular, the waste package outer barrier will be fabricated from Alloy 22 (UNS N06022), which is a nickel-chromium-molybdenum alloy.

In developing the postclosure performance assessment, DOE evaluated numerous features, events, and processes in SAR Table 2.2-5 related to chemical degradation of the waste package. These FEPs include

- General corrosion of waste packages (FEP 2.1.03.01.0A)
- Stress corrosion cracking of waste packages (FEP 2.1.03.02.0A)
- Localized corrosion of waste packages (FEP 2.1.03.03.0A)
- Hydride cracking of waste packages (FEP 2.1.03.04.0A)
- Microbially influenced corrosion (MIC) of waste packages (FEP 2.1.03.05.0A)
- Internal corrosion of waste packages prior to breach (FEP 2.1.03.06.0A)
- Early failure of waste packages (FEP 2.1.03.08.0A)
- Creep of metallic materials in waste packages (FEP 2.1.07.05.0A)
- Localized corrosion on waste packages' outer surface due to deliquescence (FEP 2.1.09.28.0A)
- Thermal sensitization of waste packages (FEP 2.1.11.06.0A)

DOE included general corrosion, stress corrosion cracking, localized corrosion, MIC, and early failure in the postclosure performance assessment. Other FEPs were screened from the performance assessment on the basis of low consequence or low probability (SAR Table 2.2-5). The NRC staff's evaluation of DOE's bases for excluding these FEPs from the performance assessment is found in SER Section 2.2.1.2.1.3.

In the TSPA analysis, DOE calculated that, due to its corrosion resistance, the waste package will significantly reduce the amount of water contacting the waste form for hundreds of thousands of years after repository closure (SAR Section 2.1.2.2.6). Because of the importance of the waste package in the postclosure performance assessment, the NRC staff reviewed DOE's model abstractions for waste package chemical degradation. In the context of these reviews, the NRC staff notes that DOE attributed the high corrosion resistance of Alloy 22, in part, to the presence of its passive film. In the event of deterioration or loss of waste package passivity, the time to waste package breach may be sooner and the size of the breached area may be larger than DOE calculated in the TSPA code. As such, DOE stated that long-term persistence of the passive film on Alloy 22 is one of the key issues that it considered to determine the long-term performance of the waste package in the repository, as described in SNL (2007al, Section 6.4.1.1). In NRC (2005aa, Appendix D, Section 4.3.1), the NRC also identified the long-term persistence of the passive film on the waste package outer barrier as being of high significance to risk for waste isolation.

Therefore, in the following sections, the NRC staff first reviews DOE's approach to support its assessment of Alloy 22 passive film stability under the repository conditions. The NRC staff then conducts a detailed review of DOE's model abstractions for chemical degradation of the waste package, including general corrosion, MIC, localized corrosion, stress corrosion cracking, and early failure.

Passivity of Alloy 22

In SAR Section 2.3.6.3.1 and SNL (2007al), DOE indicated that the stability of the Alloy 22 passive film depends primarily upon its physical and chemical properties, including microstructure, composition, and thickness. On Alloy 22 corrosion specimens, DOE investigated these passive film properties with various surface analytic techniques, including Auger electron spectroscopy, transmission electron microscopy, x-ray photoelectron spectroscopy, and electron energy loss spectroscopy (Orme, 2005aa). DOE performed short-term polarization tests, exposing Alloy 22 samples at 90 °C [194 °F] to solutions with a range of chemical compositions that DOE assumed were similar to, or more aggressive than, those expected in the repository (Orme, 2005aa). The solutions used in short-term polarization tests were either buffered 1 M NaCl solutions or multi-ionic solutions, including simulated acidified water, simulated concentrated water, and basic saturated water (compositions given in SAR Table 2.3.6-1). To assess the long-term passive film behavior, DOE examined 5-year U-bend samples of Alloy 22 exposed to simulated acidified water, simulated concentrated water, and simulated dilute water at 90 °C [194 °F] (Orme, 2005aa).

For both short- and long-term tests, DOE observed a thin, adherent passive oxide film on the surface of Alloy 22 corrosion specimens. The film typically had thickness in the range of 2 to 7 nm [7.87×10^{-8} to 2.76×10^{-7} in] and tended to be rich in chromium (III) oxides (Cr_2O_3 and/or $NiCr_2O_4$). In the solutions of acidic and near-neutral pH, a thick outer layer was also observed on the top of the inner chromium-rich oxide layer (Orme, 2005aa). The outer layer was porous and consisted mostly of nickel oxide and the oxides of some other alloying elements, including iron, tungsten, and molybdenum. In basic saturated water (pH ~12–13), DOE observed a thick silica deposit on Alloy 22 specimens (Orme, 2005aa), which DOE concluded arose from dissolution of test cell glassware or precipitation of silica from the test solution. In the case of 5-year U-bend samples exposed to simulated acidified water, simulated concentrated water, and simulated dilute water, all of the immersed samples had 100 to 5,000 nm [3.94×10^{-6} to 1.97×10^{-4} in] thick carbon and iron deposits on their surfaces. DOE determined the deposits are formed as leachates from either the walls of the test tanks or other metals in the tanks (Orme, 2005aa). Oil from the mill processing is also considered to be included in the deposits. DOE stated that, underneath these deposits, the passive film was still close to 5 nm [1.97×10^{-7} in] thick after 5 years of exposure. The presence of chromium-rich oxide passive film on the Alloy 22 surface was also observed at high temperatures {in the range of 120 to 220 °C [248 to 428 °F]} in $NaCl$–$NaNO_3$–KNO_3 solutions (Orme, 2005aa; Dixit, et al., 2006aa).

To support the assessment of long-term passive film stability, DOE performed thermodynamic modeling with the EQ3/6 program (Orme, 2005aa). DOE concluded that this demonstrated that chromium-rich oxides are stable on Alloy 22, which is consistent with empirical observation of passive film chemistry. Although the tests that DOE used to characterize the passive film of Alloy 22 were for a period of, at most, 5 years, DOE referenced the point defect film growth model, which states that the passive film on Alloy 22 will maintain steady-state thickness as the porous outer layer dissolves and the compact chromium-rich oxide inner layer is continuously regenerated.

After the license application was submitted, upon the NRC staff's request, DOE examined some 5- and 9.5-year Alloy 22 specimens from the Long-Term Corrosion Test Facility

(SNL, 2009aa, ab). DOE identified thick organic deposits on some specimens. DOE responded to the NRC staff's RAI on evaluation of the effects of the carbon deposits on DOE's assessment of long-term passivity and corrosion behavior (DOE, 2009cl, 2010ae). In its response, DOE stated that the organic deposits on the Alloy 22 specimens most likely originated from lubricant or grease from mechanical equipment in the corrosion test facility. DOE did not identify evidence of either increased general corrosion rate or localized corrosion attack on the specimens. For these specimens, DOE measured a corrosion rate of 3 to 5 nm/yr [1.18×10^{-7} to 1.97×10^{-7} in] in simulated concentrated water at 60 °C [140 °F] after 9.5 years. DOE determined that this corrosion rate was consistent with that of uncontaminated specimens, as well as the waste package corrosion rate used in the TSPA model.

Therefore, DOE concluded that the organic deposits did not affect the assessment of long-term passivity or corrosion behavior.

NRC Staff's Review

The NRC staff reviewed DOE's approach to establish the stability of the waste package passive film in repository conditions. The NRC staff determines that DOE's assumption that the passive film will remain stable during the postclosure period is based, in part, on tests of Alloy 22 specimens in a range of corrosion test solutions, including simulated acidified water, simulated concentrated water, simulated dilute water, and basic simulated water (NRC staff's evaluation of FEP 2.1.09.28.0A, SER Section 2.2.1.2.1.3.2). The NRC staff concludes that stability of the waste package passive film may depend on such factors as the pH and concentration of ionic species in water that contacts the waste package. Therefore, the NRC staff reviewed the corrosion test solutions to determine whether they are adequate in assessing waste package passive film stability in repository conditions. The NRC staff finds that the corrosion test solutions are more chemically aggressive than waters expected to occur within repository drifts, including starting water compositions in DOE's near-field chemistry model described in SAR Section 2.3.5.5, and waters considered in NRC staff's independent analysis of in-drift water evolution, described in SER Section 2.2.1.3.3.3.2. Therefore, the NRC staff concludes that DOE's observations of Alloy 22 passive film stability in simulated acidified water, simulated concentrated water, simulated dilute water, and basic simulated water support DOE's assumption of waste package passive film stability in repository conditions because the simulated water chemistries bound repository conditions. This conclusion is supported by similar descriptions of the passive film for Alloy 22 and analogous nickel-based alloys (e.g., Alloy C-4, C276, 600, 625, and 690) reported by NRC staff (Dunn, et al., 2005aa) and others (Lloyd, et al., 2003aa, 2004aa; Gray, et al., 2006aa; Montemor, et al., 2003aa; Hur and Park, 2006aa; Mintz and Devine, 2004aa).

Regarding the organic deposits on the corrosion specimens, the NRC staff reviewed DOE reports (SNL, 2009aa,ab) and determines that the presence of the deposits did not significantly affect the measured corrosion rates, given the range of uncertainty for such measurements. The NRC staff also determines that there is no evidence of localized corrosion (e.g., pitting) on the specimens. The NRC staff finds that DOE conducted a reasonable analysis to show that organic deposits are not likely to affect anodic and cathodic reactions on the Alloy 22 surface. Therefore, the NRC staff concludes that the carbon deposits on some Alloy 22 specimens did not affect the assessment of the passive film stability.

The NRC staff also reviewed DOE's use of thermodynamic modeling and the point defect film growth model to support the assessment of the waste package passive film stability. The NRC staff conducted independent analyses and obtained similar results as DOE for temperatures up to 180 °C [356 °F] (Pensado, et al., 2002aa; Jung, et al., 2008aa). With respect to application of the point defect film growth model in DOE's assessment of passive film stability, the NRC staff concludes that DOE's test results (Orme, 2005aa) indicate that a passive film with steady-state thickness would provide a nearly constant corrosion rate over time at a particular temperature, potential, and pH. The NRC staff finds that the results from the Miserque, et al. (2006aa) study confirm the acceptability of DOE's approach for using a point defect film growth model to support passive film stability.

Although in the license application, DOE provided information to support the waste package passive film stability in repository conditions, the NRC staff identified three primary technical issues and requested additional information from DOE. These issues involved passive film degradation by (i) anodic sulfur segregation, (ii) dripping seepage water, and (iii) silica deposits on the waste package. The NRC staff's reviews of these technical issues are presented next.

Effect of Anodic Sulfur Segregation on Passive Film Stability

By independent analyses and review of the technical literature, the NRC staff identified anodic sulfur segregation as a potential mechanism that could compromise the long-term stability of the passive film on the waste package outer barrier (NRC, 2005aa; U.S. Nuclear Waste Technical Review Board, 2001aa). Anodic sulfur segregation is a process that reduces the corrosion resistance of nickel and nickel-iron alloys by inhibiting the formation of the passive film (Marcus and Talah, 1989aa; Marcus, et al., 1988aa, 1984aa,ab, 1980aa). During anodic sulfur segregation, sulfur, which may be an impurity in Alloy 22, segregates to the metal-passive film interface because of selective dissolution of bulk metal elements such as nickel and iron. When the amount of sulfur at the metal-passive film interface reaches a critical concentration of about one atomic layer thickness, passive film breakdown has been observed (Marcus and Grimal, 1990aa). Assuming that 100 percent of the sulfur atoms in the alloy are retained at the metal-film interface, Marcus (2001aa) estimated that it would take about 900 years for the passive film of Alloy 22 to break down if the sulfur content in Alloy 22 is 5 weight parts per million (ppm) and the passive current density is 1 nA/cm^2 [6.45 nA/in^2].

In DOE (2009cl, Enclosure 5), DOE stated that the potential for anodic sulfur segregation would be mitigated by the presence of the alloying elements chromium and molybdenum in Alloy 22. In particular, molybdenum would bond with sulfur to form a molybdenum sulfide that dissolves under aqueous conditions, thus preventing a stable sulfur monolayer from forming at the alloy-passive film interface. Citing the work of NRC (Jung, et al., 2007aa), DOE also stated that chromium oxides are thermodynamically stable compared with sulfides. Thus, the presence of chromium will promote passivation in spite of adsorbed sulfur.

NRC Staff's Review

The NRC staff reviewed DOE's assessment of anodic sulfur segregation of the waste package outer barrier in DOE (2009cl, Enclosure 5). The NRC staff finds that the effects of chromium and molybdenum on sulfur segregation were not considered in the time to passive film breakdown estimated in Marcus (2001aa). Further, the NRC staff determines that reports of passive film breakdown by anodic sulfur segregation are primarily associated with iron and iron-nickel alloys that lack the alloying elements molybdenum and chromium. The NRC staff finds that the presence of the alloying elements molybdenum and chromium has been shown to prevent passive film breakdown by anodic sulfur segregation in materials similar to Alloy 22. Literature reports (Costa and Marcus, 1993aa; Marcus and Grimal, 1990aa), indicate that chromium in Ni-xCr-10Fe alloys (x = 8, 19, and 34 at%) counteracted the detrimental effects of sulfur and promoted alloy passivation. Further, it was reported that preadsorbed sulfur monolayers on the surface of Ni-2~6Mo and Fe-17Cr-14.5Ni-2.3Mo stainless steel, respectively, were removed in the form of soluble molybdenum sulfides during alloy dissolution (Marcus and Moscatelli, 1989aa; Elibiache and Marcus, 1992aa). It was also reported in the literature (Mulford and Tromans, 1988aa) that Alloy 625 (Ni-21.5Cr-9Mo-4Fe) was resistant to localized creviced corrosion in 1 M NaCl solution containing 0.01 M $Na_2S_2O_3$ up to 80 °C [176 °F], whereas Alloy 600 (Ni-15.5Cr-8Fe) with no molybdenum was attacked by crevice corrosion in the same solution at 20 °C [68 °F].

The NRC staff further determines that the information provided in DOE (2009cl, Enclosure 5) is consistent with independent investigations NRC conducted (Ahn, et al., 2008aa; Jung, et al., 2007aa, 2008aa). In particular, NRC performed electrochemically accelerated dissolution tests in aggressive solutions to assess whether sulfur would segregate to the waste package outer surface of Alloy 22 during anodic dissolution in a longer time. Surface analysis showed that almost all sulfur on the surface dissolved during the tests, such that a critical sulfur concentration for passive film breakdown was not reached. Scratch repassivation tests of Alloy 22 in sulfide-containing solutions showed that even if passive-film breakdown occurred, the material repassivated within a few seconds, which would limit the potential for corrosion degradation (Jung, et al., 2008aa). These experimental results also are supported by thermodynamic calculations that calculate a formation of stable chromium oxide (Cr_2O_3) and possible soluble molybdenum sulfide (MoS_2) in the range of temperatures, potentials, and pH expected for repository conditions (Jung, et al., 2007aa, 2008aa).

The NRC staff therefore concludes that in the aerated oxic environment expected in the repository, sulfur that accumulates on the waste package surface will most likely be removed by dissolution in either the reduced form or as soluble molybdenum sulfides. The NRC staff finds that these processes would mitigate the potential for anodic sulfur segregation to affect the long-term stability of passive films on Alloy 22. Therefore, on the basis of the previous evaluation, the NRC staff finds that DOE has provided sufficient information to demonstrate that the potential effects of sulfur segregation do not need to be considered in DOE’s model for passive film stability on the waste package outer barrier.

Dripping Effects on the Passive Film Stability

DOE used corrosion data from immersion experiments to develop the model abstraction for general corrosion of the waste package outer barrier (SAR Section 2.3.6.3.2). DOE identified the possibility, however, that conditions in the repository may lead to dripping seepage water contacting the waste package surface (SAR Section 2.1.2.2.6). The NRC staff determines that literature information in Ashida, et al. (2007aa, 2008aa) and Oka, et al. (2007aa), shows that general corrosion processes may be affected if environmental conditions changed from immersion to dripping. Ashida, et al. (2008aa) observed salt deposit formation and localized corrosion (i.e., pitting and intergranular corrosion) on Alloy 22 specimens exposed to dripping of simulated concentrated water for 40 days at 90 °C [194 °F]. The micropits observed were, however, not stable, and there was no evidence for propagation of these micropits. Ashida, et al. (2007aa) also reported an increase of the passive current density of Alloy 22 due to dripping-induced temperature fluctuations at 90 °C [194 °F].

In DOE (2009cm, Enclosure 1), DOE assessed the corrosion behavior of Alloy 22 under dripping conditions in the repository environment. DOE stated that the Alloy 22 sample tested in Ashida, et al. (2008aa) was thermally aged, resulting in a significant second phase precipitation.

This precipitate can decrease a resistance to localized corrosion. DOE stated that the Alloy 22 for the waste package outer barrier will be solution annealed, eliminating the second phase precipitates. Therefore, DOE concluded the material Ashida, et al. (2008aa) evaluated was not relevant for the waste package. In the case of the change in passive current density due to temperature fluctuation, as shown in the polarization test in Ashida, et al. (2007aa), DOE stated that the current TSPA model of general corrosion of Alloy 22 considers the change of general corrosion rate depending on the temperature (see SER Section 2.2.1.3.1.3.2.1); therefore, DOE concluded that the increase of corrosion rate observed in Ashida, et al. (2007aa) is consistent with the TSPA general corrosion model.

NRC Staff's Review

The NRC staff reviewed the information provided in DOE (2009cm, Enclosure 1). The NRC staff finds that solution annealing of the waste package will eliminate the second phase particles that decreased localized corrosion resistance in the specimens tested by Ashida, et al. (2008aa). However, the NRC staff determines that the second phase may not be eliminated in the region of the waste package closure weld because this weld will not be solution annealed (SAR Section 1.5.2.7). Although Ashida, et al. (2008aa) observed micropits in such material conditions, the NRC staff concludes that the same report shows no evidence for stable pit propagation (Ashida, et al., 2008aa). Moreover, NRC's independent analyses (Dunn, et al., 2006ab) identified no localized corrosion (e.g., pitting or intergranular corrosion) of the mill-annealed Alloy 22 after dripping simulated pore waters onto the Alloy 22 specimens at 110 °C [230 °F] for 10 days. Although Jung, et al. (2011aa) observed micropits under dripping seepage water on Alloy 22, the pits appeared to be shallow and there was no clear evidence of pit propagation after testing of 185 days. Therefore, the NRC staff concludes that micropitting of the waste package closure weld is unlikely to have a significant effect on the waste package barrier capability.

Further, the NRC staff noted that the temperature of the water used in the dripping tests in Ashida, et al. (2007aa) was close to room temperature. This resulted in a relatively large difference in the temperature between the room temperature of the dripped water and the hot surface of the Alloy 22 test specimen {i.e., 90 °C [194 °F]}, thereby contributing to the observed increase in passive current due to temperature fluctuation. The NRC staff finds that such temperature fluctuations will be much smaller for the waste package in the repository because of a relatively smaller difference in temperature between the waste package and the drift wall (Jung, 2010aa). The NRC staff finds that this small temperature fluctuation is not likely to have a significant effect on the waste package general corrosion rate.

Therefore, on the basis of the aforementioned evaluation, the NRC staff finds that DOE has provided sufficient information to demonstrate that dripping conditions in the repository will not affect the assessment of the waste package passive film stability.

Effect of Silica Deposits on Alloy 22 Passivity

DOE data and information in the technical literature indicate that silica (solid) deposits on Alloy 22 may affect the passive film. In basic simulated water (pH ~12 to 13), DOE observed a thick silica deposit on Alloy 22 specimens (Orme, 2005aa), which DOE concluded arose from dissolution of test cell glassware or precipitation of silica from the test solution. DOE also noted the presence of silica in salt deposits on Alloy 22 specimens exposed to simulated dilute water, simulated acidified water, and simulated concentrated water in the Long-Term Corrosion Test Facility, as shown in Wong, et al. (2004aa, Table 4, Figures 2 and 3). In another experiment

(Dixit, et al., 2006aa) DOE observed silica deposits on Alloy 22 specimens that experienced localized corrosion in a deaerated concentrated solution at 220 °C [428 °F]. Finally, information in Sala, et al. (1993aa, 1996aa, 1998aa, 1999aa) indicated that the presence of silica deposits can be associated with intergranular attack and stress corrosion cracking in nickel-based alloys in steam-generator environments.

In DOE (2009cl, Enclosure 4) and DOE (2009cm, Enclosure 2), DOE stated that the presence of silicate (aqueous species) in the test solutions did not significantly impact the corrosion potential and corrosion rate of Alloy 22 for tests conducted in simulated acidified water and NaCl solutions. DOE also presented experimental data (Andresen and Kim, 2007aa) for Alloy 22 tests in solutions of nitrate, chloride, and bicarbonate with 0.27 molal silicate. The tests showed that the general corrosion rate of Alloy 22 was 3 to 4 nm/yr [1.18×10^{-7} to 1.57×10^{-7} in/yr] after 62 months' immersion at 95 °C [203 °F], which is close to the measured corrosion rate in solutions without silicate. Finally, DOE stated that Sala, et al. (1993aa, 1996aa, 1998aa, 1999aa) and Dixit, et al. (2006aa) considered more aggressive environmental conditions than those expected in the repository. Therefore, DOE concluded that the observations are not relevant to the waste package in the repository.

NRC Staff's Review

The NRC staff reviewed DOE's approach to assess the effects of silica deposits on the waste package passive film stability. On the basis of its review of the information DOE provided (e.g., Andresen and Kim, 2007aa), the NRC staff finds that silica deposits on Alloy 22 specimens do not adversely impact Alloy 22 passivity because there is not a significant difference in measured corrosion rates for specimens with and without silica deposits. Moreover, the NRC staff also finds that Sala, et al. (1993aa, 1996aa, 1998aa, 1999aa) and Dixit, et al. (2006aa) considered more aggressive environmental conditions than those expected in the repository.

Therefore, on the basis of the evaluation regarding passive film stability, the NRC staff finds that DOE has provided sufficient information in Section 2.3.6.3.1 of the SAR and in responses to the NRC staff's request for additional information to support the conclusion that the waste package passive film will be stable during the postclosure period.

2.2.1.3.1.3.2.1 General Corrosion of the Waste Package Outer Barrier

In SAR Section 2.3.6.3, DOE defined general corrosion of the waste package outer barrier as uniform thinning by electrochemical processes at its corrosion potential. General corrosion could lead to the release of radionuclides from the waste package if the waste package wall is breached. General corrosion thinning may also make the waste package more susceptible to degradation processes such as stress corrosion cracking (SAR Section 2.3.6.5) or impacts caused by seismic ground motion (SAR Section 2.3.4.5). This section of the SER includes the NRC staff's review of DOE's model abstraction for general corrosion of the waste package outer barrier.

Waste Package General Corrosion Conceptual Model

In DOE's conceptual model for general corrosion of the waste package outer barrier, general corrosion starts at the time of repository closure (SAR Section 2.3.6.2.2). DOE assumed aqueous conditions because wet conditions give higher corrosion rates than dry conditions (SAR Section 2.3.6.3). In DOE's model, the general corrosion rate is a function of the waste package temperature and, at a given temperature, it is assumed to be constant over time (SAR Section 2.3.6.3.1). DOE used an Arrhenius-type equation (SAR Equation 2.3.6-3) to calculate the temperature-dependent general corrosion rate of the waste package outer barrier in the temperature range of 25 to 200 °C [77 to 392 °F]. DOE also considered using a decreasing general corrosion rate over time as an alternative conceptual model (SNL, 2007al) but concluded that this would calculate a longer time to waste package failure.

DOE also considered that microbial activity in the repository could affect the waste package corrosion behavior—a phenomenon called MIC (SAR Section 2.3.6.3.3.2). DOE stated that microorganisms can change the electrochemical reactions on the material surface and change the type or degree of corrosion compared to that which would be measured in the absence of microorganisms. For example, MIC can enhance the general corrosion rate of Alloy 22. In DOE's conceptual model, the waste package outer barrier is subject to MIC when the relative humidity is sufficiently high for microbial activities. The effect of MIC on the general corrosion rate is quantified by a unitless scalar called the microbially influenced corrosion enhancement factor. If the relative humidity is sufficiently high, the general corrosion rate in the absence of the microorganisms (SAR Equation 2.3.6-3) is multiplied by the MIC enhancement factor to give the enhanced general corrosion rate (SAR Equation 2.3.6-4).

NRC Staff's Review

The NRC staff reviewed DOE's conceptual model for general corrosion of the waste package outer barrier. The NRC staff reviewed DOE's model assumption that the temperature dependence of the general corrosion rate can be quantified with the Arrhenius-type equation. Corrosion involves chemical and/or electrochemical reactions and the transport of reacting species and ions on the metal surface—a process known to be thermally activated (Fontana and Greene, 1978aa). Further, the NRC staff finds that the Arrhenius relationship is commonly used to characterize the temperature dependence of thermally activated processes (ASM International, 1987aa) and has frequently been used to describe the temperature dependence of the general corrosion rate (e.g., Pensado, et al., 2002aa; Dunn, et al., 2005aa; Lloyd, et al., 2003aa; Hua and Gordon, 2004aa). As such, the NRC staff finds DOE's use of SAR Equation 2.3.6-3 to derive the temperature-dependent general corrosion rate for the waste package outer barrier acceptable.

The NRC staff also reviewed DOE's model assumption that the corrosion rate is constant over time at a given temperature. DOE provided experimental data showing that the measured general corrosion rate of Alloy 22 decreases over time at a given temperature for experiments up to 5 years in duration (SAR Figure 2.3.6-13). This decrease of the corrosion rate with time is also observed in other Alloy 22 corrosion tests (Hua and Gordon, 2004aa; Evans, et al., 2005aa, ab), including independent tests NRC performed (Dunn, et al., 2005aa). The NRC staff also reviewed DOE's alternative conceptual model where the temperature was realistically decreased as a function of time. The NRC staff finds that this alternative model would result in a longer time to waste package failure compared to DOE's primary conceptual model. Therefore, the NRC staff concludes that the use of a constant corrosion rate in the TSPA calculation is acceptable because it would not overestimate the time to waste package breach by general corrosion.

Finally, the NRC staff reviewed DOE's model assumption that microbially influenced corrosion will occur above a relative humidity threshold value. The NRC staff determines that, although adequate water supply may be a critical requirement for microbial growth in the repository, other factors may limit microbial growth even if there is sufficient water. The NRC staff finds that, as detailed in SNL (2004ab, Section 6.4) and Amy, et al. (2002aa), DOE's conceptual model did not take credit for a number of these factors, including

- Waste package temperatures may be too high to support microbial growth
- The seepage water brine's ionic strength may be too high to support microbial growth
- Nutrient supplies may be inadequate to support microbial growth
- The oxic environment in the repository may inhibit MIC caused by sulfate- or nitrate-reducing microbes

The NRC staff finds DOE's use of a threshold relative humidity value for the onset of Microbial Induced Corrosion acceptable because this will not underestimate the probability of MIC.

General Corrosion Rate by Long-Term Weight-Loss Measurements

In SAR Equation 2.3.6-3, DOE established the general corrosion rate at the baseline temperature of 60 °C [140 °F] from 5-year weight-loss experiments (SAR Section 2.3.6.3.2.1). DOE performed corrosion tests in the Long-Term Corrosion Test Facility at 60 and 90 °C [140 and 194 °F], using Alloy 22 specimens with two different geometries: weight-loss specimens and crevice specimens. For both specimen types, tests were performed on specimens with different metallurgical conditions (i.e., mill annealed and as-welded) and in different corrosion test solutions, including simulated acidified water, simulated concentrated water, and simulated dilute water. After 5 years of exposure to the test solutions, every specimen was covered with surface deposits. Therefore, the posttest specimens were cleaned and descaled in accordance with ASTM G 1-90 (ASTM International, 1999aa). DOE stated that the cleaning methods used to remove the scale from the tested samples did not significantly affect untested control samples. Therefore, without correction of any possible mass loss from the replicate untested control foil sample, DOE determined the general corrosion rates of Alloy 22 based upon the formula defined in ASTM G 1-90.

DOE summarized the results of its corrosion tests in SNL (2007al, Section 6.4.3.2). DOE stated that there was no appreciable difference between the general corrosion rates for mill-annealed and as-welded specimens, as shown in SNL (2007al, Figures 6-14 and 6-19). However, the measured corrosion rates for crevice specimens were higher than those for weight-loss specimens, as shown in SNL (2007al, Figure 6-22). For the weight-loss specimens, DOE determined that the mean general corrosion rate was 3.15 nm/yr [1.24×10^{-7} in/yr], with the ±1 standard deviation of 2.71 nm/yr [1.07×10^{-7} in/yr]. For the crevice specimens, DOE calculated a mean general corrosion rate of 7.36 nm/yr [2.90×10^{-7} in/yr] with ±1 standard deviation of 4.93 nm/yr [1.94×10^{-7} in/yr]. Because the crevice specimens tend to give higher corrosion rates, DOE only used the crevice data to develop the distribution from which the 60 °C [140 °F] general corrosion rate parameter was sampled in the TSPA code.

DOE determined that uncertainty and variability in the measured corrosion rate could be attributed both to measurement uncertainty, given the very small weight loss associated with low corrosion rates, and to actual variation in the corrosion processes on the material surface, as outlined in SNL (2007al, Section 6.4.3.3). In the model abstraction, DOE accounted for this uncertainty by fitting the 5-year corrosion data to the Weibull cumulative distribution functions, which are sampled in the TSPA code, as detailed in SNL (2007al, Section 6.4.3.3.2). DOE stated that the Weibull distribution was determined to be the best fit to the experimental data as compared to other fits such as uniform distribution, normal distribution, lognormal distribution, and gamma distribution. DOE characterized the Weibull distribution with two parameters: the scale factor and the shape factor. DOE used three different scale factor/shape factor pairs, corresponding to low, medium, and high uncertainty levels, to define three different Weibull distributions for the 60 °C [140 °F] general corrosion rate parameter, as shown in SNL (2007al, Table 6-7). In the TSPA code, the low, medium, and high general corrosion rate distributions were sampled such that the low and high distributions were each used for 5 percent of the realizations and the medium distribution was used for 90 percent of the realizations. DOE stated that a 5–90–5 percent uncertainty partitioning was selected to ensure the general corrosion rate distributions are separated from each other and yet are sampled enough times to be meaningful (SAR Section 2.3.6.3.3.1). The Weibull distributions from which the 60 °C [140 °F] general corrosion rate of the waste package outer barrier is sampled in the TSPA code are shown in SAR Figure 2.3.6-9.

In response to the NRC staff's request for additional information on the representation of uncertainties associated with cleaning the long-term corrosion specimens, DOE responded in DOE (2009cl, Enclosure 3) that the specimens were not adequately cleaned prior to performing weight-loss measurements. In particular, DOE determined that the initial weight of the specimens was artificially high because of the failure to remove mill-annealed oxide and surface contamination. This oxide and surface contamination, however, were removed during posttest cleaning. Nevertheless, DOE assumed that the associated weight loss was attributable to the general corrosion of Alloy 22. Thus, DOE concluded that it overestimated the actual weight loss of Alloy 22 and, in turn, overestimated the general corrosion rate. DOE stated that the specimens were recleaned and reanalyzed following the procedures in ASTM G 1-03 (ASTM International, 2003ab). DOE adequately provided all data from the reanalysis for the weight-loss specimens because DOE (2009cl, Enclosure 3) showed and stated that those data gave the most accurate estimate for the general corrosion rate of Alloy 22. As shown in DOE (2009cl, Enclosure 3, Figure 8), the corrosion rate from the recleaned weight-loss specimens is close to or lower than that calculated by the three Weibull distributions DOE used in the TSPA code for the Alloy 22 general corrosion rate—particularly for corrosion rates with high cumulative probabilities. DOE stated that only corrosion rates with cumulative probabilities of 0.96 and above {corrosion rate greater than ~15 nm/yr [5.91×10^{-7} in/yr]} are important for waste package performance. Because the data from the recleaned, reanalyzed specimens provide lower corrosion rates than those calculated by the Weibull distributions at the high cumulative probabilities, DOE concluded that use of the Weibull distributions shown in SAR Figure 2.3.6-9 are acceptable because the waste package failure time was not overestimated.

NRC Staff's Review

The NRC staff reviewed DOE's approach to establish the waste package general corrosion rate by long-term tests. As to the material conditions for the corrosion tests, the NRC staff finds that DOE tested Alloy 22 specimens with mill-annealed and as-welded microstructures. The NRC staff finds that DOE acceptably performed tests on materials with these microstructures because they are representative of those expected for the waste package base metal and weld region based on the fabrication procedures set forth in SAR Section 1.5.2.7. As to the test solutions for the corrosion tests, the NRC staff finds that DOE tested Alloy 22 specimens in a range of solutions, including simulated acidified water, simulated concentrated water, and simulated dilute water. The NRC staff concludes that the corrosion rate for the waste package in the repository may depend on such factors as the pH and concentration of ionic species in water that contacts the waste package. Therefore, the NRC staff reviewed the corrosion test solutions to determine whether they are adequate to model the waste package corrosion rate in repository conditions. The NRC staff finds that the corrosion test solutions are more chemically aggressive than waters expected to occur within repository drifts, including starting water compositions in DOE's near-field chemistry model described in SAR Section 2.3.5.5, and waters considered in the NRC staff's independent analysis of in-drift water evolution, described in SER Section 2.2.1.3.3.3.2. Therefore, the NRC staff finds that it is acceptable for DOE to model the waste package corrosion rate on the basis of long-term tests in these simulated brines.

The NRC staff also finds that DOE adequately identified deficiencies with specimen preparation and cleaning that led to unquantified experimental uncertainties in the general corrosion rate for the waste package outer barrier reported in SAR Section 2.3.6.3. Further, the NRC staff finds that the methodology DOE used for recleaning and reanalyzing weight-loss specimens is appropriate for such measurements giving the most accurate estimate and follows the standards specified in ASTM G–1-03 (ASTM International, 2003ab). Therefore, the NRC staff finds that the recleaned, reanalyzed weight-loss data presented in DOE (2009cl, Enclosure 3) are acceptable to represent the general corrosion rate of Alloy 22 at 60 °C [140 °F].

The NRC staff compared the general corrosion rates measured from the recleaned, reanalyzed weight-loss specimens to the general corrosion rates from Weibull distributions calculated in the TSPA code. The NRC staff finds that, at lower cumulative probabilities, the corrosion rate calculated from the recleaned, reanalyzed weight-loss specimens may be slightly higher {by less than approximately 3 nm/yr [1.18×10^{-7} in/yr]} than the rate calculated from the Weibull distributions. However, the NRC staff determines that because the corrosion rates are very low at this cumulative probability {less than approximately 7 nm/yr [2.76×10^{-7} in/yr]}, a difference of less than 3 nm/yr [1.18×10^{-7} in/yr] will have little effect on the waste package performance during the postclosure period. The NRC staff notes that, at higher cumulative probabilities (greater than approximately 0.8), the corrosion rate calculated from the recleaned, reanalyzed weight-loss specimens is less than the corrosion rates calculated from the Weibull distributions, even at the low uncertainty level. The NRC staff finds that the corrosion rates at the high end of the cumulative distribution, where the corrosion rate may be 10 nm/yr [3.94×10^{-7} in/yr] or higher, are most important for waste package performance. For corrosion rates at this level, the waste package may fail by general corrosion breach during the 1-million-year postclosure period or may be susceptible to damage in the event of seismic ground motion. Therefore, the NRC staff concludes that, for corrosion rates most significant to waste package performance, the Weibull distributions DOE used to sample the 60 °C [140 °F] general corrosion rate in the TSPA calculate a higher corrosion rate than the most accurate experimental measures of the general corrosion rate. Hence, the NRC staff finds that sampling from the low, medium, and high from the Weibull distributions in 5–90–5 percent of the realizations, respectively, is an acceptable approach to represent uncertainty. Greater partitioning between the distributions (e.g., 1–98–1 percent) would give too few samples from the low and high distributions to make a statistically meaningful contribution to the cumulative distribution, whereas smaller partitioning between the distributions (e.g., 30–40–30 percent) would not give a statistically meaningful distinction between the respective distributions. On the basis of this information, the NRC staff finds that the Weibull distributions from which DOE sampled the 60 °C [140 °F] general corrosion rate in the TSPA analysis are acceptable because they will not underestimate the general corrosion rates at high cumulative probabilities, which are most important for waste package performance.

Temperature Dependence of the General Corrosion Rate

DOE conducted experiments to determine the temperature dependence of the general corrosion rate of the waste package outer barrier by measuring the activation energy for general corrosion of Alloy 22 (SAR Section 2.3.6.3.2.2). DOE used the short-term electrochemical polarization resistance technique following the ASTM G 59-97 (ASTM International, 1998aa). Mill-annealed and welded specimens were tested in a range of solutions containing NaCl and KNO_3 at temperatures ranging from 60 to 100 °C [140 to 212 °F] (SAR Table 2.3.6-4). DOE used these solutions because they simulate the conditions of moderate relative humidity where calcium is expected to be a minor component in the aqueous environment in the repository, as outlined in SNL (2007al, Section 6.4.3.4).

From these data (SAR Figure 2.3.6-7), DOE used a linear mixed-effects statistical analysis to calculate a mean activation energy of 40.78 kJ/mol [9.74 kcal/mol], with a standard deviation 11.75 kJ/mol [2.81 kcal/mol]. DOE selected a normal distribution to represent the temperature-dependence term on the basis of statistical fitting techniques. The activation energies for the individual solutions used to determine the distribution of the activation energy are shown in SAR Table 2.3.6-5. DOE confirmed the activation energy calculated from these short-term polarization tests by comparisons to the activation energy from the long-term 5-year weight-loss data of Alloy 22 specimens immersed in simulated concentrated water at 60 and 90 °C [140 and 194 °F], respectively, as described in SNL (2007al, Section 6.4.3.4). DOE calculated a mean activation energy of 40.51 kJ/mol [9.68 kcal/mol] for the 5-year corrosion data, which is close to the mean calculated from the short-term polarization technique. From the 5-year corrosion data, the activation energy distribution was also obtained. DOE stated that this distribution was best represented by truncating the normal distribution of the short-term polarization tests at −3 and +2 standard deviations.

The deficiencies with cleaning and weighing Alloy 22 corrosion specimens discussed in DOE (2009cl, Enclosure 3), however, led DOE to reevaluate the calculation of the temperature dependence of the general corrosion rate. DOE stated that the deficiencies were not associated with the short-term polarization data, but rather the comparison of 5-year general corrosion rates for specimens immersed in simulated concentrated water. For the latter, DOE recalculated the activation energy using the corrosion rates measured for the recleaned, reanalyzed weight-loss specimens. From these data, DOE calculated a mean activation energy of approximately 32.26 kJ/mol [7.71 kcal/mol], with minimum and maximum values of 3.37 and 60.05 kJ/mol [0.81 and 14.3 kcal/mol], respectively, as shown in DOE (2009cl, Enclosure 3, Figure 9). These values are approximately 20 percent lower than the activation energies sampled from the truncated normal distribution described in SAR Section 2.3.6.3, which was sampled in the TSPA code. Using both the updated distribution for the activation energy, as shown in DOE (2009cl, Enclosure 3, Figure 9), and the updated distribution for the 60 °C [140 °F] general corrosion rate, as shown in DOE (2009cl, Enclosure 3, Figure 8), DOE calculated the temperature-dependent general corrosion rate of Alloy 22, using SAR Equation 2.3.6-3. DOE compared the corrosion rates calculated using the updated distributions to the corrosion rates calculated using the model described in SAR Section 2.3.6.3. As shown in DOE (2009cl, Enclosure 3, Figures 10–12), for the temperature range of 25 to 200 °C [77 to 392 °F], the updated distributions derived from recleaned, reanalyzed weight-loss specimens give lower corrosion rates than obtained by the model described in SAR Section 2.3.6.3, which was implemented in the TSPA code. As such, DOE concluded that the TSPA code did not underestimate the waste package general corrosion rate.

NRC Staff's Review

The NRC staff reviewed DOE's approach to calculate the temperature dependence of the waste package general corrosion rate. The NRC staff reviewed the material and environmental conditions at which DOE measured the activation energy for general corrosion of the waste package outer barrier that was sampled in the TSPA code. As to the material conditions for the tests, the NRC staff finds that DOE tested Alloy 22 specimens with mill-annealed and as-welded microstructures. The NRC staff finds that it is acceptable for DOE to perform tests on materials with these microstructures because they are representative of those expected for the waste package base metal and weld region, respectively, on the basis of the fabrication procedures set forth in SAR Section 1.5.2.7. For the test solutions for the corrosion tests, the NRC staff concludes that the corrosion behavior of the waste package in the repository depends upon such seepage water characteristics as the pH and concentration of ionic species. The NRC staff concludes that the corrosion rate for the waste package in the repository depends on such factors as the pH and concentration of ionic species in water that contacts the waste package. Therefore, the NRC staff reviewed the corrosion test solutions described in SAR Section 2.3.6.2.2 to determine whether they are adequate to calculate the waste package general corrosion rate activation energy in repository conditions. The NRC staff finds that the corrosion test solutions are more chemically aggressive than waters expected to occur within repository drifts, including starting water compositions in DOE's near-field chemistry model described in SAR Section 2.3.5.5, and waters considered in NRC staff's independent analysis of in-drift water evolution, described in SER Section 2.2.1.3.3.3.2. As such, the NRC staff concludes that it is acceptable for DOE to calculate the activation energy for general corrosion of the waste package on the basis of tests in the brines described in SAR Section 2.3.6.2.2. The NRC staff also reviewed the experimental methodology used to measure the activation energy that was sampled in the TSPA code. The NRC staff finds that the short-term polarization tests used are appropriate to measure the activation energy because the tests conformed with ASTM G59-97 (ASTM International, 1998aa).

Additionally, the NRC staff compared the activation energy for general corrosion of the waste package outer barrier calculated from the recleaned, reanalyzed weight-loss specimens to the activation energy calculated by the short-term polarization test. The NRC staff finds that both calculations, with means of 40.78 kJ/mol [9.74 kcal/mol] and 32.26 kJ/mol [7.71 kcal/mol], respectively, give an activation energy in the range that is independently reported in the technical literature, which is between approximately 25 and 55 kJ/mol [5.97 and 11.9 kcal/mol] (Lloyd, et al., 2003aa; Scully, et al., 2001aa; Hua and Gordon, 2004aa; Smailos and Köster, 1987aa). In particular, the independent testing previously conducted by NRC gives an activation energy of approximately 45 kJ/mol [10.7 kcal/mol] (Dunn, et al., 2005aa; Pensado, et al., 2002aa). This activation energy is close to the mean value of 40.78 kJ/mol [9.74 kcal/mol] used in the TSPA code. Therefore, the NRC staff finds that DOE appropriately identified and analyzed the uncertainties in activation energy introduced by deficiencies in DOE's weighing and measuring the 5-year weight-loss specimens. The NRC staff also finds that the application of the truncated normal distribution in the TSPA code is acceptable to represent the range of the activation energy.

Furthermore, for DOE's calculated temperature-dependent general corrosion rates used in the TSPA model, the NRC staff assessed DOE's calculated corrosion rates by comparing them to the rates reported in independent studies in the technical literature. At 25 °C [77 °F], DOE's model calculates that the corrosion rate is less than 7 nm/yr [2.76×10^{-7} in/yr] (SAR Figure 2.3.6-11). This corrosion rate is lower than the rates measured in other studies (McMillion, et al., 2005aa; Dunn, et al., 2005aa; Jung, et al., 2011aa), which ranged from about

7 to 137 nm/yr [2.76 × 10^{-7} to 5.39 × 10^{-6} in/yr]. However, the NRC staff notes that the data of McMillion, et al. (2005aa) , Dunn, et al. (2005aa), and Jung, et al. (2008aa) are from short-term tests, whereas long-term experimental results show that the general corrosion rate of Alloy 22 decreases by up to two orders of magnitude as experiments progress beyond a few years. Therefore, the NRC staff concludes that the corrosion rates reported by McMillion, et al. (2005aa) and Dunn, et al. (2005aa) were overestimated in terms of long-term general corrosion rate by as much as two orders of magnitude at room temperature. The corrosion rate DOE calculated for 25 °C [77 °F] is not underestimated, and except as noted, is consistent with the corrosion rate in the literature (McMillion, et al., 2005aa; Dunn, et al., 2005aa). The NRC staff finds that DOE's corrosion rates at elevated temperatures of 150 and 200 °C [302 and 392 °F] (SAR Figure 2.3.6-11) are similar to, or greater than, those NRC measured (Yang, et al., 2007aa) for Alloy 22 at the same temperature range. The NRC also finds that these corrosion rates are greater than the corrosion rate of 120 nm/year [4.72 × 10^{-6} in/yr] that Smailos (1993aa) reported for Alloy C-4 (an analogue for Alloy 22) after 6 months' exposure to NaCl-rich brines at 150 °C [302 °F]. On the basis of this information, the NRC staff finds that the distribution from which DOE samples the temperature dependence of the waste package outer barrier general corrosion rate in the TSPA model is acceptable because it is unlikely that DOE underestimated the corrosion rate for the range of temperatures that may be reasonably expected in the repository.

Microbially Influenced Corrosion Effects (MIC)

The NRC staff reviewed DOE's approach to establish the conditions for the onset of MIC and to quantify the extent to which MIC may affect general corrosion behavior.

Threshold Relative Humidity for the Onset of MIC

As discussed in SAR Section 2.3.6.3.3.2 and DOE (2009cl, Enclosure 10), DOE concluded that the relative humidity in the repository must be greater than a threshold value for MIC to increase the general corrosion rate of the waste package outer barrier. DOE used experimental data and information from the independent studies to determine this threshold-relative humidity. DOE performed experiments in which Alloy 22 specimens were embedded in crushed Yucca Mountain tuff and placed in chambers with Yucca Mountain native microorganisms at different temperature and humidity levels (Else, et al., 2003aa). DOE reported that the optimum condition for microbial growth is at a temperature of 30 °C [86 °F] and 100 percent relative humidity. Microbial growth was extremely limited at higher temperatures or lower humidity. In SNL (2004ab, Section 6.4), DOE also cited several studies that indicate that robust growth of most microorganisms requires a relative humidity of 90 percent or higher, although limited growth is seen at relative humidity as low as 75 percent (e.g., Brown, 1976aa; Pedersen and Karlsson, 1995aa). In the TSPA code, DOE accounts for uncertainty in the threshold relative humidity by sampling this threshold relative humidity from a uniform distribution between 75 and 90 percent (SAR Section 2.3.6.3.3.2).

NRC Staff's Review

The NRC staff reviewed DOE's approach to establish the threshold relative humidity for the onset of MIC. The NRC staff finds that DOE appropriately used experimental data and information from independent studies to establish the threshold relative humidity for microbial growth in conditions of crushed Yucca Mountain tuff with microorganisms. Moreover, the NRC staff also finds that DOE did not credit additional factors that may preclude microbial growth in the repository even if the relative humidity exceeded the threshold value. Therefore, the NRC staff finds that DOE's distribution for the threshold relative humidity is acceptable because it will not underestimate the probability of the onset of MIC.

Microbially Influenced Corrosion (MIC) Factor

If the relative humidity at the waste package surface is greater than the threshold value, the MIC-enhanced general corrosion rate for the waste package outer barrier is calculated by multiplying the general corrosion rate in the absence of the microorganisms (given by SAR Equation 2.3.6-3) by the MIC enhancement factor. DOE performed laboratory tests to determine the extent to which MIC may affect the general corrosion rate of Alloy 22 (SAR Section 2.3.6.3.2.3). DOE used the electrochemical polarization technique to measure the corrosion rate of Alloy 22 specimens in nutrient-enriched, simulated Yucca Mountain well water, with and without the presence of microbes (Lian, et al., 1999aa). Test results are shown in SNL (2007al, Table 6-16 and Figure 6-54). DOE found that the general corrosion rate for Alloy 22 in the microbial-rich solution was up to a factor of approximately two higher than the general corrosion rate in sterile solution. DOE represented epistemic uncertainty in the MIC enhancement factor to account for natural variation in the expected extent of microbial activity in repository conditions. Thus, in the TSPA code, DOE sampled the MIC enhancement factor from a uniform distribution between one and two, to account for the corrosion rate variability due to the effect of MIC. An enhancement factor value of one represents no enhancement, and an enhancement factor value of two represents maximum enhancement.

NRC Staff's Review

The NRC staff reviewed DOE's approach to calculate the MIC enhancement factor. The NRC staff finds that the corrosion tests DOE performed to measure the MIC enhancement factor were performed in conditions that would support the growth of microbes. Further, the NRC staff determined that the polarization tests used to measure the MIC effect conformed to ASTM G–59-97 (ASTM International, 1998aa), which is appropriate for such measurements. The NRC staff finds acceptable that DOE did not use sterile conditions in the long-term (5 years) corrosion tests used to determine the nominal general corrosion rate for the waste package outer barrier, because sterile conditions do not represent realistic Yucca Mountain conditions. DOE indicated that some samples from these tests contain a significant amount of microbial bacteria, even though no bacteria were deliberately introduced (Horn, et al., 2005aa). Therefore, the NRC staff finds that DOE has selected an acceptable range to represent the MIC enhancement factor in the TSPA analysis because it is unlikely to underestimate the extent to which Yucca Mountain microorganisms may increase the general corrosion rate of the waste package outer barrier.

Abstraction and Integration

The model abstraction for general corrosion of the waste package outer barrier is implemented in the Waste Package and Drip Shield Degradation Submodel in the TSPA code in SNL (2008ag, Section 6.3.5). The inputs that are needed for the model abstraction are temperature of the waste package and the relative humidity in the drift. These inputs are provided in the Engineered Barrier System Thermohydrologic Environment Submodel. The Waste Package and Drip Shield Degradation Submodel includes both the CSNF configuration that uses the transportation, aging, and disposal canister configuration parameters and the codisposal waste package configuration that uses the 5 high-level waste/1 DOE spent nuclear fuel long configuration parameters. DOE assumed a total of 11,629 waste packages divided into

5 percolation subregions, each of which is subject to different environmental conditions, as detailed in SNL (2008ag, Section 6.3.5.1.3). The Waste Package and Drip Shield Degradation Submodel general corrosion calculations are performed for both waste package configurations in each of the percolation subregions.

In the Waste Package and Drip Shield Degradation Submodel, the waste package surface is divided into subareas, referred to as patches, to account for the spatial variability of general corrosion on the waste package surface, as outlined in SNL (2008ag, Section 6.3.5.1.2). Each patch may have a different general corrosion rate. The submodel uses a patch area of 231.5 cm^2 [35.88 in^2], therefore the CSNF and the codisposal waste packages have 1,430 and 1,408 patches, respectively. For each realization, each patch is assigned a different value for the 60 °C [140 °F] general corrosion rate, which is sampled from the Weibull distributions derived from 5-year weight-loss corrosion data from creviced Alloy 22 specimens (SAR Section 2.3.6.3.3.1). Because the size of the crevice specimens that DOE used to measure the 5-year general corrosion was about one-fourth the patch size, DOE sampled the 60 °C [140 °F] general corrosion rate four times for each patch and applied the highest of the four sampled rates to the patch, as described in SNL (2008ag, Section 6.2.5.1.2). The effect of rescaling the 60 °C [140 °F] general corrosion rate distribution, shown in SNL (2008ag, Figure 6.3.5-6), resulted in rates that are approximately twice those of the nominal distribution. To account for the temperature dependence of the general corrosion rate, a single value of the temperature-dependent parameter is sampled in each realization from the distribution derived from short-term polarization tests and applied to all waste package patches. To account for potential MIC, the value of the threshold relative humidity for MIC is sampled once per realization from a uniform distribution in the range of 75 to 90 percent and applied to all waste packages. If the relative humidity in the drift exceeds the threshold, the MIC enhancement factor is sampled from a uniform distribution in the range of one to two and applied to the patches.

DOE considered a waste package outer barrier to be breached by general corrosion when one or more patches are penetrated. When the waste package was breached, the general corrosion model was also applied to the inner surface of the waste package outer barrier. The output for the general corrosion model gave the percentage of breached waste packages as a function of time and the average number of patch penetrations per breached waste package as a function of time. This output was transferred to and used in the waste form degradation, the mobilization model component, and the engineered barrier system flow and engineered barrier system transport submodels. SAR Figures 2.1-10(b) and 2.1-16(b) showed the fraction of CSNF waste packages breached by general corrosion and the fraction of the waste package surface area breached per breached waste package, respectively, for the CSNF waste package in the Nominal Modeling Case.

A mean of less than 10 percent of CSNF waste packages are breached over 1 million years, and of the breached waste packages, the mean breached area is less than 0.3 percent of the total waste package surface area. The results for the codisposal waste package in the Nominal Modeling Case are similar (SAR Figure 2.1-17[b]). DOE (2009bj, Enclosure 1, Figures 9 and 10) showed the fraction of CSNF and codisposal waste packages, respectively, breached by general corrosion in the Seismic Ground Motion Modeling Case. For both waste packages, the mean is approximately 10 percent breached in 1 million years. DOE Enclosure 1, Figures 11 and 12 of its response to the NRC staff's request for additional information (DOE, 2009bj) showed the fraction of the surface area breached for the CSNF and the codisposal waste packages breached by general corrosion in the Seismic Ground Motion Modeling Case. For both waste packages, the fraction is approximately 1 percent of the surface area.

As to the activation energy for general corrosion in the TSPA model, DOE performed sensitivity analyses, which show that the expected dose has a strong correlation to the activation energy for general corrosion of the waste package outer barrier and tends to increase with decreasing activation energy for general corrosion (SAR Figures 2.4-151 and 2.4-155).

NRC Staff's Review

The NRC staff reviewed the implementation and integration of the model abstraction for general corrosion of the waste package outer barrier in the postclosure performance assessment. The NRC staff finds that DOE has provided sufficient information for the NRC staff to understand how the conceptual model is implemented in the TSPA code and how the model inputs and outputs are integrated with other model components. The NRC staff determined that the model abstraction is consistent with the design features of the waste package, including materials of construction and dimensions given in SAR Section 1.5.2.7. Further, the NRC staff finds that, with respect to the general corrosion rates, DOE adequately justified the data and model used because DOE showed the ranges of these parameters and accounted for uncertainty in the model abstraction, as summarized in Section 2.2.1.3.1.3.2.1. Moreover, the NRC staff finds that DOE appropriately accounted for spatial variability in the general corrosion rate on the waste package surface by applying the corrosion rate on a patch scale and by rescaling the corrosion rate to account for the difference between the patch size and size of the corrosion specimens. The NRC staff performed independent calculations to confirm the waste package and Drip Shield Degradation Submodel in the TSPA (Jung, 2010aa). The NRC staff's calculations, with respect to the timing and magnitude of waste package breach by general corrosion, were consistent with DOE's calculations. Therefore, the NRC staff finds acceptable DOE's implementation of the waste package general corrosion model abstraction in the TSPA code because it would not overestimate the timing or underestimate the magnitude of radionuclide release to the accessible environment.

NRC Summary of Evaluation Findings for the General Corrosion of the Waste Package Outer Barrier

The NRC staff reviewed DOE's model abstraction for general corrosion of the waste package outer barrier that was implemented in the TSPA code. On the basis of its review, the NRC staff finds that DOE used appropriate experimental tests and other independent technical literature to provide adequate support for the model abstraction. In addition, DOE appropriately identified and adequately considered features and processes such as spatial variability, temperature and MIC effects, and general corrosion rates of the waste package outer barrier that affect the barrier capabilities for the initial 10,000 year period, and projected these features and processes beyond the 10,000 year post-disposal period through the period of geologic stability. Therefore, the NRC staff finds that DOE acceptably accounted for general corrosion of the waste package outer barrier in the TSPA model.

2.2.1.3.1.3.2.2 Localized Corrosion of the Waste Package Outer Barrier

Localized corrosion is a process where corrosion occurs at discrete sites, in contrast to general corrosion, which uniformly thins the entire surface of a material. Localized corrosion usually occurs in metals and alloys, such as Alloy 22, whose corrosion resistance is attributed to the presence of a passive oxide film. Localized corrosion can initiate if the passive film is removed or damaged. When localized corrosion does occur, it tends to cause degradation much faster than general corrosion. In SAR Section 2.3.6.4, DOE considered that localized corrosion could lead to the release of radionuclides from the waste package if the waste package wall is breached.

DOE determined that localized corrosion requires the presence of a liquid water film on the waste package surface, which may come from dripping seepage water or salt deliquescence in dust particles, as outlined in SNL (2008ag, Section 6.3.5.2). DOE's evaluation of salt deliquescence indicated that brines produced from dust deposits will not lead to localized corrosion (FEP 2.1.09.28.0A; SNL, 2008ac). Consequently, DOE excluded localized corrosion caused by deliquescence from the performance assessment (SAR Table 2.2-5) and concluded that seepage water must contact the waste package for localized corrosion to occur.

In the TSPA code, DOE calculated that in-drift conditions (i.e., temperature, pH, and concentration of ionic species in seepage water) may support localized corrosion of the waste package for approximately 12,000 years after repository closure, as described in DOE (2009dg, Enclosure 1). The TSPA code, however, also calculates that few drip shields fail within 12,000 years after repository closure. Therefore, the probability of waste package breach by localized corrosion is low in DOE's model. Following 12,000 years after repository closure, DOE calculated that there is a low probability for conditions in the drift to support localized corrosion of the waste package even if the drip shield fails and allows seepage water to contact the waste package.

In SER Sections 2.2.1.3.1.3.1 and 2.2.1.3.2.6, respectively, the NRC staff finds that chemical and mechanical degradation of the drip shield are very unlikely to cause failure of the drip shield plates and allow seepage water to contact the waste package within 12,000 years of repository closure. Therefore, the NRC staff finds that there is a low probability of waste package breaches by localized corrosion within 12,000 years of repository closure. Nevertheless, there are uncertainties related to the barrier capability of the drip shield beyond 12,000 years after repository closure, and seepage water may contact the waste package during this period. As such, this section provides the NRC staff's review of DOE's model abstractions for initiation and propagation of localized corrosion, focusing on localized corrosion behavior beyond 12,000 years after repository closure.

Waste Package Localized Corrosion Conceptual Models

DOE implemented models for both initiation and propagation of localized corrosion of the waste package outer barrier in the TSPA code. The following addresses the NRC staff's review of these models.

Corrosion Initiation Models

DOE considered two potential mechanisms by which localized corrosion could be initiated on the waste package outer barrier under seepage conditions. In SAR Section 2.3.6.4, DOE described the first mechanism, which is related to the waste package open-circuit corrosion potential, or corrosion potential. DOE's model initiates localized corrosion if the corrosion potential for the waste package is greater than or equal to a critical potential. DOE defined critical potential as the potential above which a passive film will not spontaneously reform if damaged (SAR Section 2.3.6.4.1). While localized corrosion typically encompasses both pitting and crevice corrosion, DOE treated all waste package localized corrosion as crevice corrosion because crevice corrosion initiates in less aggressive thermal and chemical conditions than pitting corrosion, as described in SNL (2007al, Section 6.4.4). As such, DOE assumed that the critical potential for localized corrosion is equivalent to the crevice repassivation potential (SAR Section 2.3.6 4.1).

The second initiation mechanism, referred to as salt separation, is described in SAR Section 2.3.5.5.4.2.1. During salt separation, the relative humidity at the waste package surface drops below a salt precipitation threshold while seepage is occurring, causing chloride salts to precipitate out of the solution. Nitrate that is still in the solution moves away by advection. A chemically aggressive chloride-rich, nitrate-depleted brine forms when the relative humidity increases above this threshold value. DOE did not, however, model localized corrosion by salt separation in the TSPA code, as summarized in SAR Section 2.4. Consequently, in a request for additional information, the NRC staff requested that DOE provide technical details evaluating the significance of salt separation effects on the performance assessment of waste packages in the proposed repository environment. DOE (2009dg, Enclosure 1) provided additional information indicating that the salt separation aspects of localized corrosion initiation were implemented as described in SNL (2008ag, Appendix O). DOE further stated that the information and analysis provided in DOE (2009dg, Enclosure 1) (i.e., that the salt separation aspects of localized corrosion initiation were not implemented) will be included in a future license application update.

NRC Staff's Review

The NRC staff reviewed DOE's conceptual models for localized corrosion initiation on the waste package outer barrier, as described in SAR Sections 2.3.6.4 and 2.3.5.5. The NRC staff finds that DOE's model assumption that localized corrosion will initiate when the corrosion potential for the waste package is greater than or equal to a critical potential (i.e., the crevice repassivation potential) is consistent with the understanding of corrosion processes in the technical literature (Evans, et al., 2005aa,ab; Dunn, et al., 2000aa). Moreover, the NRC staff finds that this initiation model will not underestimate the probability of localized corrosion initiation because the initiation of Alloy 22 crevice corrosion generally requires less aggressive conditions than the initiation of pitting corrosion (Rebak, 2005aa; Cragnolino, et al., 1999aa; Dunn, et al., 2000aa). Although micropits were observed from dripping seepage water (Jung, et al., 2011aa; Ashida, et al., 2008aa), there is no evidence for pit propagation. For initiation by salt separation, the NRC staff finds that DOE's model of chloride salt precipitation from brines in low humidity conditions is consistent with the thermodynamic physics of salt solutions (Yang, et al., 2006aa). Also, the NRC staff finds that the relative humidity threshold in the range of approximately 65 to 77 percent in DOE's model of salt separation will not underestimate the probability of localized corrosion initiation because the sodium chloride salt with deliquescence and efflorescence relative humidity in this range is less aggressive than the magnesium and calcium chloride salts with lower relative humidities.

The NRC staff also finds that crevice corrosion is typically associated with small volumes of stagnant solution in holes, under surface deposits, or underneath fasteners (Fontana and Greene, 1978aa). The NRC staff finds that experimental data indicate that some crevice couples that could form in the repository do not support localized corrosion even if they form a tight crevice (He, et al., 2007ab; Shan and Payer, 2007aa). Alloy 22-to-Alloy 22, Alloy 22-to-titanium, and Alloy 22-to-ceramic couples have low susceptibility to crevice corrosion in concentrated sodium chloride solutions, which represents the nitrate-depleted brine that forms after salt separation. The NRC staff finds that in DOE's model, localized corrosion could occur on any part of the waste package surface exposed to seepage water. Therefore, the model will not underestimate the fraction of the waste package surface that undergoes localized corrosion.

Corrosion Propagation Model

In DOE's model for localized corrosion propagation, corrosion propagates at a constant rate over time (SAR Section 2.3.6.4.3.2). DOE also considered an alternative conceptual model in which the corrosion rate decreased over time (SAR Section 2.3.6.4.3.2.2). The alternative model, based on pit growth, gives a corrosion propagation rate lower than that calculated using the primary model assumption of constant corrosion rate. Therefore, DOE implemented the model with constant corrosion rate in the TSPA code because it calculated an earlier waste package breach time.

NRC Staff's Review

The NRC staff reviewed DOE's model for propagation of localized corrosion of the waste package outer barrier, as described in SAR Section 2.3.6.4.3.2. The NRC staff finds that DOE presented experimental evidence for a decreasing localized corrosion rate over time for a range of metals and alloys (Hunkeler and Boehni, 1983aa; Marsh, et al., 1991aa; Mughabghab and Sullivan, 1989aa; Sharland, et al., 1994aa; Ishikawa, et al., 1994aa). Also, the NRC staff concludes that localized corrosion kinetics in Alloy 22 is likely to be slower than these materials because of its persistent passive film. The NRC staff finds acceptable DOE's use of the assumption that the localized corrosion rate over time for the waste package outer barrier is constant because it will not underestimate the corrosion propagation rate.

Localized Corrosion Initiation Conditions

For localized corrosion initiation by corrosion potential and salt separation, DOE used electrochemical experimental data to establish the conditions in which localized corrosion of the waste package outer barrier could initiate. The NRC staff reviewed DOE's localized corrosion initiation conditions.

Initiation by Critical Potential

In the TSPA code, DOE compared the waste package outer barrier corrosion potential to the repassivation potential to determine whether electrochemical conditions on the waste package would lead to passive film breakdown. DOE assumed that the corrosion potential and repassivation potential for the waste package outer barrier, respectively, depended on the environmental conditions in the drift, including temperature, pH, chloride concentration, and nitrate concentration. DOE derived equations to represent the potentials as functions of these parameters by performing tests in which it measured the potentials for Alloy 22 specimens while varying the environmental parameters (SAR Section 2.3.6.4.2).

DOE established the dependence between the corrosion potential of Alloy 22 and the environmental parameters using data from 5-year tests (SAR Section 2.3.6.4.2.1). For a range of temperatures, DOE measured the corrosion potentials for Alloy 22 creviced samples with various metallurgical conditions (i.e., as welded, mill annealed, stress relieved) exposed to a range of simulated brine solutions, including simulated dilute water, simulated acidified water, and simulated concentrated water, the compositions of which were given in SAR Table 2.3.6-1. The corrosion potential data used in DOE's model development were shown in SAR Table 2.3.6-6. Using regression analyses, DOE applied SAR Equation 2.3.6-7 to the data shown in SAR Table 2.3.6-6, representing the corrosion potential as a function of temperature, pH, and nitrate and chloride ion concentrations.

DOE established the dependence between the repassivation potential of Alloy 22 and the environmental parameters using data from cyclic potentiodynamic polarization tests (SAR Section 2.3.6.4.2.2). The tests were performed using the methodology of ASTM G–61-86 (ASTM International, 2003aa). In DOE (2009cl, Enclosure 9), DOE stated that while repassivation potentials can be measured by methods other than cyclic potentiodynamic polarization, the differences in measured potentials tended to be small, and the cyclic potentiodynamic polarization method generally predicted greater corrosion susceptibility in aggressive brines. Similar to DOE's tests for corrosion potential, DOE used Alloy 22 specimens with different metallurgical conditions (mill annealed and as welded) in a range of simulated brine solutions. DOE used ceramic wrapped with polytetrafluoroethylene tape to form a crevice with Alloy 22 in the cyclic potentiodynamic polarization tests. The crevice repassivation potential data used in DOE's model development were shown in SAR Table 2.3.6-7. Sample specimens that did not show evidence of localized corrosion attack were not used to develop the model. For experiments showing localized corrosion, DOE used regression analyses to fit SAR Equation 2.3.6-6 to the data shown in SAR Table 2.3.6-7, representing the crevice repassivation potential as a function of temperature and nitrate and chloride ion concentrations. In the TSPA code, DOE accounted for fitting uncertainty in SAR Equations 2.3.6-6 and 2.3.6-7 by varying the values of the fitting parameters of the respective equations according to a Monte Carlo algorithm and by using data in the regression analysis from multiple samples for a given environmental condition, as detailed in SNL (2008ag, Section 6.3.5.2.2).

DOE compared the model calculations of corrosion potential and repassivation potential, respectively, to establish the environmental conditions that would support initiating localized corrosion on the waste package outer barrier (SAR Section 2.3.6.4.3.1.2). In DOE's model, the probability for initiation of localized corrosion at temperatures less than 90 °C [194 °F] generally increases with decreasing pH and decreasing nitrate-to-chloride ion ratio. DOE compared the localized corrosion initiation conditions to experimental observations of localized corrosion initiation on Alloy 22 specimens. DOE's model calculated that localized corrosion may initiate on Alloy 22 exposed to simulated acidified water at 90 °C [194 °F], whereas in experimental tests, localized corrosion was not observed on Alloy 22 specimens exposed to this solution for 5 years (SAR Table 2.3.6-12). This conclusion is supported by a number of additional corrosion test solutions (SAR Table 2.3.6-13). Therefore, DOE concluded that its model overestimates the probability of localized corrosion initiation.

NRC Staff's Review

The NRC staff reviewed DOE's experimental methodology to measure the corrosion potential and the crevice repassivation potential of Alloy 22. The NRC staff finds that DOE's tests were consistent with the general practice for measuring the corrosion potential and repassivation potential of metallic materials. In particular, the NRC staff finds that the cyclic potentiodynamic polarization method in electrochemistry is an acceptable way to measure the repassivation potential because it predicts higher corrosion susceptibility than other measures of this parameter, as confirmed by independent analyses by the NRC staff (He, et al., 2009aa). Also, the NRC independent analyses (He, et al., 2007aa) and Shan and Payer (2007aa), show that ceramic wrapped with polytetrafluoroethylene tape, which DOE used to form a crevice with Alloy 22 in the cyclic potentiodynamic polarization tests, is a material combination that is more favorable for localized corrosion than many combinations that would be expected in the repository, including Alloy 22-to-Alloy 22, Alloy 22-to-titanium, and Alloy 22-to-ceramic couples.

The NRC staff reviewed DOE's calculated values for the corrosion potential and repassivation potential, respectively, to confirm the environmental conditions (i.e., temperature, pH, concentration of ionic species in seepage water) under which DOE's model calculates that localized corrosion of the waste package will initiate. The NRC staff examined the model functional dependencies and confirmed that, in DOE's model, the probability for localized corrosion initiation at a given temperature generally increases with decreasing pH and decreasing nitrate-to-chloride ion ratio. Therefore, the NRC staff finds that DOE's model may calculate that localized corrosion initiates in acidic solutions such as simulated acidified water or in solutions with very low nitrate-to-chloride ratio (SAR Tables 2.3.6-12 and 2.3.6-13). Nevertheless, the NRC staff concludes that Alloy 22 specimens did not show evidence of localized corrosion initiation during 5 years of immersion in a range of corrosion test solutions, including simulated acidified water, at temperatures up to 90 °C [194 °F] (SAR Table 2.3.6-12) and that for shorter tests localized corrosion was only observed in concentrated chloride brines (SAR Table 2.3.6-13). On the basis of these data, the NRC staff finds that Alloy 22 is resistant to localized corrosion in brine solutions that are more chemically aggressive than the waters expected to occur within the repository drifts, including starting water compositions in DOE's near-field chemistry model described in SAR Section 2.3.5.5, and waters considered in the NRC staff's independent analysis of in-drift water evolution described in SER Section 2.2.1.3.3.3.2. Therefore, the NRC staff finds that DOE's model is acceptable because it calculates a higher probability of localized corrosion initiation than is expected based on experimental data provided by DOE.

Initiation by Salt Separation

DOE used thermodynamic analyses to calculate the threshold relative humidity below which chloride-bearing salts could precipitate out of seepage water (SNL, 2007ak). In these analyses, DOE considered that the threshold relative humidity depends upon the group water type (i.e., 1–4 as defined in SAR Section 2.3.5), quantity of alkali feldspar to be titrated into the seepage waters [i.e., the water–rock interaction parameter (WRIP)], the partial pressure of CO_2 in the drift, and the waste package temperature. In SAR Figure 2.3.5-55, the thermodynamic analyses showed that the chloride-to-nitrate ratio in a range of conditions is nearly constant until the activity of water (i.e., relative humidity) drops below a value in the range of approximately 65 to 77 percent. In the TSPA model, DOE concluded that this range represents the threshold relative humidity below which localized corrosion initiation by salt separation can occur (SAR Section 2.3.5.5).

NRC Staff's Review

The NRC staff reviewed DOE's approach for establishing the relative humidity threshold for the initiation of localized corrosion by salt separation, as DOE described in SAR Section 2.3.5.5. The NRC staff finds that the methodology DOE used to determine the threshold relative humidity is acceptable because it is based on well-established concepts regarding the thermodynamic stability of aqueous solutions. In particular, the threshold values DOE calculated are consistent with calculations for pure sodium chloride by Greenspan (1977aa), which are incorporated into ASTM E104-02 (ASTM International, 2007aa). Also, the NRC staff performed independent tests and observed that even in pure 5 M sodium chloride solution, localized corrosion was not initiated on Alloy 22 specimens in open circuit conditions without the addition of copper chloride as an oxidant (He and Dunn, 2005aa). DOE's model assumes that salt separation can cause the corrosion potential to exceed the repassivation potential. In experimental tests, however, DOE did not observe localized corrosion in many cases where the corrosion potential was greater than the repassivation potential, including 5-year tests in simulated acidified water (SAR Tables 2.3.6-12 and 2.3.6-13). On the basis of this information, the NRC staff finds that the threshold relative humidity that DOE calculated is acceptable because it will not underestimate the probability of localized corrosion initiation.

Localized Corrosion Propagation Rate

In the TSPA code, DOE sampled the propagation rate for localized corrosion on the waste package outer barrier using a log-uniform distribution in the range of 12.7 μm/yr to 1,270 μm/yr [5×10^{-4} to 5×10^{-2} in/yr] with a median value of 127 μm/yr [5×10^{-3} in/yr] (SAR Section 2.3.6.4.3.2). This range was based on corrosion testing of Alloy 22 in aggressive environments, including 10 percent $FeCl_3$ (Haynes International, 1997aa) and concentrated HCl (Haynes International, 1997ab). DOE compared the corrosion rate distribution sampled in the TSPA code to independently measured corrosion rates for similar, but less corrosion-resistant alloys, including Alloy C-276 and Alloy C-4 (SAR Section 2.3.6.4.4.2.2). DOE concluded that the measured corrosion rates fall within the bounds of the distribution sampled in the TSPA code.

NRC Staff's Review

The NRC staff reviewed DOE's approach, described in SAR Section 2.3.6.4, that established the distribution sampled in the TSPA code for the propagation rate of localized corrosion on the waste package outer barrier. The NRC staff determines that the distribution sampled in the TSPA code is based on corrosion data from studies that considered thermal and chemical conditions that were more aggressive than those expected in the repository. Also, DOE's calculated range for the corrosion rate is consistent with the NRC staff's independent measurements of the Alloy 22 localized corrosion rate (He and Dunn, 2005aa). On the basis of this information, the NRC staff finds that the distribution from which DOE sampled the waste package outer barrier localized corrosion propagation rate in the TSPA code is acceptable because it will not underestimate the propagation rate. Localized corrosion at the rates DOE calculated would penetrate the 25-mm [0.98-in] thick waste package outer barrier in approximately 20 to 2,000 years, which is short relative to the repository performance period. Therefore, the NRC staff finds that parameter values, assumed ranges, probability distributions, and bounding assumptions in the localized corrosion rate would not significantly affect the timing or magnitude of radionuclide release.

Effects of Microorganisms on Localized Corrosion

As DOE addressed in SAR Section 2.3.6.3.2.3, microorganisms in the repository may affect the corrosion processes on the waste package outer barrier such that the type and extent of corrosion in the presence of the microorganisms may be different from the corrosion in the absence of the microorganisms. Although DOE incorporated MIC effects into its model abstraction for general corrosion of the waste package outer barrier (SAR Section 2.3.6.3.2.3), experimental observations indicate that MIC may also affect the localized corrosion behavior. In particular, a DOE study showed that small pits, or micropores, were observed on the surface of Alloy 22 corrosion specimens exposed in a borosilicate glass vessel with unsterilized Yucca Mountain tuff rock, whereas no such micropores were observed in sterilized conditions (Martin, et al., 2004aa). In DOE (2009cl, Enclosure 10), DOE stated that the micropores that Martin, et al. (2004aa) observed started to form during the first 17 months of exposure, after which the size of the micropores was less than 1 μm [0.039 mil] in diameter. DOE further stated that the same specimens were observed after an additional 40 months exposure, after which there were more pores, but no significant increase in pore size compared to that measured after 17 months. DOE determined that, if the micropores were initiated pits, they quickly repassivated before they could propagate. Therefore, DOE concluded that it was appropriate to incorporate MIC into its model abstraction for general corrosion of the waste package outer barrier, to be consistent with DOE observations in SNL (2007al, Section 6.4.5) that MIC may enhance corrosion on the entire material surface.

NRC Staff's Review

The NRC staff reviewed the effects of microbes on Alloy 22 corrosion described in DOE (2009cl, Enclosure 10). The NRC staff finds acceptable DOE's analysis that the pores may represent initiated pits that quickly repassivated. Because the pores did not grow after repassivation, the NRC staff finds that the micropores are not indicative of localized corrosion that could significantly affect the timing or magnitude of radionuclide release from the waste package. The NRC staff also identified an independent report, which confirmed that Alloy 22 is highly resistant to localized corrosion in microorganism-rich environments, including seawater, which also has low nitrate content (Aylor, et al., 1999aa). On the basis of the NRC staff's review of DOE's evaluation and literature data, the NRC staff finds acceptable DOE's exclusion of MIC effects in the model abstraction for localized corrosion of the waste package outer barrier.

Abstraction and Integration

DOE did not directly include and calculate the effects of localized corrosion of the waste package outer barrier in the TSPA code. Rather, DOE performed a Localized Corrosion Initiation Uncertainty Analysis, as described in SNL (2008ag, Appendix O) and DOE (2009dg, Enclosure 1), that calculated the fraction of waste packages in the repository that are susceptible to localized corrosion as a function of drip shield breach time (i.e., the time at which seepage water could contact the waste package). The Localized Corrosion Initiation Uncertainty Analysis implements the Localized Corrosion Initiation Submodel, as detailed in SNL (2008ag, Section 6.3.5.2.3), which determines whether the environmental conditions in the drift will initiate localized corrosion, and gives this input to the TSPA code. The Localized Corrosion Initiation Submodel is similar to the TSPA code, but it incorporates only those submodels that are needed to calculate localized corrosion initiation conditions. In particular, the Localized Corrosion Initiation Submodel uses information primarily from the

- Engineered Barrier System Thermohydrologic Environment Submodel to determine the temperature and relative humidity history at the waste package

- Drift Seepage Submodel to determine whether seepage occurs at a repository location

- Engineered Barrier System Chemical Environment Submodel to determine the chemical composition of seepage water

- Seismic Ground Motion Damage Submodel to determine the time of drip shield plate failure due to seismic damage

- Waste Package and Drip Shield Degradation Submodel to account for corrosion in determining drip shield plate and waste package failure times

In the Localized Corrosion Initiation Submodel, the repository was discretized into 3,264 subdomains of equal area, at the center of which were 6 CSNF and 2 codisposal waste packages. The subdomains were distributed through the five percolation subregions. In the

Localized Corrosion Initiation Submodel, localized corrosion could initiate because of the waste package corrosion potential or by salt separation. For each subregion, at every timestep in a realization, the Localized Corrosion Initiation Submodel compares the corrosion potential, as calculated by SAR Equation 2.3.6-7, to the crevice repassivation potential, as calculated by SAR Equation 2.3.6-6. If the corrosion potential was greater than or equal to the crevice repassivation potential in seepage conditions, the Localized Corrosion Initiation Submodel assumed that localized corrosion initiated at that subregion. Similarly, if the relative humidity at the waste package surface fell below the salt separation threshold humidity in seepage conditions, the Localized Corrosion Initiation Submodel assumed that localized corrosion initiated at that subregion.

The Localized Corrosion Initiation Submodel calculated that, if the drip shield were breached at the time of the repository closure (i.e., no drip shield present), there is approximately 34 percent probability that localized corrosion will initiate on a given waste package surface (24 percent probability contribution by salt separation and 10 percent probability contribution by corrosion potential), as shown in DOE (2009dg, Enclosure 1, Figure 1). DOE calculated that as the time to drip shield breach increases, the probability of localized corrosion initiation decreases and is negligible if the drip shield fails beyond approximately 12,000 years after repository closure. In particular, DOE calculated that localized corrosion will not initiate by salt separation if drip shield breach occurs after approximately 1,000 years from the time of repository closure, because the relative humidity will remain above the threshold value. DOE also stated that changes in the repository environmental and chemical conditions (e.g., decreasing temperature) make initiation by corrosion potential less probable as the time to drip shield breach increases in its model.

Given the results of the Localized Corrosion Initiation Uncertainty Analysis, DOE concluded that localized corrosion of the waste package outer barrier could affect the timing and magnitude of the release of radionuclides from the waste package only if the overlying drip shield plate was breached within approximately 12,000 years after repository closure. In regard to drip shield early failure, DOE assumed that localized corrosion under seepage conditions occurs on the waste packages located beneath the failed drip shield. Other than drip shield early failure, DOE modeled that no drip shield failure would occur within approximately 12,000 years. Therefore, as shown in DOE (2009dg, Enclosure 1), DOE concluded that localized corrosion of the waste package outer barrier will have no significant effect upon either the timing or magnitude of radionuclide release calculated in the TSPA modeling cases.

NRC Staff's Review

The NRC staff reviewed the implementation and integration of the model abstraction for localized corrosion of the waste package outer barrier in the postclosure performance assessment. The NRC staff finds that DOE has provided sufficient information for the NRC staff to understand how the conceptual model is implemented in the TSPA code and how the model inputs and outputs are integrated with other model components. The NRC staff determines that the model abstraction is consistent with the design features of the waste package, including materials of construction and dimensions given in SAR Section 1.5.2.7. Further, the NRC staff finds that, with respect to the parameters used in the model abstraction (i.e., corrosion potential, corrosion repassivation potential, relative humidity, and pit growth rate), DOE adequately justified the data and model used because DOE showed the ranges of these parameters and accounted for uncertainty in the model abstraction.

The NRC staff finds acceptable DOE's analysis that the consequence of drip shield breach within 12,000 years on the overall radiological dose is negligible because it does not underestimate the onset of seismic-induced breach. The NRC staff also concludes that the probability of waste package localized corrosion initiation beyond 12,000 years after repository closure is low, even if drip shield failure allows seepage water to contact the waste package. In the Localized Corrosion Initiation Uncertainty Analysis, the probability for localized corrosion initiation decreases with increasing pH and increasing nitrate-to-chloride ion ratio. DOE demonstrated that the pH and nitrate-to-chloride ion ratio of in-drift waters will generally be too high to initiate localized corrosion beyond 12,000 years after repository closure. DOE's model also demonstrated that localized corrosion may initiate in low pH solutions or solutions with low nitrate-to-chloride ratio. DOE's experimental data, however, showed that localized corrosion does not initiate at 90 °C [194 °F], even in corrosion test solutions with lower pH or lower nitrate-to-chloride ratio than waters expected to be present in drifts at such temperatures, including starting seepage waters used by DOE (SAR Section 2.3.5.5), and waters considered in the NRC staff's independent analysis of in-drift water evolution (SER Section 2.2.1.3.3.3.2). As such, the NRC staff finds that waste package breach by localized corrosion is unlikely beyond 12,000 years after repository closure, even if drip shield failure allows seepage water to contact the waste package. The NRC staff concludes that the implementation of the TSPA model abstraction for localized corrosion of the waste package outer barrier is acceptable because it would not overestimate the timing or underestimate the magnitude of radionuclide release to the accessible environment.

NRC Summary of Evaluation Findings for Localized Corrosion of the Waste Package Outer Barrier

The NRC staff reviewed DOE's models for localized corrosion of the waste package outer barrier that were implemented in the TSPA code. On the basis of its review, the NRC staff finds that DOE used appropriate experimental tests and applicable technical literature to provide adequate support for the localized corrosion initiation and propagation models. For the first 12,000 years after repository closure, the NRC staff concluded that waste package breach by localized corrosion is unlikely because the intact drip shields will prevent seepage waters from contacting the waste package (SER Sections 2.2.1.3.1.3.1 and 2.2.1.3.2.6). For the time period beyond 12,000 years after repository closure, the NRC staff concludes that DOE's models showed a low probability for localized corrosion initiation because the proposed repository environment (i.e., temperature, pH, and chemical composition of seepage waters) will not support the initiation of localized corrosion. In addition, DOE appropriately identified and adequately considered features and processes such as corrosion potential, relative humidity, localized corrosion initiation and propagation rate, temperature, pH, and chemical composition of seepage waters, that affect the waste package outer barrier capabilities for the initial 10,000 year period, and projected these features and processes beyond the 10,000 year post-disposal period through the period of geologic stability. Therefore, the NRC staff finds acceptable DOE's analytic models for localized corrosion of the waste package outer barrier in the TSPA model.

2.2.1.3.1.3.2.3 Stress Corrosion Cracking of the Waste Package Outer Barrier

Stress corrosion cracking generally refers to a process whereby cracks form in metals or alloys in a corrosive environment and under sustained tensile stresses. DOE presented data indicating that Alloy 22 is highly resistant to stress corrosion cracking in the environmental conditions (e.g., temperature, pH, and chemical constituents of seepage water brines) that are expected to occur in the repository, as detailed in SAR Section 2.3.6.5.2 and SNL (2007bb, Section 6.2). Because of uncertainty regarding the long-term environmental conditions in the repository, however, DOE's model in the TSPA code assumes that the repository environment supports stress corrosion cracking, such that sufficient residual tensile stress was the only criterion for stress corrosion cracking occurrence (SAR Section 2.3.6.5.1). In SAR Section 2.3.6.5, DOE evaluated stress corrosion cracking caused by residual stresses from waste package fabrication. In SAR Section 2.3.4.5, DOE also evaluated stress corrosion cracking caused by the residual stresses resulting from impacts to the waste package during seismic ground motions. This section of the SER includes NRC staff's review of DOE's model abstractions for stress corrosion cracking of the waste package outer barrier.

Conceptual Models

DOE's models for stress corrosion cracking of the waste package outer barrier treat crack initiation (i.e., the formation of cracks on the waste package surface) and crack propagation (i.e., growth of cracks from the surface through the outer barrier) as distinct phenomena. In the TSPA code, DOE assumed that cracks initiated on areas of the waste package surface where the magnitude of the sustained tensile stress was greater than a threshold value, which DOE referred to as the residual stress threshold (RST) (SAR Section 2.3.6.5.2.1). Cracks initiated by this sustained stress were referred to as incipient cracks. In DOE's model, the residual stresses for crack initiation could result from fabrication processes such as welding or from impacts to the waste package during a seismic ground motion event. DOE stated that the concept of a threshold stress that must be exceeded for the onset of stress corrosion cracking is widely accepted in the technical literature (e.g., ASM International, 1987aa). In the TSPA code, DOE also assumed that waste package weld flaws (e.g., voids and slag inclusions) were initiated cracks, regardless of the magnitude of the residual stress (SAR Section 2.3.6.5.3.1).

DOE used different conceptual models for the propagation of cracks initiated by fabrication stresses and weld flaws and those initiated by seismically induced stresses, respectively. With regard to cracks initiated by seismically induced stresses, DOE did not explicitly model crack propagation. Rather, DOE assumed that cracks instantaneously propagated through the wall at the time of initiation (SAR Section 2.3.4.5.1.2.1). This assumption minimizes the time for through-wall propagation of cracks. With regard to cracks initiated by fabrication stresses and weld flaws, DOE assumed that the stress intensity factor at the tip of the initiated crack must be greater than a threshold value for the crack to propagate (SAR Section 2.3.6.5.3.2). DOE stated that the concept of a threshold stress intensity factor is consistent with the general understanding of crack fracture mechanics (Jones and Ricker, 1987aa; Sprowls, 1987aa).

To calculate the rate of growth for cracks with a stress intensity factor greater than the threshold value, DOE used the slip-dissolution film-rupture (SDFR) model, as discussed in SAR Section 2.3.6.5.3.2 and SNL (2007bb, Section 6.4.2). In the SDFR model, crack growth is related to the rupture and subsequent reformation of the passive metal oxide film at the crack tip. DOE stated that several studies (Ford and Andresen, 1988aa; Andresen, 1991aa; Andresen and Ford, 1994aa) used the SDFR model to accurately calculate crack growth rates in stainless steel and nickel-based alloys similar to Alloy 22 (e.g., Alloys 182 and 600). In SAR Section 2.3.6.5.3.3, DOE described an alternative conceptual model for the crack growth rate: the coupled environmental fracture model. The coupled environmental fracture model is based on conservation of electrons involved in the corrosion process (Macdonald and Urquidi-Macdonald, 1991aa; Macdonald, et al., 1994aa). It incorporates the effects of oxygen concentration, its flow rate, and the conductivity of the external environment and accounts for the effect of stress on crack growth. DOE did not use the coupled environmental fracture model in the TSPA calculation because it calculated a slower crack growth rate than the SDFR model.

NRC Staff's Review

The NRC staff reviewed DOE's models for stress corrosion cracking initiation. The NRC staff finds that DOE's use of a threshold stress for the onset of stress corrosion cracking in the waste package outer barrier is acceptable because this concept is consistent with reports of stress corrosion cracking behavior in a range of passive alloys similar to Alloy 22 (e.g., ASM International, 1987aa). The NRC staff concludes that the initiation of stress corrosion cracking in Alloy 22 is similar to initiation in other passive alloys involving repetitive rupture and regeneration of a passive film. The NRC staff also finds that, because weld flaws may be present in the waste package outer barrier at the time of emplacement, it is acceptable for DOE to model the flaws as initiated cracks. Therefore, the NRC staff finds that DOE's conceptual models for crack initiation are acceptable.

The NRC staff also reviewed DOE's models for stress corrosion cracking propagation. For the propagation of cracks initiated by seismically induced stresses, the NRC staff finds that DOE's model does not take credit for the possibility that initiated cracks could arrest before propagating through the barrier or the time it would take for cracks to pass through the barrier. Therefore, the NRC staff concludes that DOE's model for propagation of seismically induced cracks is acceptable because it will underestimate the time it takes for cracks to breach the waste package outer barrier. For propagation of weld flaws and cracks initiated by fabrication stresses, the NRC staff finds that DOE's use of a threshold stress intensity factor is consistent with the technical understanding of crack fracture mechanics. Further, DOE has validated the SDFR model predictions with experimental results (SAR Figure 2.3.6-29). Therefore, the NRC staff finds that the use of SDFR model for calculating the crack growth rates is acceptable. The NRC staff also finds that the slip-dissolution film-rupture model calculates higher crack growth rates than DOE measured using the reversing direct current measurement technique on compact-tension-type Alloy 22 fracture mechanics specimens (SAR Figure 2.3.6-34), or greater than was calculated by the alternative coupled environmental fracture model (Ford and Andresen, 1988aa). As such, the NRC staff finds that DOE's models for propagation of weld flaws and cracks initiated by fabrication stresses are acceptable because they will not overestimate the time it takes for cracks to breach the waste package outer barrier.

Stresses for Crack Initiation and Propagation

The NRC staff reviewed DOE's approaches to establish residual stress threshold values for crack initiation and the threshold stress intensity factor for the propagation of weld flaws and cracks initiated by fabrication stresses.

Residual Stress Threshold

DOE performed laboratory tests to establish the value of the residual stress threshold for Alloy 22. DOE performed constant-load crack initiation tests (SAR Section 2.3.6.5.2.1.1), slow strain rate tests (SAR Section 2.3.6.5.2.1.2), and U-bend stress corrosion cracking initiation tests (SAR Section 2.3.6.5.2.1.3). These tests were performed for up to 5 years for Alloy 22 specimens with metallurgical conditions representative of waste package metallurgical conditions in the repository (i.e., welded, thermally aged, cold worked). The tests were performed in the temperature range of 25 to 165 °C [77 to 329 °F] in different brines, including basic simulated water, simulated dilute water, simulated concentrated water, and simulated acidified water, the compositions of which are shown in SAR Table 2.3.6-1.

For the constant-load crack initiation tests (SAR Section 2.3.6.5.2.1.1), DOE exposed Alloy 22 specimens to basic simulated water (pH of 10.3) at 105 °C [221 °F] for up to 28,000 hours (approximately 3 years). The test specimens were subjected to tensile stress up to 2.1 times the at-temperature yield strength for as-received materials and 2.0 times the yield strength of the welded materials, which corresponds to approximately 95 percent of the ultimate tensile strength of Alloy 22 in the respective material conditions. DOE reported that no sample ruptured during the test, as shown in SAR Figure 2.3.6-28. For the slow strain rate testing (SAR Section 2.3.6.5.2.1.2), Alloy 22 specimens were exposed to simulated acidified water, basic simulated water, simulated concentrated water, and calcium-chloride-type brines over a range of temperatures, with and without applied potential (SAR Table 2.3.6-14). DOE stated that it did not observe stress corrosion cracking in most experimental conditions, though stress corrosion cracking was observed in simulated concentrated water with large applied anodic potentials. DOE concluded in SNL (2007bb, Section 6.2.1.3), however, that such potentials are not representative of repository conditions. For the U-bend stress corrosion cracking initiation tests (SAR Section 2.3.6.5.2.1.3), Alloy 22 specimens were tested in simulated dilute water, simulated concentrated water, and simulated acidified water for 5 years with no evidence of stress corrosion cracking initiation.

Even though stress corrosion cracking of Alloy 22 was not observed in the experimental testing, DOE concluded that cracks may initiate at lower stresses on the repository time scale, yet would not be observed in short-term laboratory tests, as described in SNL (2007bb, Section 6.2.2). Therefore, DOE stated that there existed some uncertainty associated with the value of the residual stress threshold. Thus, to establish the residual stress threshold value for the TSPA code, DOE applied a safety factor of two to the maximum stress that Alloy 22 specimens withstood with no evidence of stress corrosion cracking initiation. As outlined in SNL (2007bb, Section 6.2.2), DOE determined that this maximum stress was 210 percent of the Alloy 22 at-temperature yield strength, as measured during constant-load crack initiation testing in basic simulated water. This approach established the upper bound for the residual stress threshold to be 105 percent of the Alloy 22 at-temperature yield strength, with a safety factor of two. DOE stated that use of the safety factor of two is consistent with general engineering practice and has been used to establish the allowable long-term fatigue stress on engineering components (American Society of Mechanical Engineers, 1969aa). To further account for uncertainty in the residual stress threshold, DOE established 90 percent of the Alloy 22 at-temperature yield stress as a lower bound. Thus, in the TSPA code, DOE sampled the residual stress threshold from a uniform distribution between 90 and 105 percent of the Alloy 22 at-temperature yield stress, as shown in SNL (2007bb, Table 6-3).

In response to the NRC staff's request for additional information that DOE assess the observed case of stress corrosion cracking initiation in simulated concentrated water, DOE provided new data in DOE (2009cj, Enclosure 1) from U-bend testing of as-welded and mill-annealed Alloy 22 specimens at 165 °C [329 °F] in aerated simulated concentrated water (Andresen and Kim, 2009aa). After 32,000 hours (approximately 3.5 years), no stress corrosion cracking was observed for stresses estimated to be at or slightly above the at-temperature yield strength of Alloy 22. DOE cited an additional study in which low strain rate crack initiation tests were performed on Alloy 22 specimens in simulated concentrated water at 86.0 and 89.0 °C [187 and 192 °F] (Fix, et al., 2003aa). DOE reported that crack initiation did not occur until the tensile stress exceeded a value of 605 MPa [87.7 ksi], which is approximately 160 percent of the room temperature yield strength of Alloy 22. On the basis of this information, DOE concluded that the range of the residual stress threshold given in SAR Section 2.3.6.5 was adequate.

DOE evaluated the expected fabrication-induced residual stresses during the postclosure period to determine whether such stresses could exceed the residual stress threshold. DOE stated that it plans to do a stress-relief heat treatment to mitigate the stresses in the waste package outer barrier after fabrication (SAR Section 1.5.2.7.1) following the standards specified in American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC-4600 (American Society of Mechanical Engineers, 2001aa). DOE concluded that fabrication-induced residual stresses will not exceed the residual stress threshold for portions of the waste package that are heat treated. In DOE's fabrication process, however, the heat treatment takes place before the waste is placed in the waste package and the outer lid is welded onto the shell. Welding the outer lid onto the waste package shell may induce residual stresses in the region of the closure weld that the heat treatment process cannot mitigate, as described in SNL (2007bb, Section 6.5.3.1). Therefore, DOE concluded that the region of the waste package closure weld is the only part of the waste package outer barrier where fabrication-induced stresses could cause stress corrosion cracking. DOE plans to implement a process called low plasticity burnishing (a process whereby the material surface is plastically deformed to create a layer with compressive residual stress) to delay the initiation of incipient stress corrosion cracking (SAR Section 1.5.2.7.2.2). The current waste package design requires a compressive residual stress to a depth of at least 3.0 mm [0.12 in] below the weld surface (SAR Table 1.9-9, Design Control Parameter 03-17). DOE concluded that the initiation of stress corrosion cracking would be delayed by the time it would take for general corrosion to corrode through at least the 3.0-mm [0.12-in] burnished layer.

DOE performed finite element analyses to calculate the residual stress profile of the weld, as detailed in SAR Section 2.3.6.5.2.3 and SNL (2007bb, Section 6.5.3.3), when the closure weld was plasticity burnished resulting in compressive stresses to a depth of 3.0 mm [0.12 in] below the weld surface. The analysis simulated multipass welds, with the residual stress represented as a function of welding parameters, thermal transients, temperature-dependent material properties, and elastic–plastic stress reversals. DOE analyses indicated that the residual stress decays rapidly with increasing radial distance from the weld line and is negligible at a distance from the weld line approximately equal to the thickness of the waste package wall, as shown in SNL (2007bb, Figures 6-19 through 6-22). Given the rapid decay in weld-induced stress with increasing distance from the weld line, DOE assumed that initiation of stress corrosion cracking by fabrication-induced stress could only occur in patches representing the waste package closure weld in the TSPA model. These patches represent approximately 2.67 percent and 2.95 percent of the total surface area for the CSNF and the codisposal waste packages, respectively, as outlined in SNL (2008ag, Section 6.3.5.1.2).

In the region of the closure weld, DOE calculated that radial stresses do not exceed the residual stress threshold through the entire thickness of the weld, but that hoop stresses can exceed the residual stress threshold at a depth of approximately 5.0 to 7.5 mm [0.20 to 0.30 in] below the weld surface, as described in SNL (2007bb, Section 6.5.5.2.2). In the TSPA code, DOE represents the hoop stress as a function of depth from the weld surface in SNL (2007bb, Equation 64). DOE also considered angular variability in the residual stress around the circumference of the waste package closure weld in SNL (2008ag, Equation 6.3.5-6). On the basis of literature reports (e.g., Shack, et al., 1980aa), DOE calculated that the residual stress may have circumferential variation up to ± 2.50 ksi [± 17.24 MPa] from the mean stress, as detailed in SNL (2007bb, Section 6.5.6.1). SAR Figure 2.3.6-30 shows the angular variability in the residual hoop stress profile in the waste package closure weld. More generally, DOE identified literature reports (e.g., Mohr, 1996aa; Pasupathi, 2000aa), which indicated that welding and stress mitigation processes introduce uncertainty into the weld residual stress profile. The reports indicated that the uncertainty range for residual stress may be between

5 percent and 35 percent of the material yield strength. On the basis of the fabrication techniques and process controls planned for the waste package closure weld, DOE selected a three-standard deviation uncertainty range, equivalent to ±15 percent of the at-temperature yield strength of Alloy 22, as outlined in SNL (2007bb, Section 6.5.6.2). This is implemented in the TSPA model by applying a scaling factor to the residual stress. The scaling factor is sampled from a truncated (± 3-standard deviations) normal distribution where the mean is 0 and the standard deviation is 5 percent of the Alloy 22 at-temperature yield strength.
SAR Figure 2.3.6-32 shows the uncertainty in the residual hoop stress profile in the waste package closure weld.

DOE compared the residual stress profile calculated by the finite element analysis to the residual stress experimentally measured by Woolf (2003aa) for plasticity-burnished Alloy 22 simulated closure welds, as described in SNL (2007bb, Section 6.5.6.5). As shown in SNL (2007bb, Figure 6-60), Woolf (2003aa) measured compressive residual stress to a depth of more than 7.0 mm [0.28 in] from the weld surface. DOE concluded that the calculated residual stress profile implemented in the TSPA code underestimates the extent of stress mitigation by plasticity burnishing.

NRC Staff's Review

The NRC staff reviewed DOE's approach to establish the value of the residual stress threshold for the waste package outer barrier. The NRC staff finds that DOE used different types of stress corrosion cracking initiation tests, including constant-load crack initiation tests (SAR Section 2.3.6.5.2.1.1), slow-strain rate tests (SAR Section 2.3.6.5.2.1.2), and U-bend stress corrosion cracking initiation tests. These tests are appropriate for measuring the stress corrosion cracking susceptibility because they are consistent with applicable standards, including ASTM G 30-94 (ASTM International, 1994aa); ASTM E 399-90, (ASTM International, 1991aa); ASTM G 129-00 (ASTM International, 2000ab); and ASTM G 49-85 (ASTM International, 2000aa). For the material conditions for the stress corrosion cracking initiation tests, the NRC staff finds that DOE tested Alloy 22 specimens with microstructures that are representative of those expected for the waste package based on the fabrication procedures set forth in SAR Section 1.5.2.7. For the test solutions for the corrosion tests, the NRC staff finds that DOE tested Alloy 22 specimens in a range of solutions, including simulated acidified water, simulated concentrated water, simulated dilute water, and basic simulated water. Based on stress corrosion cracking susceptibility of Alloy 22 studies published by Chiang, et al. (2005aa, 2006aa) and Shukla, et al. (2006aa), the NRC staff concludes that the residual stress threshold for the waste package in the repository may depend on such factors as the pH and concentration of ionic species in water that contacts the waste package. Therefore, the NRC staff reviewed the corrosion test solutions to determine whether they are adequate to measure the residual stress threshold in repository conditions. The NRC staff finds that the corrosion test solutions are more chemically aggressive than waters expected to occur within repository drifts, including starting water compositions in DOE's near-field chemistry model described in SAR Section 2.3.5.5, and waters considered in the NRC staff's independent analysis of in-drift water evolution, described in SER Section 2.2.1.3.3.3.2. As such, the NRC staff concludes that it is acceptable for DOE to measure the residual stress threshold on the basis of tests in these simulated brines.

The NRC staff reviewed DOE's use of the data from the stress corrosion cracking tests to establish the range for the residual stress threshold. The NRC staff finds that stress corrosion cracking initiation was observed only on Alloy 22 in simulated concentrated water under applied potential. The NRC staff finds that this condition is not representative of repository conditions because experimental studies showed that Alloy 22 specimens submerged in simulated concentrated water underwent stress corrosion cracking only when the applied potential was higher than the corrosion potentials (Fix, et al., 2003aa; Chiang, et al., 2005aa, 2006aa; Shukla, et al., 2006aa). Therefore, the NRC staff finds that DOE did not observe stress corrosion crack initiation in any condition representative of the repository environmental conditions. As such, the NRC staff concludes that reducing the stress that Alloy 22 withstood without evidence of stress corrosion cracking by a safety factor of 2 is acceptable to establish the upper bound for the residual stress threshold to be 105 percent of the at-temperature yield stress. The NRC staff finds that a residual stress threshold lower bound of 90 percent of the at-temperature yield stress appropriately quantifies the range of uncertainty associated with the value of this parameter. The NRC staff finds that the distribution from which DOE sampled the residual stress threshold in the TSPA code is acceptable because it is based on the experimental data with lower and upper bounds of 90 and 105 percent at-temperature yield stress and, therefore, is unlikely to overestimate the stress at which stress corrosion cracking initiates in the waste package outer barrier.

The NRC staff reviewed DOE's analysis of fabrication-induced stresses in the waste package outer barrier. The NRC staff finds that a heat treatment process that follows the standards specified in American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC-4600 (American Society of Mechanical Engineers, 2001aa) is consistent with nuclear industry practice to relieve fabrication-induced stresses in components fabricated with materials similar to Alloy 22. Therefore, the NRC staff finds it acceptable that in the TSPA analysis, DOE considered that stress corrosion cracking caused by fabrication-induced stresses could only occur in the region of the waste package closure weld, which is not heat treated.

The NRC staff also reviewed DOE's calculation of the closure weld residual stress profile, as described in SAR Section 2.3.6.5.2.3. The NRC staff finds that DOE's finite element stress analyses were performed with a well-established methodology that is accepted in the technical literature (e.g., NRC, 1977aa; Rybicki and Stonsifer, 1979aa). Further, the NRC staff concludes that DOE's calculated residual stress profile is consistent with literature reports, which show that residual stresses in highly controlled welds tend to persist for only a short distance from the weld line (ASM International, 1993aa). The NRC staff also finds that plasticity burnishing is an effective stress mitigation technique in engineered components (Prevey and Cammett, 2001aa) and that the stresses DOE calculated are lower than measured values (Woolf, 2003aa) for plasticity burnished Alloy 22 welds. Additionally, the NRC staff finds acceptable the uncertainty and variability in the residual stress profile implemented in the TSPA code because the model is consistent with weld stress analyses reported in the technical literature (Mohr, 1996aa; Pasupathi, 2000aa). Therefore, the NRC staff finds that the residual stress profile for the waste package outer barrier closure weld used in the TSPA code is acceptable because it will not underestimate the residual stress.

Threshold Stress Intensity Factor

DOE determined the numerical value of the stress intensity factor threshold using a crack blunting criterion, as detailed in SAR Section 2.3.6.5.1 and SNL (2007bb, Section 6.4.5). According to the crack blunting criterion, crack growth will arrest if the crack tip radius decreases, because the general corrosion rate at the sides of the crack is greater than the rate at which the crack tip is advancing (Andresen and Ford, 1994aa). DOE calculated the threshold stress intensity factor as a function of the Alloy 22 general corrosion rate and the repassivation slope, a parameter related to the rate at which Alloy 22 repassivates following a passive film rupture, as shown in SNL (2007bb, Equation 19). DOE used a point value of 7.23 nm/yr [2.85×10^{-7} in/yr] for the general corrosion rate of Alloy 22, on the basis of the 5-year corrosion data described in SAR Section 2.3.6.3. DOE determined the value of the repassivation slope by measuring the crack growth rate for fatigue precracked Alloy 22 compact tension specimens at 110 °C [230 °F] in basic simulated water and 150 °C [302 °F] in simulated concentrated water, as outlined in SAR Section 2.3.6.5.2.4 and SNL (2007bb, Section 6.4.4.2). For these conditions, the measured values of the repassivation slope are shown in SAR Table 2.3.6-17. DOE considered epistemic uncertainty in the repassivation slope and, in turn, the threshold stress intensity factor. In the TSPA code, DOE sampled the repassivation slope from a normal distribution. The mean threshold stress intensity factor was calculated as 6.62 MPa-$m^{0.5}$ [6.02 ksi-$in^{0.5}$], with lower and upper bounds of 1.96 MPa-$m^{0.5}$ [1.78 ksi-$in^{0.5}$] and 15.38 MPa-$m^{0.5}$ [14.00 ksi-$in^{0.5}$], respectively. DOE stated that this range corresponds to values reported in the technical literature (e.g., Jones, 1992aa) for other corrosion-resistant chromium-nickel-iron alloys.

DOE calculated the stress intensity factor profile for the waste package closure weld to determine whether the stress intensity factor could exceed the threshold value to cause crack propagation during the postclosure period, as described in SAR Section 2.3.6.5.2.3 and SNL (2007bb, Section 6.5). DOE used an approach in which it calculated the stress intensity factor for a relatively simple crack geometry and given stress distribution, then modified the solution for the waste package closure weld with a geometry correction factor, as detailed in SNL Section (2007bb, 6.5.3.3.3). DOE represented a radially oriented crack in the closure weld with the idealized case of a semicircular crack in an infinite plate, as outlined in SNL (2007bb, Section 6-17). The geometry correction factor was obtained by comparing the simplified solution to finite element analysis solutions for a number of crack sizes. Using this approach, DOE calculated the stress intensity factor profile for the plasticity-burnished waste package closure weld, as shown in SAR Table 2.3.6-16. Because DOE's calculated stress intensity factor was a linear function of the residual stress, uncertainty and variability in the residual stress profile were also represented in the stress intensity factor profile used in the TSPA code. SAR Figure 2.3.6-30 showed the angular variability in the stress intensity factor profile for the closure weld, and SAR Figure 2.3.6-32 showed the uncertainty in the stress intensity factor profile in the waste package closure weld.

NRC Staff's Review

The NRC staff reviewed the approach to calculate the threshold stress intensity factor for the waste package closure weld. The NRC staff finds that the stress concentration at the tip of a crack generally decreases with increasing crack tip radius. In addition, the crack tip radius will increase if the crack sidewall general corrosion rate is higher than the crack tip advance rate. Therefore, the NRC staff concludes that the crack blunting criterion that DOE used is acceptable because it is consistent with the linear elastic fracture mechanics theory reported in the technical literature (e.g., Andresen and Ford, 1994aa). Further, the NRC staff finds DOE's method for calculating the repassivation slope, by measuring the crack growth rate for fatigue pre-cracked Alloy 22 compact tension specimens, acceptable because the crack growth rates were extremely low {less than 1.00×10^{-8} mm/s [3.94×10^{-10} in/s]} even though the stress intensity factor for the cracks was significantly higher than the sampled range for the threshold stress intensity factor in the TSPA code.

The NRC staff also verified DOE's calculation of the stress intensity factor profile for the waste package closure weld. The NRC staff finds DOE's modeling of the radial crack in the waste package closure weld as a semicircular crack in an infinite plate acceptable because the hoop stress in the weld region decreases rapidly with increasing distance from the weld line. Additionally, the NRC staff finds that the variability and uncertainty in the stress intensity factor profile modeled in the TSPA code adequately reflects associated variability and uncertainty in DOE's calculated residual stress profile because the stress intensity factor profile in the TSPA code parameters were determined from mean, lower, and upper bounds of calculated residual stress profiles. Therefore, the NRC staff finds the stress intensity factor used in the TSPA code for the waste package outer barrier closure weld acceptable because it will not underestimate the stress intensity factor.

Crack Size and Density

The NRC staff reviewed DOE's approaches to calculate the size and density of cracks initiated by fabrication-induced stresses and weld flaws, and those initiated by seismically induced stresses.

Cracks Initiated by Fabrication-Induced Stresses and Weld Flaws

DOE's analytic models assumed that the cracks initiated by fabrication-induced stresses exceeded the residual stress threshold and have a uniform size density. DOE assumed that it is energetically favorable for the cracks to have elliptical shape, as shown in SNL (2007aj, Figure 6.3-3), with crack length (i.e., major axis of the ellipse) of 50.0 mm [1.97 in]. DOE selected this length because it calculated that weld-induced stresses can persist on both sides of the weld centerline up to a distance approximately equal to the nominal thickness of the waste package outer barrier {i.e., 25 mm [0.98 in]}. The crack opening width was calculated using a fracture mechanics equation derived from an analysis of the energy associated with crack free surfaces. In this manner, DOE calculated a crack opening width of 0.1956 mm [7.700×10^{-3} in]. Given the crack length of 50.0 mm [1.97 in] and width of 0.1956 mm [7.700×10^{-3} in], DOE calculated that the opening area of an individual incipient crack was 7.682 mm^2 [1.190×10^{-2} in^2], which is assumed to be constant through the waste package wall (SNL, 2007aj). DOE stated that a crack of these dimensions would permit diffusive transport, but preclude advective transport by water (FEP 2.1.03.10.0A; SNL, 2008ac). Moreover, DOE assumed that the density of through-wall cracks is constrained by stress-field interactions in the area around the crack, which limit the ability of cracks in relative proximity to propagate through the waste package wall, as described in SNL Section 6.6.1 (2007bb). DOE's analysis indicated that the minimum spacing between through-wall cracks is equal to the thickness of the waste package outer barrier, which is 25 mm [0.98 in] (Structural Integrity Associates, 2002aa).

DOE used a different approach to model the size and density of weld flaw cracks in the waste package outer barrier closure weld. For the outer closure weld, DOE determined that the size and density of flaws would be small because of (i) highly controlled welding procedures that would limit flaw generation and (ii) extensive postweld nondestructive examination used to identify weld flaws, as detailed in SNL (2007aa, Section 6.3.1). DOE stated in SAR Section 1.5.2.7 that weld fabrication and inspection will follow the requirements of American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section III, Division 1, Subsection NC (American Society of Mechanical Engineers, 2001aa). The ASME Code specifies that flaws larger than 1.6 mm [6.30×10^{-2} in] be detected and repaired, a requirement that is incorporated as a waste package Design Control Parameter [SAR Table 1.9-9, Design Control Parameter 03-17(b)]. To determine the size and density of flaws that may be expected in the waste package closure welds, DOE fabricated simulated welds (Smith, 2003aa; SAR Section 2.3.6.5.2.2). DOE stated that postweld nondestructive examination detected all flaws in the simulated welds that were larger than ASME Code allowed.

DOE performed a statistical analysis using data from the simulated welds to derive probability distributions for the size and density of flaws to be sampled in the TSPA analysis, as outlined in SNL (2007aa, Section 6.3.1). DOE used a Bayesian approach, consistent with weld flaw analyses in the technical literature (e.g., American Nuclear Society/Institute of Electrical and Electronics Engineers, 1983aa). Following this approach, DOE determined that a Poisson distribution best represents the undetected weld flaw density. DOE calculated that after performing postweld non-destructive examination, the mean size of flaws that would go undetected is 1.00 mm [3.94×10^{-2} in], with 5th and 95th percentile values of 0.07 and 2.6 mm [2.76×10^{-3} and 1.02×10^{-1} in], respectively, as described in SNL (2007aa, Appendix A). DOE calculated that after the non-destructive examination, there would be a mean of approximately one weld flaw per 140 m^3 [4.94×10^3 ft^3] of weld volume, with 5th and 95th percentile values of approximately one weld flaw per 56 m^3 [1.98×10^3 ft^3] and one weld flaw per 264 m^3 [9.32×10^3 ft^3], respectively, as detailed in SNL Appendix A (2007aa). Given the expected closure weld volume, DOE calculated that there is about an 84 percent probability that a waste package will have no weld flaws, a 14 percent probability that a waste package has one flaw, and a 2 percent probability that a waste package has two or more flaws (SAR Table 2.3.6-18).

In the TSPA code, DOE calculated that only radially oriented flaws (i.e., those that make an angle of greater than 45° with respect to the weld line) are able to propagate because the primary stress component in the closure weld is the hoop stress (SAR Section 2.3.6.5.3.1). DOE determined that there is little driving force for the propagation of cracks that make an angle of less than 45° with respect to the primary stress direction, as outlined in SNL (2007bb, Section 6.3.4.3). DOE analyzed the flaws in the simulated welds to calculate a probability distribution for the orientation of flaws in the closure weld, as described in SNL (2007aa, Section 6.3.1.5). Using the Bayesian approach, DOE calculated that 0.8 percent of weld flaws will be radially oriented such that they can propagate under a hoop stress. DOE concluded that this calculation was supported by the independent analyses of Shcherbinskii and Myakishev (1970aa) who reported that most (~99 percent) weld flaws are oriented within about ±13° from the weld line.

In the TSPA code, DOE also assumed that only those weld flaws exposed to the environment by general corrosion during the postclosure period would be susceptible to propagation, as outlined in SNL (2007bb, Section 6.3.4.2). In the TSPA code, DOE calculated that 25 percent of weld flaws will be exposed and able to propagate based upon the approximate percentage of the waste package weld that would be removed by general corrosion during the postclosure period (SAR Section 2.3.6.3). On the basis of the small number of embedded weld flaws capable of propagation, DOE concluded that breach of the waste package outer barrier by weld flaw cracks is far less likely than breach by incipient cracks initiated where the residual stress is greater than the residual stress threshold, as outlined in SNL [2008ag, Section 6.3.5.1.3(a)].

NRC Staff's Review

The NRC staff reviewed DOE's approaches to calculate the size and density of cracks initiated by fabrication-induced stresses and weld flaws in the waste package outer barrier. Regarding cracks initiated by fabrication-induced stresses, the NRC staff concludes that reports in the technical literature indicate that crack propagation requires energy to create the new crack surfaces (Anderson, 2005aa). Absent external stresses on the waste package, the NRC staff finds that the energy for crack propagation would come from the internal stresses in the weld. Thus, the NRC staff concludes that crack propagation will mitigate the residual stress in the weld by creating new free surfaces, thereby causing cracks to narrow as they propagate through the waste package outer barrier. DOE assumed a constant crack opening area through the waste package wall and did not take credit for stress mitigation and crack narrowing in its model. Therefore, the NRC staff finds acceptable the opening area DOE calculated because it does not underestimate the actual crack opening area. The NRC staff also finds that stress mitigation by crack propagation will constrain the density of through-wall cracks. As such, the NRC staff finds that DOE's calculated crack density is acceptable because it does not underestimate the actual crack density.

The NRC staff also reviewed DOE's analysis of the size and density of flaws in the waste package closure weld. The NRC staff finds that the use of data from the simulated welds to evaluate the flaws in the waste package closure weld is acceptable because the simulated welds were fabricated with similar materials, procedures, equipment, and postwelding non-destructive evaluation methods as will be applied for the actual waste package welds. Further, the NRC staff finds that DOE's use of a Bayesian approach to develop probability distributions for the size and density of undetected weld flaws is acceptable for Alloy 22 for the following reasons: (i) the Bayesian approach is appropriate when direct measurements are unavailable and prior measurements are used to estimate weld flaw distribution on the waste packages and (ii) the Bayesian approach has been used in similar situations in the probabilistic risk assessment at NRC in accordance with American Nuclear Society/Institute of Electrical and Electronics Engineers (1983aa). For these reasons, DOE's use of the Bayesian approach to extrapolate information on weld flaws using prior measurements is acceptable. The NRC staff also finds that DOE's analysis was consistent with NRC analyses of flaws in dry storage cask welds, as described in NRC Appendix B (2006ab). Finally, the NRC staff finds that DOE assumed a constant crack opening area through the waste package wall and did not take credit for stress mitigation and crack narrowing in its model for weld flaws. On the basis of this information, the NRC staff finds that DOE's distributions for weld flaw size and density used in the TSPA analyses are acceptable because they will not underestimate these parameters.

Cracks Initiated by Seismically Induced Stresses

For stress corrosion cracks caused by impacts to the waste package during seismic ground motion, DOE sampled a parameter in the TSPA code called the crack area density, which is the product of the crack size and the crack density, as identified in SAR Section 2.3.4.5.1.4.1 and SNL (2007bb, Section 6.7.2). The crack area density is a unitless scalar measure that, when multiplied by the size of the damaged area on the waste package surface, gives the total open area of the through-wall cracks. For seismically induced stress corrosion cracking, DOE assumed that through-wall cracks have the same shape characteristics as cracks in the closure welds, as shown in SNL (2007aj, Figure 6.3-3). In contrast to the weld cracks, however, DOE considered uncertainty in the size and density for the cracks induced by seismic ground motion. DOE evaluated the uncertainty by calculating the crack area density using two conceptual models in which the crack size and crack density values were varied, as described in SNL (2007bb, Section 6.7.3). Both conceptual models use a regular hexagonal array of cracks on the waste package surface because this gives high effective crack density, as described in SNL (2007bb, Section 6.7.2). In the first conceptual model in SNL (2007bb, Section 6.7.3.1), cracks abutted tip-to-tip and the distance between parallel rows of cracks was the waste package wall thickness. In the second conceptual model in SNL (2007bb, Section 6.7.3.2), cracks could overlap (crack length was two times that in the first conceptual model) and the distance between crack centers was the wall thickness (crack number density was lower than that assumed in the first conceptual model). These resulted in SNL (2007bb, Equations 37 and 40) that DOE used to calculate the lower and upper bounds, respectively, for the crack area density sampled in the TSPA code.

Using this approach, DOE calculated a lower bound for crack area density of approximately 3.27×10^{-3} (i.e., stress corrosion cracking breached area is 0.327 percent of the waste package damaged area) and an upper bound of approximately 1.31×10^{-2} (i.e., stress corrosion cracking breached area is 1.31 percent of the waste package damaged area) (SAR Section 2.3.4.5.1.4.1). DOE considered an alternative conceptual model in which a single crack circumscribed the damaged area, as described in SNL (2007bb, Section 6.7.4). For this conceptual model, DOE calculated a crack area density of 7.22×10^{-3}, which is within the bounds given by the hexagonal crack network models. DOE determined that the alternative conceptual model provided support for the crack area density range calculated by the primary conceptual models. Therefore, in the TSPA code, DOE samples the crack area density from a uniform distribution between the bounding values given by the hexagonal crack network models.

NRC Staff's Review

The NRC staff reviewed DOE's approach to establish the value of the crack area density for seismically induced stress corrosion cracks. The NRC staff finds that reports in the technical literature indicate that crack propagation requires energy to create new crack surfaces (Anderson, 2005aa). Absent external stresses on the waste package, the NRC staff finds that the energy for crack propagation would necessarily come from the residual stresses generated from impacts to the waste package. Thus, the NRC staff finds that crack propagation will mitigate the seismically induced stresses by creating new free surfaces, thereby causing cracks to narrow as they propagate through the waste package outer barrier. The NRC staff finds that DOE assumed a constant crack opening area through the waste package wall and did not take credit for stress mitigation and crack narrowing in its model. Moreover, the NRC staff finds that stress mitigation by crack propagation will constrain the density of through-wall cracks. As such, the NRC staff finds that DOE's calculated range for the crack area density is acceptable because it does not underestimate value of this range.

Abstraction and Integration

For the Nominal and Seismic Ground Motion Modeling Cases in the Total System Performance Assessment (TSPA) code, DOE's model abstraction for stress corrosion cracking of the waste package closure weld was implemented in the waste package and Drip Shield Degradation Submodel, as detailed in SNL (2008ag, Section 6.3.5.1). In DOE's submodel, the waste package closure weld area is represented by an annulus that is one patch wide and has the same radius as the waste package, as shown in SNL (2008ag, Figure 6.3.5-4). This results in about 40 patches to model the waste package closure weld. The waste package general corrosion abstraction and stress corrosion cracking abstraction, respectively, are implemented independently on each of the patches. As each patch thinned by general corrosion, the submodel calculated the residual stress on the patch on the basis of the through-wall residual stress profile. At each realization time step, the submodel compared the residual stress on the patch to the sampled residual stress threshold. If the residual stress on the patch was greater than the residual stress threshold, the submodel assumed that stress corrosion cracking initiated. The submodel also distributed weld flaws among the patches on the basis of the probability distributions for the weld flaw size and density. To determine whether initiated cracks in the waste package closure weld could propagate, the submodel calculated the stress intensity factor at the crack tip on the basis of the through-wall stress intensity factor profile and compared it to the sampled threshold stress intensity factor. If the stress intensity factor was greater than the threshold stress intensity factor, the submodel assumed that the crack would propagate. The crack growth rate was calculated using the SDFR model. A breached patch was assumed to have cracks with a size of 7.682 mm^2 [1.190 × 10^{-2} in^2] and spacing of 25 mm [0.98 in] (i.e., 6 cracks per patch). The output of the model was the time of waste package breach and the breach area. This output was provided to the Waste Form Degradation and Mobilization Model Component and the Engineered Barrier System Flow and Engineered Barrier System Transport Submodels.

For the Nominal Modeling Case, DOE calculated that waste packages were not breached by stress corrosion cracking in the closure weld (i.e., less than probability of 1 in 10^4) for approximately 150,000 years after repository closure, and within 1 million years, a mean of approximately 50 percent of waste packages were breached [SAR Figure 2.1-10(a)]. Of those breached waste packages, DOE calculated that the mean fraction of breached area to total waste package surface area was less than 10^{-5} over 1 million years (SAR Figures 2.1-13[b] and 2.1-15[b]). DOE calculated similar results for stress corrosion cracking of the closure weld in the Seismic Ground Motion Modeling Case, as shown in DOE (2009bj, Figures 1–4).

The model abstraction for stress corrosion cracking caused by impacts during seismic ground motion was implemented in the TSPA code in the Seismic Ground Motion Modeling Case, as outlined in SNL (2008ag, Section 6.6). In the Seismic Ground Motion Modeling Case, the residual stress threshold was sampled from a uniform distribution between 90 and 105 percent of the Alloy 22 at-temperature yield stress. Using the sampled residual stress threshold, DOE calculated the size of the waste package damaged area. The crack area density for the given damaged area was sampled from a uniform distribution, bounded by 3.27 × 10^{-3} and 1.31 × 10^{-2}. The product of the size of the damaged area and the crack area density gave the total open area of the stress corrosion cracking network. The output of the model was the time of waste package breach and the breach area. This output was provided to the Waste Form Degradation and Mobilization Model Component and the Engineered Barrier System Flow and Engineered Barrier System Transport Submodels.

For stress corrosion cracking induced by impacts during seismic ground motion, DOE calculated that a mean of approximately 10 percent of CSNF waste packages were breached within about 250,000 years of repository closure and for the codisposal waste packages, a mean of approximately 40 percent were breached within about 150,000 years of repository closure (DOE, 2009bj). For both waste package types, the fraction of the waste package surface consisting of open cracks was less than 10^{-3}, as shown in DOE (2009bj, Figures 7 and 8). DOE stated that the response of the respective waste package types is different because CSNF waste packages are generally not damaged by seismic ground motion until breached by another mechanism (e.g., stress corrosion cracking of the closure weld) that leads to degradation of the waste package internals. Conversely, DOE stated that codisposal waste packages are structurally weaker and can be damaged by seismic ground motion regardless of previous damage. For both waste package types, DOE stated that the probability of seismically induced stress corrosion cracking plateaus within 250,000 years after repository closure because drip shields collapse and impinge the waste packages.

NRC Staff's Review

The NRC staff reviewed the implementation and integration of the model abstraction for stress corrosion cracking of the waste package outer barrier in the postclosure performance assessment. The NRC staff finds that DOE has provided sufficient information for the NRC staff to understand how the conceptual model is implemented in the TSPA code and how the model inputs and outputs are integrated with other model components. The NRC staff finds that the model abstraction is consistent with the design features of the waste package, including materials of construction and dimensions given in SAR Section 1.5.2.7. Further, the NRC staff finds that, with respect to the parameters used in the model abstraction, DOE adequately justified the ranges of these parameters because the parameter values were obtained using standard and established experimental methods. Further, DOE accounted for uncertainty in the model abstraction by using ranges in the parameter values. Therefore, the NRC staff finds DOE's implementation of the model abstraction for stress corrosion cracking of the waste package outer barrier in the TSPA code acceptable because it would not overestimate the timing or underestimate the magnitude of radionuclide release to the accessible environment.

NRC Summary of Evaluation Findings for Stress Corrosion Cracking of the Waste Package Outer Barrier

The NRC staff reviewed DOE's model abstraction for stress corrosion cracking of the waste package outer barrier that was implemented in the TSPA code. The NRC staff finds that DOE used appropriate experimental tests and other independent technical literature to provide adequate support for its model abstraction. In addition, DOE appropriately identified and adequately considered features and processes such as stress corrosion crack initiation and propagation, crack opening area, and stress, and events such as seismic ground motion, that affect the waste package outer barrier capabilities for the initial 10,000 year period, and projected these features, events and processes beyond the 10,000 year post-disposal period through the period of geologic stability. Therefore, the NRC staff finds that DOE acceptably accounted for stress corrosion cracking of the waste package outer barrier in the TSPA model.

2.2.1.3.1.3.2.4 Waste Package Early Failure

In SAR Section 2.2.2.3, DOE defined early failure to be a through-wall penetration of the waste package caused by manufacturing- and handling-induced defects, at a time earlier than would be expected for a nondefective waste package. DOE assumed that a waste package undergoes early failure if it is emplaced in the repository with an undetected manufacturing- or handling-induced defect. On the basis of the processes associated with waste package manufacturing and handling, DOE calculated that the probability of waste package early failure is best represented in the TSPA code by a lognormal distribution with a mean of 1.13×10^{-4} per waste package and an error factor of 8.17, as shown in SNL (2007aa, Table 6-7). The NRC staff reviewed the adequacy of this probability distribution in SER Section 2.2.1.2.2.4. This section addresses the NRC staff's review of the implementation of this probability distribution in the TSPA code.

Waste Package Early Failure Conceptual Model

DOE's conceptual model for early waste package failure is that the waste package with an undetected manufacturing- or handling-induced defect completely fails (i.e., is removed as a barrier to the flow of water) at the time of repository closure (SAR Section 2.3.6.6.1). DOE selected this representation because there are uncertainties associated with the timing and extent of breach for defective waste packages and a completely degraded waste package at the time of repository closure will not overestimate the timing and underestimate the magnitude of radionuclide releases, as described in SNL (2007aa, Section 6.5.2). DOE concluded that this is a conservative representation of the early failure because the most likely consequence of improper waste package manufacturing or handling would be introduction of stress corrosion cracking, which tends to cause tight cracks that would limit the extent of radionuclide transport (SAR Section 2.3.6.6.4.1).

NRC Staff's Review

The NRC staff reviewed DOE's conceptual model for waste package early failure. The NRC staff finds that DOE attributed no barrier capability to the early failed waste packages related to manufacturing- or handling-induced defects. The NRC staff concluded, however, that consequences of early failure of waste packages would likely allow the waste package to maintain some barrier capability, which limits radionuclide releases. Because early failed waste packages in DOE's model have no barrier capability, the NRC staff finds that the model will not cause DOE to overestimate the timing or underestimate the magnitude of radionuclide releases. Therefore, the NRC staff finds DOE's conceptual model for waste package early failure analysis acceptable.

Abstraction and Integration

DOE's model abstraction for early failure of the waste package was implemented in the TSPA code in the waste package Early Failure Modeling Case, as detailed in SAR Section 2.4.2.1.5.2 and SNL (2008ag, Section 6.4.2). This modeling case uses most of the same modeling components and submodels as were implemented in the Nominal Modeling Case. In the Nominal Modeling Case, however, the waste package and Drip Shield Degradation Submodel provides the waste package and drip shield breached areas as a function of time to the Engineered Barrier System Flow and Transport Submodels and the Waste Form Degradation and Mobilization Model Components. In the waste package Early Failure Modeling Case, the waste package and Drip Shield Degradation Submodel was replaced with the waste package early failure mode, which simulated early failure by treating all patches on the failed waste package as breached at the time of repository closure.

DOE projected that the dose consequence of a waste package early failure would depend primarily upon the type of waste package, the environmental conditions at the waste package (e.g., temperature and relative humidity), and whether the waste package was in seepage conditions. Therefore, the waste package Early Failure Modeling Case calculated the dose consequence of a single early failure of a CSNF and codisposal waste package in each of the five percolation subregions with and without seepage conditions. The TSPA code then calculated the expected dose using the early failure probability [sampled from the distribution given in SNL (2007aa, Table 7-1), the distribution for the waste package type, and the seepage fraction for each percolation bin.

DOE calculated that there is approximately 55.8 percent probability of no waste package early failures, approximately 22.4 percent probability of one early failure, approximately 9.6 percent probability of two early failures, and approximately 12.3 percent probability of three or more early failures in the repository, as outlined in SNL (2008ag, Table 6.4-1). Waste package early failure makes a small contribution to DOE's calculated mean annual dose within approximately 20,000 years following closure {less than 10^{-6} Sv [10^{-1} mrem]}, with a declining contribution thereafter (SAR Figure 2.4-18).

NRC Staff's Review

The NRC staff reviewed the implementation of the waste package early failure model in the TSPA code. The NRC staff finds that DOE has provided sufficient information for the NRC staff to understand how the conceptual model is implemented in the TSPA code and how the model inputs and outputs are integrated with other model components. The NRC staff determined that the model abstraction was consistent with the design features of the waste package, including materials of construction and dimensions given in SAR Section 1.5.2.7. The NRC staff finds DOE's implementation of the waste package early failure model abstraction in the TSPA code acceptable, because it used standard and established methods, and therefore it would not overestimate the timing or underestimate the magnitude of radionuclide release to the accessible environment.

NRC Summary of Evaluation Findings for Waste Package Early Failure

The NRC staff reviewed DOE's model abstraction for early failure of the waste package outer barrier that was implemented in the TSPA code. The NRC staff finds that DOE provided adequate support for the model abstraction. In addition, DOE appropriately identified and adequately considered features and events such as waste package type, waste package early failure probability, that affect the waste package outer barrier capabilities for the initial 10,000 year period, and projected these features and events beyond the 10,000 year post-disposal period through the period of geologic stability. Therefore, the NRC staff finds DOE's accounting for waste package early failure in the TSPA model acceptable.

2.2.1.3.1.4 Evaluation Findings

The NRC staff has reviewed the SAR and other information submitted in support of the license application, which includes information required by 10 CFR 63.21(c)(3),(9),(10) and(15), and finds, with reasonable expectation, that the relevant requirements of 10 CFR 63.114and 63.342 are satisfied regarding the abstraction of degradation of engineered barriers in the TSPA model. In particular, the NRC staff finds that DOE

- Included appropriate data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the surface and subsurface from the site and the region surrounding Yucca Mountain, and provided adequate information on the design of the engineered barrier system to define parameters and conceptual models used in the performance assessment calculation, in compliance with 10 CFR 63.114(a)(1)

- Accounted for uncertainty and variability in the parameter values used to model degradation of engineered barriers, in compliance with 10 CFR 63.114(a)(2)

- Considered and evaluated alternative conceptual models that are consistent with currently available data and scientific understanding, in compliance with 10 CFR 63.114(a)(3)

- Provided a technical basis for the inclusion of features, events, and processes affecting degradation of engineered barriers, consistent with the limits on performance assessment in 10 CFR 63.342, and evaluated in sufficient detail those processes that would significantly affect repository performance, in compliance with 10 CFR 63.114(a)(4-6)

* Provided technical bases for the models of degradation of engineered barriers used in the performance assessment to represent the 10,000 years after disposal, in compliance with 10 CFR 63.114(a)(7)

* Used performance assessment methods for the period of geologic stability consistent with the methods used to demonstrate compliance for the initial 10,000 years following permanent closure, in compliance with 10 CFR 63.114(b)

* Included in the performance assessment for the period of geologic stability those FEPs used for the initial 10,000-year period, consistent with specific limits, in compliance with 10 CFR 63.342(c)

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CHAPTER 5

2.2.1.3.2 Mechanical Disruption of Engineered Barriers

2.2.1.3.2.1 Introduction

Safety Evaluation Report (SER) Section 2.2.1.3.2 evaluates the performance of the proposed Engineered Barrier System (EBS) that the U.S. Department of Energy (DOE) presented in its Safety Analysis Report (SAR), Section 2.3.4 (DOE, 2008ab). The design aspects of the EBS were described in SAR Sections 1.3.4 and 1.5.2, while the performance aspects were described in SAR Sections 2.1, 2.3.4, 2.3.5, 2.3.6, and 2.3.7. DOE stated that the following EBS features contribute to barrier performance: emplacement drifts, drip shields, waste packages, waste forms, waste form internals, waste package pallets, and emplacement drift inverts. According to DOE, the EBS features are designed to work together with the natural barriers to prevent or substantially reduce the release rate of radionuclides from the repository to the accessible environment. A disruption of the EBS components has the potential to affect their barrier performance. DOE anticipates that mechanical disruption of EBS components could generally result from external loads generated by accumulating rock rubble. Rubble accumulation can result from processes such as (i) degrading emplacement drifts due to thermal loads, (ii) time-dependent natural weakening of rocks, and (iii) effects of seismic events (vibratory ground motion or fault displacements). SER Section 2.2.1.3.2 evaluates the performance of the various EBS components under a reasonable range of anticipated loading conditions.

To estimate the timing and extent of rubble accumulation, rocks in the repository block need geologic characterization. DOE characterized the repository rock mass as consisting of two major rock types: lithophysal and nonlithophysal. Lithophysal rocks (approximately 85 percent of the repository emplacement area) are characterized as relatively more deformable rocks with low compressive strength because of the voids of varying sizes contained within the rock. The nonlithophysal rocks (approximately 15 percent) are characterized as hard, strong, and jointed rocks. According to DOE, these two rock types are expected to behave differently under thermal and seismic loads and thus require different modeling approaches to account for different modes of failure (e.g., rock blocks separating from the mass and falling due to gravity, gradual unraveling due to time-dependent weakening, or tensile failure during vibratory loads). On the basis of geologic mapping and testing, DOE categorized the lithophysal rocks into five categories according to their rock mass qualities (to represent the variability in mechanical properties). DOE has conducted laboratory and *in-situ* testing on small and large rock samples and developed a range of input parameters for the numerical models. DOE has presented several approaches to estimate the timing and extent of degradation, including numerical modeling results and the resulting rubble accumulation.

According to DOE, the functions of the drip shield are to prevent rocks from falling on the waste packages and to prevent water from contacting the waste package surface after emplacement when waste packages are still hot, thereby minimizing the potential for corrosion. The purpose of the waste package is to protect the waste form and isolate the radionuclides or slow down their rate of release to the accessible environment. To estimate the effects on timing and magnitude of radionuclide release, DOE analyses considered potential loads from seismic events and the resulting mechanical disruption of the EBS components. DOE considered gradual drift degradation due to thermal loads, time-dependent weakening, and seismic events as sources of generating loads from rubble accumulation on and around the drip shields.

However, DOE excluded the effects of drift degradation due to thermal loads and time-dependent weakening from its Total System Performance Assessment (TSPA) code. The NRC staff's review of DOE's technical bases for exclusion of features, events, and processes (FEPs) (FEP 2.1.07.02.0A, Drift Collapse) is presented in SER Section 2.2.1.2.1.3.2. The scope of this SER Section is limited to reviewing how DOE considered the effects of seismic disruption (i.e., vibratory ground motion and fault displacement) and used the results in the performance analysis.

The U.S. Nuclear Regulatory Commission (NRC) staff's review followed the guidance provided in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa), as supplemented by NRC (2009ab). YMRP Section 2.2.1 provides guidance to the NRC staff on applying risk information throughout the review of the performance assessment. The NRC staff used DOE's risk information that it derived from a review of DOE's treatment of multiple barriers, as appropriate. The NRC staff's review approach is to assess the DOE design and analyses of EBS components under anticipated demands generated by drift degradation due to seismic loads. For those cases in which the design capacities may be exceeded, the NRC staff examined the potential for continued functionality of the components under a range of anticipated conditions. On the basis of the risk insights developed, the NRC staff's review focuses primarily on the seepage barrier functionality of the drip shield and the potential for loads from accumulated rubble to be transferred onto the waste package. In considering the range of possible loads and temperature conditions that can be anticipated during the repository life, the NRC staff takes into account uncertainty and variability in (i) rock characterization data, (ii) laboratory and *in-situ* test results, (iii) modeling approaches and conceptualization of failure modes, and (iv) NRC staff's independent verifications.

2.2.1.3.2.2 Regulatory Requirements

The applicant is required to provide information specified in 10 CFR 63.21(c)(1)-(3), (9), (10), (15), and (19) that is related to abstraction of mechanical disruption of EBS components. The requirements in 10 CFR 63.114 (Requirements for Performance Assessment) and 10 CFR 63.342 (Limits on Performance Assessments) include postclosure performance assessment to demonstrate compliance with 10 CFR 63.113 (Performance Objectives for the Geologic Repository after Permanent Closure). Compliance with 10 CFR 63.113 is reviewed in SER Section 2.2.1.4.1.

The regulation at 10 CFR 63.114 requires, in part, that a performance assessment

- Include data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the surface and subsurface from the site and the region surrounding Yucca Mountain [10 CFR 63.114(a)(1)]

- Account for uncertainty and variability in the parameter values [10 CFR 63.114(a)(2)]

- Consider and evaluate alternative conceptual models [10 CFR 63.114(a)(3)]

- Provide technical bases for either the inclusion or exclusion of FEPs, including effects of degradation, deterioration, or alteration processes of engineered barriers that would adversely affect performance of the natural barriers, consistent with the limits on performance assessment in 10 CFR 63.342, and evaluate in sufficient detail those processes that would significantly affect repository performance [10 CFR 63.114(a)(4-6)]

- Provide a technical basis for the models used in the performance assessment to represent the 10,000 years after disposal [10 CFR 63.114(a)(7)]

The NRC staff's evaluation of DOE's inclusion or exclusion of FEPs is in SER Section 2.2.1.2.1. Requirements for performance assessment for the initial 10,000 years following disposal are specified in 10 CFR 63.114(a). Requirements for the performance assessment methods for the time from 10,000 years through the period of geologic stability, defined in 10 CFR 63.302 as 1 million years following disposal are in 10 CFR 63.114(b) and 10 CFR 63.342. These sections provide that through the period of geologic stability, with specific limitations, the applicant

- Use performance assessment methods consistent with the performance assessment methods used to demonstrate compliance for the initial 10,000 years following permanent closure [10 CFR 63.114(b)]

- Include in the performance assessment those FEPs used in the performance assessment for the initial 10,000-year period (10 CFR 63.342)

The requirements in 10 CFR 63.342(c)(1) pertain to the effects of seismic and igneous activity on the repository performance, subject to the probability limits in 10 CFR 63.342(a) and 10 CFR 63.342(b). Specific constraints on the seismic and igneous activity analyses are in 10 CFR 63.342(c)(1).

The NRC staff's review of the SAR and supporting information follows the guidance in the YMRP, NUREG–1804, Section 2.2.1.3.2, Mechanical Disruption of Engineered Barriers, (NRC, 2003aa), as supplemented by additional guidance for the period beyond 10,000 years after permanent closure (NRC, 2009ab). The YMRP acceptance criteria for model abstractions that provide guidance for the NRC staff's review of the applicant's abstraction of mechanical disruption of engineered barriers are

1. System description and model integration are adequate.
2. Data are sufficient for model justification.
3. Data uncertainty is characterized and propagated through the abstraction.
4. Model uncertainty is characterized and propagated through the abstraction.
5. Model abstraction output is supported by objective comparisons.

In its review of the SAR and supporting information, the NRC staff used a risk-informed approach and guidance provided by the Yucca Mountain Review Plan (YMRP), as supplemented by NRC (2009ab), for aspects of mechanical disruption of engineered barriers important to repository performance. The NRC staff considered all five YMRP acceptance criteria in its review of information provided by DOE. In the context of these criteria, only those aspects of the model abstraction that substantively affect the performance assessment results, as determined by the NRC staff, are discussed in detail in this SER Section. The NRC staff's determination is based both on risk information provided by DOE and on the NRC staff's review and independent analyses.

2.2.1.3.2.3 Technical Review

2.2.1.3.2.3.1 Seismic and Fault Displacement Inputs for Mechanical Disruption of Engineered Barriers

DOE investigated the geological, geophysical, and seismic characteristics of the Yucca Mountain region to obtain sufficient information to estimate how the site would respond to vibratory ground motions from earthquakes. In SAR Section 1.1.5.2, DOE provided its description of site seismology. DOE described its analysis of potential seismic hazards in SAR Section 1.1.5.2.4, the overall approach to developing a seismic hazard analysis for Yucca Mountain in SAR Section 2.2.2.1, and the conditioning (adaption or modification) of the ground motion hazard at Yucca Mountain in SAR Section 1.1.5.2.5.1. Additional information was provided in DOE responses to the NRC staff's request for additional information (RAI) in DOE (2009ab, Enclosure 19) and DOE (2009aq, Enclosures 6, 7, and 8), and the references cited therein.

The DOE overall approach to developing a seismic hazard analysis for Yucca Mountain, including fault displacement hazards as described in SAR Section 2.2.2.1, involves the following three steps:

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

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

3. Modify the conditioned PSHA results using site-response modeling. This accounts for site-specific rock material properties of the tuff, in and beneath the emplacement drifts, and the site-specific rock and soil material properties of the strata beneath the site.

Probabilistic Seismic Hazard Analysis Methodology

DOE conducted an expert elicitation on PSHA in the late 1990s (CRWMS M&O, 1998aa; BSC, 2004bp) on the basis of the methodology described in the Yucca Mountain Site Characterization Project (DOE, 1997aa). DOE stated that its PSHA methodology followed the guidance of the DOE–NRC–Electric Power Research Institute-sponsored Senior Seismic Hazard Analysis Committee (Budnitz, et al., 1997aa). In SAR Section 2.2.2.1.1.1, DOE concluded that the methodology used for the PSHA expert elicitation is consistent with NRC expert elicitation guidance, which is described in NUREG–1563 (NRC, 1996aa).

To conduct the PSHA, DOE convened two panels of experts as described in SAR Section 2.2.2.1.1.1. The first expert panel consisted of six 3-member teams of geologists and geophysicists (seismic source teams) that developed probabilistic distributions to characterize relevant potential seismic sources in the Yucca Mountain region. These distributions included location and activity rates for fault sources, spatial distributions and activity rates for background sources, distributions of moment magnitude and maximum magnitude, and site-to-source distances. The second panel consisted of seven seismology experts (ground motion experts) who developed probabilistic point estimates of ground motion for a suite of earthquake magnitudes, distances, fault geometries, and faulting styles. These point estimates incorporated random uncertainties that were specific to the regional crustal conditions of the western Basin and Range. The ground motion attenuation point estimates were then fitted to yield the ground motion attenuation equations used in the PSHA. The two expert panels were supported by technical teams from DOE, the U.S. Geological Survey, and Risk Engineering, Inc., which provided the experts with relevant data and information; facilitated the formal elicitation, including a series of workshops designed to accomplish the elicitation process; and integrated the hazard results.

According to the DOE–NRC–Electric Power Research Institute-sponsored Senior Seismic Hazard Analysis Committee (Budnitz, et al., 1997aa), the basic elements of the PSHA process are: (i) identification of seismic sources such as active faults or seismic zones; (ii) characterization of each of the seismic sources in terms of their activity, recurrence rates for various earthquake magnitudes, and maximum magnitude; (iii) development or incorporation of ground motion attenuation relationships to model the distribution of ground motions that will be experienced at the site when a given magnitude earthquake occurs at a particular source; and (iv) incorporation of the inputs into a logic tree to integrate the seismic source characterization and ground motion attenuation relationships, including associated uncertainties. According to the Budnitz, et al. (1997aa) methodology, each logic tree pathway represents one expert's weighted interpretations of the seismic hazard at the site. The computation of the hazard for all possible pathways results in a distribution of hazard curves that is representative of the seismic hazard at a site, including variability and uncertainty.

NRC Staff's Review

The NRC staff reviewed DOE's PSHA methodology described in SAR Sections 1.1.5.2.4 and 2.2.2.1.1 using the guidance provided in the YMRP, NUREG–1563 (NRC, 1996aa), and NUREG–2117 (NRC, 2012aa). The NRC staff also evaluated the DOE PSHA development to ensure that it included the four basic elements described in Budnitz, et al. (1997aa). In addition, the NRC staff observed all expert elicitation meetings and reviewed summary reports of those meetings as they were produced. On the basis of the NRC staff's evaluation with respect to Budnitz, et al. (1997aa) and the NRC staff's direct observations of the expert elicitation process, the NRC staff finds that DOE's elicitation for the PSHA is consistent with the framework for conducting an expert elicitation described in NUREG–1563 (NRC, 1996aa) and NUREG–2117 (NRC, 2012aa). Therefore, the NRC staff finds that the DOE implementation of the PSHA expert elicitation is acceptable to develop estimates of seismic hazards for use in DOE's TSPA.

Probabilistic Seismic Hazard Analysis—Input Data and Interpretations

During the expert elicitation, DOE's seismic source teams considered a range of information from many resources including DOE, the U.S. Geological Survey, project-specific Yucca Mountain studies, and information published in scientific literature. This information included (i) data and models for the geologic setting as summarized in BSC (2004bp); (ii) seismic sources and seismic source characterization, including earthquake recurrence and maximum magnitude (BSC, 2004bp); (iii) historical and instrumented seismicity, as described in

CRWMS M&O (1998ab, Appendix G); (iv) paleoseismic data (Keefer, et al., 2004aa); and (v) ground motion attenuation (Spudich, et al., 1999aa). DOE also supported the PSHA with a broad range of data, process models, empirical models, and seismological theory (CRWMS M&O, 1998ab). The expert panels built their respective inputs to the PSHA on the basis of this information and information presented to the experts during the elicitation meetings (CRWMS M&O, 1998ab). The resulting set of hazard curves was intended to provide DOE with sufficient representation of the seismic hazard for use in the TSPA analysis.

DOE expressed the PSHA curves in increasing levels of ground motion as a function of the annual probability that the ground motion will be exceeded. These curves were developed for bedrock conditions with a mean shear wave velocity (V_s) of 1,900 m/sec [6,233 ft/sec]. Such rocks are located adjacent to Yucca Mountain, as described previously in the PSHA methodology subsection of this SER section. Estimates of uncertainty in the hazard curves are also included (see SAR Figure 1.1-74, e.g., hazard curves). The SAR provided PSHA results on horizontal and vertical components of peak acceleration (defined at 100 Hz); spectral accelerations (SAs) at frequencies of 0.3, 0.5, 1, 2, 5, 10, and 20 Hz; and peak ground velocity (PGV).

NRC Staff's Review

The NRC staff reviewed DOE's PSHA input data and interpretations as described in SAR Sections 1.1.5.2 and 2.2.2.1.1. The NRC staff concludes that DOE adequately developed the geological, geophysical, and seismological information necessary to support the expert elicitation. This conclusion is based in part on the NRC staff's evaluations of this information in NUREG–1762 (NRC, 2005aa). In NUREG–1762, the NRC staff found that the existing DOE information was consistent with site conditions at Yucca Mountain. This conclusion of adequacy is also based on the NRC staff's first-hand knowledge of the geology and seismic characteristics of the Yucca Mountain region, which includes more than a decade of independent geological and geophysical research and study (e.g., Ferrill, et al., 1999ab, 1996aa; Stamatakos, et al., 2007aa, 1998aa,1997ab; Waiting, et al., 2003aa; Gray, et al., 2005aa). The NRC staff also finds that the resulting suite of ground motion hazard curves; horizontal and vertical components of peak acceleration (defined at 100 Hz); spectral accelerations at frequencies of 0.3, 0.5, 1, 2, 5, 10, and 20 Hz; and PGV is acceptable because it is consistent with the NRC guidance outlined in Regulatory Guide 1.165 (NRC, 1997ab) and Regulatory Guide 3.73 (NRC, 2003ae).

The NRC staff also reviewed additional geological, geophysical, and seismological information in Wernicke, et al. (2004aa) and Hanks, et al. (2013aa), which was developed after the DOE elicitation was performed. Wernicke, et al. (2004aa), provided updates to the Global Positioning Satellite (GPS) data for Yucca Mountain to include data from a continuously operating network. These results showed that the anomalously large crustal strain rates, indicated by GPS results (Wernicke, et al., 1998aa) considered in the PSHA, were in part transient strains associated with the 1992 Little Skull Mountain earthquake and not indicative of increased seismic hazard at the site. Results in Hanks, et al. (2013aa) are based on two studies: one on the physical limits of ground velocities that the lithology at Yucca Mountain could have experienced since deposition based on the physical limits of rock strength; and a second detailed analysis of the age, distribution, and geometries of precariously balanced rocks along the steep hill slopes in the Yucca Mountain region. Both the physical limits and precarious rock studies in Hanks, et al. (2013aa) indicate that the upper limits on the amplitudes of earthquake ground motions that could have occurred in the geological past at Yucca Mountain are smaller than those used by DOE. These results suggest that extremely large ground motions at low annual exceedance probabilities in the DOE PSHA are conservative. These new results, therefore, further support the NRC staff's conclusion that DOE's PSHA input data and interpretations are acceptable.

Conditioning of Ground Motion Hazard

DOE provided in SAR Section 1.1.5.2.5.1 the conditioning of ground motion hazard at the reference bedrock outcrop where the PSHA was developed. Since completion of the PSHA in 1998, several studies and reports, including ones from the NRC staff (NRC, 1999aa), the Nuclear Waste Technical Review Board Panel on Natural System and Panel on Engineered Systems (Coraddini, 2003aa), and DOE itself (BSC, 2004bj), questioned whether the very large ground motions the PSHA predicted at low annual exceedance probabilities (below ~10^{-6}/yr) were physically realistic. These ground motion values are well beyond the limits of existing earthquake accelerations and velocities from even the largest recorded earthquakes worldwide. They are deemed physically unrealizable because they require a combination of earthquake stress drop, rock strain, and fault rupture propagation that cannot be sustained without wholesale fracturing of the bedrock which is not observed at Yucca Mountain (Kana, et al., 1991aa).

For Yucca Mountain, however, the seismic hazard curves were extrapolated to estimate ground motions with annual exceedance probabilities as low as 10^{-8} (SAR Section 1.1.5.2.5.1). At these low probabilities, the seismic hazard estimates are driven by the tails of the untruncated Gaussian distributions (the tail is not defined by the data, but by the assumed distribution) of the input ground motion attenuation models (Bommer, et al., 2004aa). As Anderson and Brune (1999aa) pointed out, overestimates of the hazards may also arise because of the way in which uncertainty in ground motion attenuation from empirical observations or theory is distributed between aleatory and epistemic uncertainties.

To account for these large ground motions, DOE modified or conditioned the hazard using both a shear-strain-threshold approach and an extreme-stress-drop approach, as described in SAR Section 1.1.5.2.5.1. The applicant used these two independent methods for conditioning the PSHA results to make the seismic hazards consistent with the geologic setting of Yucca Mountain. The first method in the SAR used geological observations at the repository level to develop a limiting distribution on shear strains experienced at Yucca Mountain (BSC, 2005aj). The second method in the SAR used expert judgment (BSC, 2008bl) to develop a distribution of extreme stress drop in the Yucca Mountain vicinity. The distribution is based on available data (stress drop measurements and apparent stress drops from laboratory experiments) and interpretations. As discussed in SAR Section 1.1.5.2.5.1 and BSC (2008bl), the applicant conducted conditioning using the shear-strain-threshold and extreme-stress-drop approaches in series (the combined conditioning) because these two methods are independent.

Rather than reconvene the PSHA expert elicitation and redo the hazard analysis, DOE chose to treat the issue as part of the ground response analysis. Accordingly, DOE's second step in developing ground motion inputs for postclosure analyses, after the development of PSHA, was to condition the ground motion hazard. This second step in the three-step DOE process includes information on the level of extreme ground motion that is consistent with the geological setting of Yucca Mountain. Conditioning of ground motion hazard is a unique study developed for the Yucca Mountain project.

The unconditioned hazard curve DOE developed, which is the annual probability of exceedance (APE) as a function of ground motion, is convolved with the distribution of extreme ground motion for the reference bedrock outcrop to produce the conditioned ground motion hazard of the same bedrock outcrop. The impact of conditioning at higher probabilities is less significant and increases as the probability of exceedance decreases (i.e., annual probabilities of exceedance of 10−5, 10−6, 10−7, and 10−8) (SAR Section 1.1.5.2.5.1). SAR Figures 1.1-79 and 1.1-80 compared the unconditioned and conditioned peak ground accelerations (PGAs) and PGV mean hazard curves for the reference bedrock outcrop.

For the extreme stress-drop approach, BSC (2008bl, Appendix A) outlined the workshops, which included presentations, discussions, and assessments that were conducted to develop the expert judgment. The stress drop data from the United States and other countries were used in the expert elicitation. The parameter variability involved in the empirical ground motion attenuation relationship and numerical simulations of ground motions that the experts relied on was included in the conditioning. Variability in velocity profile, stress drop, source depth, and kappa (the site- and distance-dependent parameter representing the effect of intrinsic attenuation of the wave field as it propagates through the crust from source to the receiver) were considered in the modeling to map the stress drop into ground motion distribution.

In response to NRC staff RAIs (DOE, 2009aq), DOE provided information explaining its application of the two methods in series where the output of the extreme-stress-drop conditioning becomes the input for the shear-strain-threshold approach. In the RAI responses, DOE also clarified and updated the formulations for the two conditioning methods, as described in BSC (2008bl, Appendix A).

NRC Staff's Review

The NRC staff reviewed the applicant's methods for conditioning PSHA results in SAR Section 1.1.5.2.5.1 and the applicant's responses to NRC staff RAIs (DOE, 2009aq) to evaluate whether the applicant's two independent conditioning methods are adequate. The NRC staff finds that the shear-strain-threshold approach is adequate because it follows appropriate mechanical, material, and seismological principles and is based on laboratory rock mechanics data and corroborated by numerical modeling.

The NRC staff finds that the extreme stress drop method is adequate because it is supported by observations from worldwide earthquake recordings (SAR Section 1.1.5.2.5.1). These earthquake observations were used by the applicant's experts to develop limits on stress drop.

The NRC also finds acceptable that the DOE applied these two methods in series because, as DOE described in SAR Section 1.1.5.2.5.1 and its RAI response (DOE, 2009aq), they are independent from each other. The NRC staff therefore concludes that applying both in series would not duplicate or double count their respective effects on conditioning the hazard curve. Moreover, NRC staff notes that the shear-strain-threshold approach has less of an effect on reducing the hazard as compared to the extreme-stress-drop approach. For example, for an APE of 1×10^{-8}, the shear-strain-threshold conditioned PGV hazard is reduced from 1,200 cm/sec to about 1,100 cm/sec [472 to 433 in/sec] or about 10 percent; the stress-drop-conditioned PGV hazard is reduced from 1,200 cm/sec to about 480 cm/sec [472 to 189 in/sec] or about 60 percent, as identified in BSC (2008bl, Section A4.5.1).

The NRC staff also finds that the final conditioned ground motion levels at very low APE are conservative when compared with the observed worldwide strong motion data, which include records from earthquakes much greater than those expected in the Yucca Mountain region. DOE assumed that the tectonic setting, and therefore the stress drops of earthquakes from the existing faults at Yucca Mountain are not going to change significantly during the next

1 million years. The NRC staff concludes that this assumption is reasonable given the basic tectonics in the Yucca Mountain region and provides the basis for the conditioning at very low annual probabilities of exceedance.

Summary of NRC Staff's Evaluation of Probabilistic Seismic Hazard Analysis

The NRC staff reviewed SAR Sections 1.1.5.2 and 2.2.2.1 and DOE's responses to the NRC staff's RAIs and finds the methodologies, input data, interpretation of the PSHA, and conditioning of the PSHA results for the Yucca Mountain site to be acceptable. The NRC staff finds that DOE used appropriate methods for relying on the collective judgment of established experts by following an acceptable procedure to elicit and document the experts' conclusions. In addition, the NRC staff finds that DOE supported the expert elicitation program with sufficient technical and scientific information.

New information about the seismic hazards at Yucca Mountain published since DOE completed its expert elicitation [published in Hanks, et al. (2013aa)], was reviewed by the NRC staff and found to further support the NRC staff's conclusion that the PSHA is acceptable. This new information suggests that the DOE PSHA provided in the SAR is conservative at low annual exceedance probabilities.

2.2.1.3.2.3.1.1 Seismic Site-Response Modeling

To address the effects of earthquakes during the postclosure period, DOE performed a site-response analysis, which incorporates the effects of the upper rock and soil layers on the input ground motion at the reference rock (the conditioned ground motion hazards discussed previously).

Site-Response Modeling

In SAR Section 1.1.5.2.5.2, DOE discussed how the various types and thicknesses of rocks, alluvium, and soils that comprise the site would likely respond to earthquake ground motions. The results of site-response modeling include understanding and quantifying the amplification or damping of ground motion and how much the vertical-to-horizontal motion ratio varies from place to place.

NRC Staff's Review

The NRC staff reviewed DOE's approach to site-response modeling using the guidance in NUREG/CR–6728 (McGuire, et al., 2001aa) and YMRP Section 2.1.1.3.3. The NRC staff finds that DOE input ground motion from the conditioned PSHA results for assessing site response in its modeling analyses are appropriate because DOE followed standard practice in NRC Regulatory Guidance 1.165 (NRC 1997ab) and NUREG/CR–6728. Specifically, the NRC staff concludes that DOE's adoption of Approach 3 from NUREG/CR–6728 is appropriate because it is the most accurate method available and is recommended by NUREG/CR–6728. The NRC staff also concludes that the two frequency ranges (1–2 and 5–10 Hz) used in the calculations of input control motions are acceptable because they are consistent with NRC guidance provided in NRC Regulatory Guide 1.165 (NRC, 1997ab, Appendix C). Thus, the NRC staff finds DOE's site response model is acceptable.

Ground Motion Inputs (Site Profiles, Hazard Curves, Earthquake Time Histories)

DOE provided ground motion inputs developed for the repository block in SAR Section 1.1.5.2.6. For the site surface, 52 combinations of site properties were evaluated in the site-response modeling (SAR Section 1.1.5.2.6.1). These combinations were from two base case velocity profiles (south and northeast of the Exile Hill Fault splay), two base case sets of dynamic material property curves for tuff and alluvium separately, four values of alluvium thickness northeast of the fault splay, and three values of alluvium thickness south of the fault splay (resulting in a total of seven combinations). Each combination incorporated aleatory variability by averaging the amplification factors from 60 randomized velocity profiles and dynamic material property curves.

DOE combined the seven combinations of alluvium and tuff hazard curves into two sets: the northeast and south fault splay sets. The four and three combinations of hazard curves for four and three alluvium thicknesses were enveloped separately for south and northeast of the fault splay. These two sets of hazard curves were enveloped again to produce mean horizontal and vertical hazard curves (BSC, 2008bl). The final mean horizontal and vertical hazard curves for PGAs; 0.05, 0.1, 0.2, 0.5, 1.0, 2.0, and 3.3 seconds spectral acceleration (SA); and PGV were provided in BSC (2008bl, Figures 6.5.2-34 to 6.5.2-42) for the surface facilities area and BSC (2008bl, Figures 6.5.3-9 to 6.5.3-16) for the repository block. The data for these plots are identified in BSC (2008bl, Section 6.5.2.2).

The repository block time histories for postclosure analyses were developed differently for APE of 10^{-5}, 10^{-6}, and 10^{-7} (SAR Section 1.1.5.2.6.2), where 17 sets of time histories were developed: one horizontal (H1) component of each seed time history was scaled according to the PGV from site-response modeling and the other two components were scaled to maintain the inter-component variability of the seed time history (SAR Section 1.1.5.2.6.2).

NRC Staff's Review

The NRC staff reviewed DOE's ground motion inputs including site profiles, hazard curves, and earthquake time histories. Based on its review of this information, the NRC staff finds acceptable the processes and procedures DOE used to develop site-specific hazard curves, time histories, strain-compatible soil properties for the site, and the ground motions for postclosure analyses for the following reasons. DOE used an averaging process to account for the data (velocity profiles and dynamic material properties) and site-response model uncertainties and an enveloping process to accommodate the alluvium thickness change (spatial variability) when it developed the hazard curves. Then DOE followed the recommended (McGuire, et al., 2001aa) routine procedures in engineering seismology for ground motion inputs. The strain-compatible soil properties are the products of the previously described site-response analysis.

2.2.1.3.2.3.1.2 Fault Displacement Hazard Analysis

Fault displacement (the relative displacement between opposite sides of a fault) is a potential hazard to the underground facility because it could damage or shear the emplacement drifts and/or waste packages, trigger rockfall within the drifts, degrade drift walls and ground-support systems, and degrade other EBS components. These hazards might affect the postclosure performance of the engineered barriers.

Probabilistic Fault Displacement Hazard Analyses—Input Data and Interpretations

DOE's PFDHA integrated two data types: (i) known and/or documented faulting activity consisting of measurements of regional and local earthquakes and measurements of fault displacements within the last ~1.8 million years, and (ii) inferred potential faulting activity on the basis of analysis of mapped geological faults, overall tectonic setting, and regional estimates of ongoing crustal strain. DOE analyzed 100 earthquakes in the Basin and Range region to determine the relationships among the amounts and patterns of both principal and distributed fault displacements, the minimum magnitude at which an earthquake may produce surface faulting, and the maximum magnitude at which an earthquake does not displace the surface.

For the largest mapped faults at Yucca Mountain, the probabilistic fault displacement hazard curves were largely based on the same detailed paleoseismic and earthquake data used to characterize these faults as potential seismic sources. The expert elicitation relied on both anecdotal evidence and expert judgment to develop conceptual models of distributed faulting and to estimate the probabilities of secondary faulting of smaller faults and fractures in the repository (Youngs, et al., 2003aa; CRWMS M&O, 1998aa).

The expert elicitation teams used two methods to generate fault displacement hazard curves, as applied in the PFDHA: the displacement approach and the earthquake approach. The displacement approach uses fault-specific data, such as cumulative displacement, fault length, paleoseismic measurements from fault trench studies, or data from records of earthquakes correlated with the known seismogenic faults. The displacement approach relies on direct observational evidence of faulting. The experts derived fault displacement and displacement probability over time directly from (i) paleoseismic displacement and recurrence rate data, (ii) geologically derived slip rate data, or (iii) scaling relationships that relate displacement to fault length and cumulative fault displacement.

The earthquake approach relates the frequency and magnitude of the faults' slip events to the frequency and magnitude of earthquakes on the seismic sources, as they were defined in the seismic-source models in the PSHA (CRWMS M&O, 1998aa). The earthquake approach uses earthquake recurrence models from the seismic hazard analysis. For this approach, the experts assessed three probabilities: (i) the probability that an earthquake will occur; (ii) the probability that this earthquake will produce surface rupture on the source fault; and (iii) the probability that the earthquake will produce distributed surface displacements.

DOE chose the following nine sites around Yucca Mountain as demonstration sites of the application of the PFDHA, as identified in SAR Chapter 1, Table 1.1-67, p. 1.1-304: (i) Bow Ridge fault, (ii) Solitario Canyon fault, (iii) Drill Hole Wash fault, (iv) Ghost Dance fault, (v) Sundance fault, (vi) an unnamed fault west of Dune Wash, (vii) a location 100 m [328 ft] east of the Solitario Canyon fault, (viii) a location between Solitario Canyon fault and Ghost Dance fault, and (ix) a location within Midway Valley. These demonstration sites were selected to represent a range of faulting and related fault deformation conditions at the site, including large block-bounding faults, such as the Solitario Canyon and Bow Ridge faults; smaller mapped faults within the repository footprint, such as the Ghost Dance fault; and unmapped minor faults near the larger faults, fractured tuff, and intact tuff.

Results of the PFDHA (CRWMS M&O, 1998aa) show that, except for the Bow Ridge and Solitario Canyon faults, mean fault displacements are less than 1 m [3.28 ft] over the next 10 million years (SAR Table 2.2-15). Mean displacements for the demonstration sites within the current repository footprint [demonstration sites (v), (vii), and (viii) as identified in the previous paragraph] do not exceed 0.40 m [1.3 ft] in 10 million years. For a 10,000-year period, the mean displacements were calculated to be less than 0.01 m [0.03 ft] for all 9 demonstration sites (SAR Table 1.1-67).

Individual fault displacement hazard curves were developed to characterize fault displacements at each of the nine demonstration sites. These fault displacement hazard curves are analogous to seismic hazard curves, in which increasing levels of fault displacements are computed as a function of the annual probability that those displacements will be exceeded. Example fault displacement curves for the nine demonstration sites were provided in SAR Figure 2.2-13.

NRC Staff's Review

The NRC staff evaluated DOE's input to the PFDHA in the SAR and supporting documents. Specifically, the NRC staff evaluated the applicant's information regarding site characterization input data and interpretations to the PFDHA in SAR Section 1.1.5.1, references therein, and supporting documents. The NRC staff also conducted an independent analysis using slip tendency (Morris, et al., 1996aa) of faults within the Yucca Mountain region (e.g., Morris, et al., 2004aa). Based on the NRC staff's professional experience and knowledge gained from its own independent field, laboratory, and natural analog studies, the NRC staff finds that the input data to the PFDHA and the resulting interpretations are appropriate because (i) a range of geological and seismological information was considered by the DOE experts and the expert elicitation process allowed the range of interpretations about fault displacement in the scientific community to be evaluated by the experts (the NRC staff evaluates DOE's expert elicitation process in SER Section 2.5.4), (ii) the fault displacement and earthquake approaches used by the experts to interpret the data and develop the fault displacement curves are acceptable because they are consistent with seismological theory and supported by geological observations, and (iii) the interpretations made by the applicant are consistent with the NRC staff's independent evaluations of faulting (Ferrill and Morris, 2001aa; Dunne, et al., 2003aa; Ferrill, et al., 1999ab; Stamatakos, et al., 2000aa; NRC, 2005aa). On the basis of the NRC staff's review, the NRC staff finds acceptable DOE's methodology, input data, and interpretations of the probabilistic fault displacement hazard analyses.

2.2.1.3.2.3.2 Fault Displacement Considerations in TSPA

This section reviews the information provided in SAR Section 2.3.4.3 and selected references to evaluate the adequacy of the conceptual model of seismic fault displacement. DOE considered seismic fault displacement as one of the modeling cases in the TSPA. DOE assumed that fault displacement occurred concurrently with the ground motion during a low probability seismic event (SAR Section 2.3.4.5.5.1). DOE also considered that only the waste packages located directly above faults were subject to damage from fault displacement. DOE expected the dose related to fault displacement to be a small fraction of the total dose for the seismic scenario class because damage from fault displacement affected a small fraction of the EBS and damage occurred only for events with very low exceedance frequencies. DOE calculated the dose contribution from the seismic fault displacement on the basis of simplified calculations (SAR Section 2.4.2.2.1.2.2.2). The abstraction assumed waste package damage when fault displacement exceeded the available waste package clearance. To evaluate fault displacement, DOE assumed: (i) the faults are perpendicular to the drift axis with the displacement being vertical; (ii) the fault displacement occurs at a discrete plane, creating a sharp discontinuity; and (iii) clearances are based on emplacement drifts that are fully collapsed at the time of the seismic event. (SAR Section 2.3.4.5.5.2.1.1)

The potentially relevant FEPs in DOE's TSPA model were listed in SAR Table 2.2-1. In this abstraction, DOE evaluated and included FEP 1.2.02.03.0A, Fault Displacement Damages EBS Components.

Consideration of Clearance

DOE analyzed the clearances between the EBS components for intact and failed drip shield scenarios (SAR Section 2.3.4.5.5.2.1). For intact drip shield configurations, DOE defined the clearance as the interior height of the drip shield less the outside diameter of the waste package outer corrosion barrier without a pallet, to account for part of the substantial movement of the rubble (SAR Section 2.3.4.5.5.2.1.1). On the basis of this simplification, DOE used a maximum allowable displacement with drift collapse and an intact drip shield that varied from 67.3 to 96.9 cm [26.5 to 38.15 in] (SAR Table 2.3.4-52), according to the type of waste package.

NRC Staff's Review

The NRC staff compared the values of DOE's clearances in SAR Table 2.3.4-52 to the EBS geometry presented in SAR Figure 2.3.4-53 that showed the distances between the top of the waste packages and bottom of the drip shields {35.6 to 68.6 cm [14 to 27 in]}. Additionally, the NRC staff considered the potential for the rubble to accommodate some of the fault displacement through compaction. On the basis of a confirmatory calculation, the NRC staff estimates that approximately 72–186 cm [28.3–73.2 in] of fault displacement can be accommodated through rubble compaction and the distance between the top of the waste package and the bottom of the drip shield crown. The maximum allowable displacements used by DOE in the TSPA range between 67.3–96.9 cm [26.49–38.14 inches] for an intact drip shield condition and 43.7–51.1 cm [17.2–20.11 in] for the failed drip shield condition (i.e., a range of approximately 17 to 38 inches for both failed and intact conditions) as presented in SAR Tables 2.3.4-52 and 2.3.4.53, respectively. The NRC staff finds the values of maximum allowable displacements DOE used are adequate for their intended use because the NRC staff's confirmatory calculation based on design geometry shows that a larger range of fault displacements can be accommodated than the range DOE used in the TSPA.

For failed drip shield configurations, DOE stated that no free space existed between the top of the waste package and bottom of the drip shield. DOE concluded that rubble movement will accommodate some amount of fault displacement due to rubble consolidation (SAR Section 2.3.4.5.5.2.1.2). DOE stated that fault displacement had to exceed one-quarter of the outer diameter of the outer corrosion barrier to cause waste package failure {from 43.7 to 51.1 cm [17.2 to 20 in]} (SAR Tables 2.3.4-50 and 2.3.4-53). The NRC staff finds this approach adequate on the basis of a simplified, bounding calculation DOE presented in SNL Section 6.11.1.2 (2007ay) that is conservative. This approach is conservative because DOE showed that the potential pore space available within the accumulated rubble would accommodate 0.5–1.6 m [1.64–5.25 ft] of displacement under a range of possible estimates of rubble porosity (Porosity is a measure of void space available in a given volume of rubble.)

Expected Movements and Number of Impacted Waste Packages

DOE stated that the probability of events exceeding 0.1 cm [0.039 in] of displacement in the repository block was 10^{-5} per year (SAR Section 2.3.4.5.5.1). DOE characterized the subsurface geologic repository operations area and determined that few faults were capable of producing movements greater than calculated clearances for events with a probability of exceedance of 10^{-8} per year (SAR Table 2.3.4-55). SAR Table 2.3.4-59 showed that less than

2 percent of the waste packages can potentially be impacted by a seismic faulting event with an annual exceedance frequency of 1×10^{-8}/yr to 3×10^{-8}/yr. To mitigate the potential risk of faulting that could cause mechanical damage to the waste packages, DOE stated that waste packages would be placed 60 m [196.85 ft] from known, major faults (SAR Table 1.9.9, Design Control Parameter 01-05).

NRC Staff's Review

The NRC staff compared the probability, location, and magnitude of potential seismic fault displacement events used in DOE's PSHA (reviewed in SER Section 2.2.1.3.2.3) to the values presented for use in the TSPA model (SAR Table 2.3.4-55) and found the values were consistent. On the basis of the low probability of occurrence, the limited number of faults that could impact the waste package via fault displacement, and the 60-m [196.85-ft] offset from the location of known faults capable of impacting the waste packages, the NRC staff finds that DOE did not underestimate the number of waste packages that could be potentially impacted by seismic faulting events.

Damage to the EBS

DOE sampled a uniform distribution of open areas of waste packages to model the open area of a waste package failed by fault displacement. This distribution has a lower bound of 0 m^2 [0 ft^2] and an upper bound equal to the area of the waste package lid (SAR Section 2.3.4.5.5.4). DOE stated that the areas of the lids for the transportation, aging, and disposal (TAD) and co-disposal (CDSP) waste package groups are 2.78 and 3.28 m^2 [30 and 35 ft^2], respectively. (These areas were calculated using the waste package diameters provided in SAR Table 2.3.4-50.)

NRC Staff's Review

The NRC staff finds that the breached area distribution appropriately bounds the potential breach area expected in the fault displacement modeling case based on the following considerations. Given the uncertainty in the magnitude of displacement, the modeled breach area bounds the potential breach area, which could range from a plastically deformed waste package (with no open area) to one that has been completely sheared or sliced in half. Additionally, the average waste package is deemed to provide no barrier capability to seepage when approximately 4 percent {~1.5 m^2 [~16 ft^2]} of the surface area is breached. Thus, the waste package barrier's capability to prevent water from contacting the waste form decreases proportionally. (This aspect of DOE's performance assessment is evaluated in SER Section 2.2.1.3.3.3.3).

To determine the impact of seismic faulting on drip shield failure, DOE assumed that the drip shield fails at the instant the underlying waste package is breached. The NRC staff finds this simplification acceptable because the magnitude of a faulting event required to damage a waste package would damage the overlapping drip shield. The NRC staff also finds the loss of drip shield barrier capability appropriately bounding, in that the drip shield retains no barrier capability.

Summary of NRC Staff Evaluation of Fault Displacement

The NRC staff finds that DOE adequately represented the fault displacement modeling case in the TSPA analysis because

- The clearances are reasonably bounding considering the potential for rubble compaction.

- The expected fault displacements are consistent with the site characterization data.

- The impact on the EBS appropriately bounds the potential consequences.

2.2.1.3.2.3.3 Seismically Induced Drift Degradation

The NRC staff's evaluation in this SER section focuses on DOE's assessment of potential drift degradation due to seismic ground motions and use of the information to assess potential mechanical disruption of the engineered barriers. The NRC staff reviewed DOE's information in SAR Section 2.3.4.4 and supporting documents (BSC, 2004al) that describe potential seismically induced degradation of emplacement drifts after permanent closure. DOE's information included estimates of the amount of rubble accumulation in drifts, drip shield loading due to rubble, sizes of individual blocks that may strike the drip shield during a seismic event, and the associated impact velocity and location of the impact on the drip shield. The NRC staff's evaluation focuses on the potential occurrence of rubble loading that is large enough to damage the drip shield and the time of occurrence of such rubble loading. The time of occurrence is important because the NRC staff's review (SER Section 2.2.1.3.1) suggests that mechanical damage of the drip shield during approximately the first 12,000 years after repository closure could expose the waste packages to aggressive chemical conditions that may support localized corrosion. DOE estimated the potential for rubble accumulation in the excavated drifts through process-level analyses of the effects of seismic ground motions on drift degradation. DOE chose the analysis approach by considering the mechanical behavior of two types of rocks (i.e., lithophysal and nonlithophysal) that constitute the repository block (SAR Section 2.3.4.4.8.3). The NRC staff's review of DOE's rock characterization information relevant to drift degradation modeling is summarized in SER Section 2.2.1.2.1.3.2.

Process-Level Modeling of Drift Degradation Due to Seismic Events

According to DOE, the mechanical deformation of the nonlithophysal rock mass will be controlled predominantly by movements of rock blocks along existing fractures. DOE analyzed drift degradation in nonlithophysal rock by modeling motions of rock blocks on surfaces formed by existing fractures, as described in SAR Section 2.3.4.4.4. DOE used the analyses to estimate the characteristics of individual rock blocks that may strike the drip shields during a seismic event (SAR Table 2.3.4-19) and the volumes of rubble that may accumulate in the drifts (SAR Tables 2.3.4-20 and 2.3.4-24). The potential for rock block impacts exists only in nonlithophysal areas (approximately 15 percent of the emplacement drifts). DOE considered the failure mechanism by rock block impact and excluded it from the TSPA model (excluded FEP 1.2.03.02.0B as a result of low consequence).

For the lithophysal rock mass, DOE indicated that mechanical deformation of the rock will consist predominantly of rock material deformations aided by lithophysae and a high density of existing small-scale fractures. As described in SAR Section 2.3.4.4.5, DOE analyzed drift degradation in lithophysal rock by modeling potential fracturing of the rock through formation of new fractures and movement on existing fractures as dictated by the rock stress. DOE used the analyses to estimate potential rubble accumulation for drift sections in lithophysal rock but applied the results of the analyses to the entire repository. DOE stated that this approach would be bounding.

To assess potential drift degradation in lithophysal rock, DOE used a model that focuses on estimating the rock mass volume that could break up along failure surfaces determined by the effects of the prevailing stress field on a diffuse network of small “incipient” fractures. DOE obtained the estimates by modeling the rock mass as an assemblage of polygonal blocks generated randomly using the Voronoi tessellation model, as identified in SAR Section 2.3.4.4.5.3 and BSC Sections 6.4.2.1 and 7.6.1 (2004al). The individual blocks can deform elastically or slide or separate at block contact surfaces. The blocks are initially attached together at contact surfaces and may slide or separate if the contact resistance is overcome by the prevailing stress. Thus, the contacts represent incipient fractures that could allow the blocks to detach from the assemblage if the prevailing stress permitted such detachment. Detached blocks could fall as a result of gravity or seismically induced force. DOE stated in BSC (2004al, Section 6.4.2.1) that the model could simulate rock deformation, stress changes, rock fracturing or breakage, and free fall of broken rock blocks. DOE explained that mechanical behavior of the model is influenced by the block size, block elastic parameters (Young’s modulus and Poisson’s ratio), contact elastic parameters (shear and normal stiffness), and contact strength parameters (tensile strength, cohesion, and friction). DOE set the values of the model parameters by calibrating the unconfined compressive strength and elastic stiffness of the model (i.e., block assemblage) against the unconfined compressive strength and elastic stiffness of the rock mass on the basis of laboratory test data, as outlined in BSC (2004al, Section 7.6.1). DOE implemented the model in a two-dimensional universal distinct element code UDEC, as identified in BSC (2004al, Section 3.1), and used the code to calculate changes in drift profile and amount of rubble accumulation due to seismic events, and drip shield loading due to rubble. Additional details of the rock characterization, laboratory and field testing, numerical experiments for calibration, and field validation are summarized and reviewed in SER Section 2.2.1.2.1.3.2. The application of this model for estimating the amount of rubble accumulation and the corresponding loads that might act on the drip shields and other components of the EBS is addressed in the following sections.

DOE indicated in SAR Section 2.3.4.4.5.4 that seismic ground motions could cause partial or complete collapse of drifts in lithophysal rock, resulting in various amounts of rubble accumulation for different ground motion magnitudes and mechanical categories of lithophysal rock (BSC, 2004al). To describe the effects of ground motion magnitude and lithophysal rock categories on potential rubble accumulation, DOE performed analyses for ground motion at PGV levels of 0.4, 1.05, and 2.44 m/s [1.31, 3.44, and 8 ft/sec], which, according to DOE, correspond to an annual frequency of exceedance of 10^{-4}, 10^{-5}, and 10^{-6}, respectively (SNL, 2007ay). DOE performed the analyses using 15 ground motion time history cases at each annual frequency of exceedance and 5 sets of values of mechanical properties representing the 5 lithophysal rock categories as described in SAR Section 2.3.4.4.5.4 and BSC Section 6.4.2.2.2 (BSC, 2004al). DOE concluded that (i) ground motion with an annual frequency of exceedance of 10^{-4} will have negligible effects on drift degradation, (ii) ground motion with an annual frequency of exceedance of 10^{-6} will cause complete drift collapse, and (iii) ground motion with an annual frequency of exceedance of 10^{-5} could cause various degrees of collapse and varying amounts of rubble accumulation. DOE selected 15 analysis cases (SAR Table 2.3.4-23) to represent potential rubble accumulation due to seismic ground motions, as described in SNL Section 6.7.1.1 (2007ay).

DOE used the calculated rubble volumes from 11 of the 15 cases to develop relationships between rubble accumulation and the PGV of a seismic event, as described in SAR Figure 2.3.4-48 and SNL (2007ay, Section 6.7.1.2, Figure 6-57). SAR Section 2.3.4.4.8.3.1 stated that four cases calculated using rock mass Category 1 properties were eliminated because DOE considered the rubble volumes from the four cases to be nonrepresentative.

According to SAR Section 2.3.4.4.8.4, DOE used the resulting relationship between rubble volumes and PGV to estimate the amount of rubble accumulation due to a seismic event. DOE estimated the rubble accumulation due to multiple seismic events by adding the accumulations from the individual events, as identified in SAR Section 2.3.4.4.8.4 and SNL (2007ay, Section 6.7.1.4). The abstraction did not include weakening of the host rock due to previous events because rapid filling of drifts in lithophysal units mitigates concerns about numerous seismic events slowly weakening the rock mass, as DOE explained in SNL (2007ay, Section 6.7.1.4). Therefore, to calculate the drift volume fraction filled with rubble in lithophysal rock areas after a sampled seismic event, DOE accumulated rockfall [using relationships based on SNL (2007ay, Table 6-30 and Figure 6-56)] and divided the accumulated volume by a number sampled from a uniform distribution between 30 and 120 m^3/m [320 and 1,280 ft^3/ft]. The uniform distribution of 30–120 m^3/m [320–1,280 ft^3/ft] represents DOE's estimate of the volume of rockfall that would fill a drift with rubble, as described in SNL (2007ay, Section 6.12.2). SAR Figure 2.1-14 summarized DOE's estimates of potential rubble accumulation due to seismic events during a 1-million year period. According to this figure, at 10,000 years, the volume of accumulated rubble in the drift could reach as much as 15 percent of the drift volume (with a mean value of approximately 4 percent). Similarly, the corresponding volume of accumulated rock rubble at 100,000 years reached about 60 percent (with a mean value of about 32 percent). The volume of rubble per unit length of drift that is required to fill the drift was sampled for each epistemic realization in the TSPA analysis, and the sampled volume ranged uniformly between 30–120 m^3/m [320–1,280 ft^3/ft].

To calculate the static load on a drip shield due to rubble, DOE multiplied the volume fraction of the drift filled with rubble (on the basis of its assessment of drifts in lithophysal rock) by the drip shield load of a fully collapsed drift, as outlined in SNL (2007ay, Section 6.12.2). The magnitude of drip shield loading due to rubble at a given time depends on the amount of rubble accumulation, shape of the rubble pile, and the amount of rubble loading transmitted to the drip shield for a given amount and shape of the rubble pile. DOE used the information to determine the potential occurrence of rubble loading large enough to damage the drip shield in lithophysal and nonlithophysal areas.

NRC Staff's Review of DOE's Assessment of Drift Degradation

The NRC staff reviewed DOE's process-level modeling of drift degradation due to seismic events. Although DOE provided results of degradation of excavated drifts for the two major rock types (lithophysal and nonlithophysal), DOE stated that the rockfall volume in the nonlithophysal zones is significantly less than in the lithophysal zones for the same PGV level. This, according to DOE, is because the nonlithophysal rock mass is significantly stronger than the lithophysal rock. The NRC staff verified this DOE conclusion by comparing the strength and mechanical properties of the two rock types provided in SAR Tables 2.3.4-16 and 2.3.4-17. The NRC staff also reviewed the details of drift degradation analyses provided in BSC (2004al) and finds that DOE's assessment of behavior of the two rock types is supported by laboratory and field test data. Considering its review of the results of DOE's site characterization data that show that 85 percent of the repository block consists of lithophysal rocks, the NRC staff focused its review mainly on the degradation of lithophysal rocks, although the NRC staff also reviewed the DOE analyses of nonlithophysal rocks to the extent necessary.

According to DOE's data (SAR Table 2.3.4-25), the mean rockfall volume in the lithophysal rock is a factor of 40 to 200 greater than the mean rockfall volume in the nonlithophysal rock for the 1.05 and 2.44 m/sec [3.44 and 8 ft/sec] PGV levels. On the basis of a review of DOE's information, the NRC staff finds that DOE's assessment that the weaker lithophysal rock will usually fail before the stronger nonlithophysal rock during a seismic event is acceptable. The NRC staff verified that the rockfall volumes used for calculating the fraction of drift filled with rubble as a function of time presented in SAR Figure 2.1-14 are supported by model predictions of rubble volume calculated for lithophysal and nonlithophysal rocks (SAR Tables 2.3.4-23 and 2.3.4-24). Based on this verification, the NRC staff finds DOE's use of 30–120 m^3/m [320–1,280 ft^3/ft] as the potential range of rock rubble per unit length of emplacement drift in the TSPA represents an acceptable range to be used for both rock types. Thus, the NRC staff concludes that DOE's approach of using a single bounding range to represent rockfall in both rock types is justified because it adequately covers the appropriate range for lithophysal region and conservatively covers the potential range of rockfall in the nonlithophysal region.

The NRC staff further evaluated DOE's assessment of potential drip shield loading due to rubble accumulation in lithophysal rock sections of emplacement drifts by considering the amount of rock accumulation and the shape of rubble piles.

Amount of Rubble Accumulation

The timing and amount of rubble accumulation in a drift due to seismic events depend on the occurrence of seismic ground motions large enough to cause rock failure around the drift opening. The occurrence of rock failure during a seismic event depends on the ground motion levels and the rock mass strength. The rock mass strength around an emplacement drift during a seismic event may be affected by any previous weakening of the rock due to thermal stress, time-dependent effects, or prior seismic ground motions. In SNL (2007ay, Section 6.7.1.4), DOE's abstraction of rubble accumulation due to seismic events did not include any effects of rock weakening, because, according to DOE, rapid filling of drifts in lithophysal units mitigates concerns about the effects of rock weakening. In response to an NRC staff's RAI, DOE provided information in DOE (2010aa, Enclosure 1) that showed the following: (i) seismic ground motions with an annual frequency of exceedance of 10^{-4} will have negligible effects on drift degradation if the host rock has not been affected by thermal stress; (ii) ground motions with an annual frequency of exceedance of 10^{-4} that occur in the presence of thermal stress in the rock could cause some additional drift damage and a corresponding additional rubble accumulation, as outlined in DOE [2010aa, Enclosure 1, Figure 3(a)]; and (iii) the amount of drift damage and rubble accumulation due to the seismic event increase if the event occurs after a long period of time-dependent weakening of the rock, as described in DOE [2010aa, Enclosure 1, Figure 3(b)]. The NRC staff reviewed DOE response to the NRC staff's RAI and finds that computations presented by DOE showed, with sufficient technical basis, that the potential incremental rockfall due to (i) multiple seismic events and (ii) combined effects of time-dependent weakening and seismic events was relatively small during the first 10,000 years, as detailed in DOE (2010aa, Enclosure 1, Figure 4).

The NRC staff compared the analytical results provided by DOE to empirical mining industry data provided in DOE's response to the NRC staff's RAI, as identified in DOE (2010aa, Enclosure 1, Figure 5). DOE (2010aa, Enclosure 1, Figure 5) and BSC (2004al, Figure 6-149) summarized the caving potential of an excavated underground opening. Caving potential is expressed in terms of modified rock mass rating (a measure of rock quality and strength where a higher value indicates greater stability) and hydraulic radius [a dimension based on the geometry of the excavated opening; hydraulic radius is also a measure of stability (i.e., a larger hydraulic radius indicates a less stable opening and hence a higher caving potential)]. The rock mass rating data for Yucca Mountain rocks (between 50 and 60) show the corresponding hydraulic radii needed to cause high caving potential, which is on the order of 25 to 35 m [82 to 115 ft]. DOE analyses showed that the hydraulic radius of a degraded waste emplacement drift after 10,000 years of heating and time-dependent strength degradation coupled with a seismic ground motion would still be far less than the hydraulic radius of a degraded opening with high caving potential. Therefore, the NRC staff finds acceptable the DOE conclusion that, while the incremental rockfall accumulation due to combined effects and cumulative effects could be considerable over the entire period of repository performance (hundreds of thousands of years), DOE, in its TSPA analyses, did not underestimate the amount of rubble accumulation used as inputs in its TSPA model.

Shape of Rubble Piles

The NRC staff reviewed the process-level model that DOE used to analyze drift degradation in lithophysal rock and determined that it appeared constrained because of the upper boundary imposed in the analyses. In BSC (2004al, Figure 6-116), the tessellated domain (a mosaic pattern used to represent the domain with discrete elements) of the model is set 10.25 m [33.6 ft] above the initial drift roof. The upper boundary of the potential degradation zone DOE used was 1.86 drift diameters above the initial drift roof. However, contours of block displacement magnitude intersected the upper boundary of the tessellated domain. The calculated displacement contours indicate that some additional displacement could occur outside this domain [e.g., BSC (2004al, Figure 6-176)]. Also, plots of the final position of the Voronoi blocks after an analysis [e.g., BSC (2004al, Figures P-17, P-18, and P-24)] indicate blocks at the top of the model could be predicted to separate from the overlying elastic domain. Such a separation would suggest the caved zone might have extended higher if the model upper boundary had been higher.

In response to an NRC staff's RAI, DOE provided analyses to show that the rubble volume calculated using DOE's model is insensitive to the size of the tessellated domain (DOE, 2010aa). DOE provided results from two models with the boundary of the tessellated domain at 8.25 and 13.25 m [27 and 43.5 ft], respectively, in DOE (2010aa, Enclosure 2, Section 1.3). DOE used the models to estimate the extent of caving needed to fill a drift with rubble if the prevailing mechanical conditions were to cause complete drift collapse. This was accomplished by artificially degrading the rock mass strength to zero in a quasi-static analysis. The results of these studies demonstrated that the patterns of calculated displacements and stresses did not change appreciably as a result of changing the size of the tessellated domain. The quasi-static analyses showed that the caved zone extended approximately one drift diameter above the drift irrespective of the size of the tessellated domain. DOE also provided calculations to show that potential caved zones due to seismic events will likely extend to much less than one drift diameter above the drift, except for seismic events with an annual exceedance probability of 10^{-6} or smaller.

The NRC staff reviewed DOE's analyses with the expanded boundaries. The additional DOE analyses showed continuity in the displacement field crossing the tessellated region, as described in DOE (2010aa, Enclosure 2, Figures 8 and 9). On the basis of the review of DOE's responses to the NRC staff's RAI, the NRC staff concluded that the use of the original boundary did not significantly affect or limit the estimates of rubble accumulation. In addition, the results of the sensitivity analyses DOE provided demonstrated that with the expanded boundary of the tessellated region, the distressed zone is completely contained within the original (smaller) tessellated domain. Thus, DOE's response to NRC staff's questions satisfactorily demonstrated the sufficiency of the lateral and vertical extent of the model used for estimating rockfall due to seismic loads. Therefore, the NRC staff finds acceptable DOE's conclusion that rock rubble estimates resulting from seismic events are not underestimated in the process level models.

Rubble Loading Transmitted to the Drip Shield

DOE used the results of the discontinuum model to estimate the potential drip shield loading due to rubble (SAR Figure 2.3.4-43). DOE also presented alternative bounding analytical approaches for estimating the potential static loading on the drip shields (continuous curves in SAR Figure 2.3.4-46). The analytical model estimates drip shield loading due to rubble using the dead weight of rubble. However, it is well established in the field of rock mechanics that loads transmitted to a drip shield from rubble could differ because of frictional resistance among broken pieces of rock rubble, between rubble and the drift wall, or between rubble and the sides of the drip shield.

To justify using the loads from the numerical model, SAR Section 2.3.4.4.6.3.2 stated that DOE's process-level model accounts for load transmission among rubble particles and to the drip shield. However, DOE in BSC (2004al, Section P4) identified factors that may affect load transmission within rubble and to the drip shield, including size and shape distribution of rubble particles, rubble compaction, and deformability of the drip shield and invert. DOE represented the rock mass in the process-level model as an assemblage of roughly equidimensional polygonal blocks with a characteristic length of approximately 0.2 m [0.65 ft] (SAR Figure 2.3.4-40). SAR Section 2.3.4.4.5.3 and BSC (2004al, Section 6.4.2.1) stated that the model blocks are approximately the same size as potential lithophysal rock blocks.

However, independent NRC staff's analysis in Ofoegbu, et al. (2007aa, p. 4-16) of DOE's fracture data suggests a wide range of rock block sizes and shapes for the lithophysal rock mass, which contrasts with the approximately uniform block size and shape that DOE uses in the rock mass model. Because the potential block size and shape distributions of rubble from the lithophysal rock mass could be different from the approximately uniform block sizes and shapes DOE uses in its drift degradation modeling, the effects of load transmission through rubble could be different from what DOE used in its analysis. Therefore, the NRC staff requested additional information from DOE to demonstrate that appropriate variations in block size and shape have been considered in the DOE UDEC-Voronoi analyses of rockfall during seismic events. In response, the NRC staff received additional information in DOE (2010ab, Enclosure 1) and the response demonstrated that the bulking factors obtained from UDEC calculations are below the lower end of the ranges for rock rubble because of the assumption that particles are of approximately equal size. This would result in overestimation of the loads acting on the drip shields. DOE conducted additional sensitivity analyses considering different block sizes and demonstrated that, for the range of bulking factors of interest, the average vertical pressure on the drip shield increased only by a small amount (small compared to the standard deviation). Therefore, on the basis of the sensitivity studies, the NRC staff finds acceptable DOE's conclusion that loads were not underestimated when using roughly equidimensional polygons of characteristic length of approximately 0.2 m [0.65 ft].

Summary of NRC Staff's Evaluation of Seismic Drift Degradation

The NRC staff evaluated DOE's assessment of potential degradation of emplacement drifts due to seismic events and estimates of drip shield loading resulting from rubble accumulation and finds that

- DOE adequately considered in the process level models features such as lithophysae and rock fractures, processes such as thermal loading, and time dependent weakening of rocks and events such as seismic, in analyzing the degree of drift degradation during the initial 10,000 year period, and projected these features, events, and processes beyond the 10,000 year post-disposal period through the period of geologic stability.

- DOE used alternative conceptual models including empirical, analytical, and numerical models and acceptable methodologies for estimating the timing and extent of drift degradation due to seismic events including appropriate consideration of thermal loads and time-dependent weakening of excavated drifts that may be potentially subject to repetitive seismic events

- DOE appropriately considered relevant geologic data obtained during site characterization and the variability of mechanical properties of rocks and associated uncertainties in parameters used in the models in the supporting analyses

- DOE justified the abstraction of rock rubble loads due to seismic degradation of excavated drifts and the appropriate use of associated parameters for TSPA inputs

2.2.1.3.2.3.4 Drip Shield Structural/Mechanical Performance in the Context of Its Seepage Barrier Function

In DOE's EBS, the drip shield is a freestanding structure that surrounds the waste package and rests on crushed rock that forms the invert at the base of the drift. The drip shield is designed to protect the waste package from contact by seepage water and rockfall (SAR Section 2.3.4.5.1.1). The main structural elements of the drip shield consist of a framework that includes a bulkhead and support beams (legs) that will be made of Titanium Grade 29. Plates of Titanium Grade 7 are welded onto the framework to form a full composite structure in response to mechanical loading (SAR Section 2.3.4.5.1.1; SAR Figure 2.3.4-56).

Damage to the drip shield can occur from mechanical impacts of falling rocks, by loads from accumulated rock rubble that can be increased by seismic accelerations, and by corrosion processes. Through time, DOE expects that thinning of drip shield components will decrease the capacity of the drip shield to withstand loads and that the likelihood of the drip shield having experienced a potentially damaging load will increase (e.g., SAR Section 2.3.4.1). During the thermal period following repository closure, temperatures within the drifts decrease from around 160 °C [320 °F] to below the boiling point. At these elevated temperatures, generalized corrosion could occur if water contacts the surface of the waste packages. Thus, if the barrier capability of the drip shield fails in the first 12,000 years following repository closure, seepage water could contact the waste package and lead to localized corrosion. DOE relies on the presence of the drip shield as a barrier to preclude significant occurrences of localized corrosion (e.g., SAR Section 2.1.2.2.6).

Consistent with the guidance in the YMRP, the NRC staff focused its review on the risk-significant aspects of the drip shield performance. On the basis of DOE's reliance on the drip shield in the demonstration of multiple barrier requirements (SAR Section 2.1.2.2.6), the NRC staff focused on evaluating the performance of the drip shield as a barrier to seepage during the first 12,000 years following drift closure.

For seepage to contact a waste package, openings must occur on the drip shield of sufficient size to permit the advective flow of water through the drip shield plates. Crack openings, such as those produced by stress corrosion cracking, are too small to allow advective flow of water through the drip shield and are excluded from the performance assessment analysis, as described in SNL (2008ab) for FEP 2.1.03.10.0B. Openings large enough for advective water flow could potentially occur through (i) corrosion processes, (ii) impacts of large rock blocks causing puncture of the plates, (iii) physical separation between adjacent drip shield segments due to ground motions from seismic events, (iv) fault displacements, and (v) rupture by deformations that produce effective strains greater than the failure strains in the plates (SNL, 2007ap).

The NRC staff's evaluation of the drip shield corrosion processes is presented in SER Section 2.2.1.3.1. In that review, the NRC staff determined that DOE used acceptable corrosion rates for titanium alloys. Thus, the timing and degree of drip shield component thinning due to corrosion is appropriate for mechanical analyses of the drip shield performance in this SER section. DOE excluded large-block impacts from the TSPA as part of the screening analysis for FEP 1.2.03.02.0B (SNL, 2008ab). The NRC staff reviewed the DOE screening arguments in SER Section 2.2.1.2.1 and concluded that DOE had adequately screened out this FEP from the performance assessment analysis. Thus, the NRC staff's detailed review in this section of the SER focuses on DOE's representation of processes affecting drip shield separations caused by seismic events or fault displacements, and on the potential for plate rupture.

Separations of Drip Shields from Seismic Ground Motion

Unlikely or low probability seismic events can create ground motions that may cause adjacent drip shields to separate. Consequently, DOE assessed the potential for drip shield separations during seismic events, as described in SNL (2007ay, Section 6.7.3). DOE determined that ground motions large enough to cause potential drip shield separations also cause partial to complete collapse of the repository drifts. DOE determined that rockfall associated with drift collapse occurs during the initial few seconds of large seismic events. DOE modeled the effect of this rockfall on the ability of the drip shields to separate during seismic events and concluded that rockfall loads from partial drift collapse are sufficient to prevent horizontal separation of the drip shields, as outlined in SNL (2007ay, Section 6.7.3). While these models calculated that minor amounts of vertical separation might occur between the drip shield sections due to settling of the invert or framework damage, the 26-cm [10.24-in] overlaps between the drip shield connectors prevent rockfall or seepage water from contacting the waste package through relatively small vertical separations. In FEP 1.2.03.02.0A (SNL, 2008ab), DOE concluded that seismically induced separations of drip shields can be excluded from the TSPA analysis on the basis of low probability.

NRC Staff's Review

The NRC staff reviewed the information presented in SNL (2007ay, Section 6.7.3) and analyses in BSC (2004bq, Section 5.3.3.2.2). The NRC staff finds that DOE used rockfall rubble loading conditions that were consistent with the results documented in BSC (2004al) and are appropriate for the seismic events modeled. The NRC staff concludes that the DOE model used the same approach to evaluate the dynamic response of drip shields as was used to evaluate the dynamic response of waste packages. The NRC staff finds that this modeling approach is acceptable, as documented in the modeling approach sections. The NRC staff reviewed the dynamic analyses in SNL (2007ay, Section 6.3.7.1) and confirmed that potential separations of the drip shield only occurs in an open drift that is subjected to 5.35 m/sec [17.55 ft/sec] PGV ground motion. On the basis of its review of DOE analyses, the NRC staff finds that complete drift collapse is expected for large magnitude seismic events (BSC, 2004al).

The NRC staff finds that the DOE approach of modeling an open drift (i.e., no rockfall rubble) maximizes the potential for drip shield separations to occur during seismic events because the presence of rockfall rubble effectively pins the drip shield segments and restricts the ability for segments to separate. The NRC staff concludes that the DOE modeling approach to represent drip shield kinematics during seismic events is reasonable because this approach maximizes the potential for separations to occur. Because drip shield separation can only occur under a combination of open drift conditions and very low probability seismic events (and under such low probability events the drift will completely collapse), the NRC staff finds DOE's reasoning for excluding drip shield separations from seismic events in the TSPA analysis acceptable.

Separations of Drip Shields Caused by Fault Displacement

DOE concluded that fault displacement occurs concurrently with the ground motion during low probability seismic events (SAR Section 2.3.4.5.5.1) and determined that only EBS components located directly above the moving faults are subject to damage. In the analysis of the effects of fault displacements on EBS performance, DOE assumed that the drip shield fails completely if fault displacements are sufficient to breach the underlying waste package (SAR Section 2.3.4.5.5.4). In this analysis, DOE assumed that all seepage water entering the drift passes through the failed drip shield, with no diversion of the water.

NRC Staff's Review

The NRC staff's review of the fault displacement model with regard to waste package failure is presented in SER Section 2.2.1.3.2.3.2. The NRC staff determines that DOE's assumption of drip shield failure is acceptable for use in the fault displacement analysis because the magnitude of a faulting event required to damage a waste package would be sufficient to also damage the drip shield. The NRC staff also concludes that DOE's assumption that a damaged drip shield has no barrier capability is acceptable because this assumption provides a credible upper bound to the potential significance of fault displacement on drip shield performance.

Plate Rupture

According to DOE, to affect performance significantly, the drip shield barrier must fail and allow advective water flow to contact the waste package during the first 12,000 years of postclosure, when DOE expects that environmental conditions support localized corrosion of the waste package (e.g., SAR Section 2.1.2). Deformation of the drip shield plates can occur if the underlying framework buckles or collapses due to physical loading. DOE concluded that rupture of the drip shield plate can occur if the magnitude of effective strain on the plate exceeds the strain threshold for the Titanium Grade 7 plates (e.g., SAR Section 2.3.4.5.1.2.2).

NRC Staff's Review

The NRC staff's evaluation of the potential for drip shield plate rupture focuses on DOE's modeling approach to evaluate effective stresses in the plates following framework collapse and the basis to determine the location of potential ruptures on the drip shield. The NRC staff focused its review on plate ruptures in the crown area rather than the sides of the drip shields because the ruptures that occur on the sides of the drip shield have negligible potential to allow water contact with the waste package, whereas ruptures on the crown area of the drip shield are likely to allow seepage water to contact the waste package. On the basis of the acceptable use of titanium alloy corrosion rates (reviewed in SER Section 2.2.1.3.1), the NRC staff noted that negligible thinning of the drip shield components is expected during the first 12,000 years of postclosure. The NRC staff's review focuses on treatment of temperature effects; modeling approach; and drip shield framework deformation, as described in the following sections.

Treatment of Temperature Effects

Structural/mechanical analyses of drip shield performance are dependent on the material properties used in the numerical models. DOE used mechanical properties for the drip shield plates and framework derived from standard handbooks and manufacturer's catalogs (SAR Section 2.3.4.5.1.3.1 and Table 2.3.4-28). DOE considered that a reference temperature of 60 °C [140 °F] for these properties was appropriate, because this temperature is representative of most of the repository closure period. Although DOE recognized that temperatures as high as 300 °C [572 °F] could potentially occur soon after repository closure, DOE considered that the duration of elevated temperatures was too short to warrant consideration for drip shield performance (SAR Section 2.3.4.5.1.3.1).

DOE provided additional information to assess the potential effects of temperatures at or greater than 120 °C [248 °F] on titanium alloy material properties, as discussed in DOE (2009bp, Enclosure 7). During the first 650 years of repository closure, DOE concluded that drip shield temperatures could range from 120–300 °C [248–572 °F]. DOE expects the effect of this temperature increase would not affect titanium alloy properties significantly because the likelihood for potentially damaging rockfall or seismic events is sufficiently low to preclude significance in the performance assessment. For temperatures below 120 °C [248 °F], DOE compared expected changes in material properties (e.g., yield strength, tensile strength) to assess the effects of component thinning on the likelihood of drip shield plate or framework failure. Using small rockfall loads, DOE concluded that changes in the titanium mechanical properties between 60–120 °C [140–248 °F] are a factor of 3 to 4 less than the corresponding percentage changes in component thicknesses that have no significant effect on fragility values.

NRC Staff's Review

The NRC staff reviewed the mechanical properties DOE used for titanium alloys at 60 °C [140 °F]. The NRC staff compared these values with values available in standard reference handbooks and concluded that DOE used the appropriate mechanical material properties (e.g., yield stress and ultimate tensile strength) for drip shield performance at 60 °C [140 °F].

The NRC staff evaluated the rationale DOE provided to exclude consideration of temperatures greater than 120 °C [248 °F] on titanium material properties. The NRC staff confirmed that seismic events with $<10^{-5}$ annual probabilities of exceedance are required to produce appreciable amounts of rockfall and that annual probabilities of $<10^{-6}$ are required for reasonable likelihoods of complete drift collapse (SAR Section 2.3.4.4.5.4). On this basis, the NRC staff concludes that the likelihood of appreciable drift collapse occurring in the first 650 years of closure is small. The NRC staff also conducted independent confirmatory analyses that evaluated the effects of increasing temperature from 150 to 260 °C [302 to 500 °F] (Ibarra, et al., 2007aa). These analyses showed an approximate 20 percent decrease in drip shield capacity associated with this temperature increase. The NRC staff concludes that this decrease in capacity would not affect the performance significantly because temperatures associated with this decrease would persist only for hundreds of years and loads associated with potential damage to the drip shield are unlikely to occur.

Based on its review of the seismic hazard presented by DOE, the NRC staff determines that complete drift collapse from seismic events could occur during the first 12,000 years of repository closure although such a scenario would be unlikely. The DOE information did not adequately address the uncertainties associated with rockfall load and temperature effects for a potential scenario corresponding to this low probability. In response to an NRC staff RAI, DOE provided additional information in DOE (2009bp, Enclosure 7), which assessed the potential temperature effects at 120 °C [248 °F] at 10 percent rockfall loads. Using the methods DOE developed to evaluate 10 percent of rockfall loads, the NRC staff extended this approach to 100 percent of potential rockfall loads. Using the information in SAR Table 2.3.4-43, the NRC staff finds that an approximately 30 percent reduction in drip shield plate thickness minimally increases the likelihood of plate rupture from approximately 1 percent to approximately 5 percent. Although some strength properties show 30 percent variations from 60 to 120 °C [140 to 248 °F], the NRC staff concludes that these variations are expected to have only a small to negligible effect on the likelihood of plate rupture for 100 percent collapsed drifts. This is because the titanium plate will have increased ductility and, thus, increased its ability to accommodate deformation without rupture under loads associated with unlikely seismic events. In addition, the NRC staff considers the loads that could potentially increase the likelihood of plate rupture are associated only with earthquakes having $<5 \times 10^{-7}$ annual likelihoods. Using insights from the TSPA model, the NRC staff concludes that potential changes in the likelihood of plate rupture on the order of several percent would not affect the performance assessment significantly. The NRC staff concludes that DOE's sensitivity analyses, as presented in DOE (2010ac), also demonstrate that potential changes in the likelihood of plate rupture on the order of several percent would not affect the performance assessment significantly. Therefore, the NRC staff concludes that DOE's use of titanium alloy material properties at 60 °C [140 °F] is an acceptable approach for evaluating postclosure repository performance because uncertainties associated with potentially higher temperatures would not significantly affect the results of the performance assessment.

Modeling Approach

To evaluate drip shield plate capacity, DOE conducted numerical modeling of the drip shield under quasi-static and dynamic loading conditions. For the quasi-static analyses, DOE calculated rock rubble loads on the drip shield, multiplied these loads by the vertical component of peak ground acceleration, and modeled the drip shield response to these loads. DOE calculated the quasi-static loading conditions on the drip shield plate using FLAC3D, a three-dimensional finite-difference computer code. DOE calculated stresses and strains on one-half of the plate on the drip shield crown, which represents one segment between two framework bulkheads.

DOE used rock rubble loads calculated from UDEC analyses of rockfall during seismic events (SAR Section 2.3.4.5.3.2). DOE evaluated two static loading configurations on the drip shield. One configuration used an average of six UDEC realizations for each modeled segment of the drip shield, which DOE used to consider spatial variability in the nonuniform load. The second configuration used a single UDEC realization, which DOE considered as representative of the highest loads on the drip shield crown (SAR Section 2.3.4.5.3.2.1).

For each loading configuration, the vertical load was applied over the entire top surface of the plate and increased incrementally until a failure mechanism developed, as described in SNL (2007ap, Section 6.4.3.1.2). For each load increment, the model compared the residual tensile stresses or accumulated plastic strain against a failure criterion of 80 percent of the yield strength for Titanium Grade 7, as outlined in SNL (2007ap, Section 6.4.3.1.3). DOE concluded that plate failure occurred at the smallest applied load that exceeded either the stress or strain criterion.

By uniformly increasing the static load in the UDEC model, DOE calculated that an intact drip shield plate has a capacity (i.e., limit load) of approximately 2,500 kPa [52,218 psf], which is approximately twice the calculated capacity of the drip shield framework (SAR Section 2.3.4.5.3.3.1). To determine the likelihood of drip shield plate failure, DOE integrated the annual likelihood of exceeding levels of ground acceleration with the likelihood of rupture for plates experiencing the loads corresponding to the level of ground acceleration. For intact drip shield plates and 100 percent rockfall load, DOE calculated that seismic events with annual probabilities of exceedance $<5 \times 10^{-7}$ can lead to plate rupture on 1–7 percent of the drip shields (SAR Section 2.3.4.5.3.4).

As an alternative to the quasi-static analyses, DOE also conducted dynamic analyses for drip shield plate capacity using the UDEC computer code (SAR Section 2.3.4.5.3.3.3). These analyses used a two-dimensional cross section of the drip shield surrounded by rock rubble. The dynamic analyses used vertical ground accelerations from time histories that DOE views as representative of larger magnitude seismic events in the Yucca Mountain region. DOE applied these vertical accelerations to the basal boundary of the UDEC model, which allows the emplacement drift, rubble, and drip shield to interact dynamically for the modeled period of strong ground motion. DOE compared the results of the dynamic analyses with the quasi-static analyses and concluded that the quasi-static model underestimates the stability of the drip shield plates. DOE, therefore, concluded that the quasi-static approach provides a reasonable estimate of both the failure mode and limit loads for the complex case of strong ground motion shaking of the drip shield and rubble. Thus, DOE also concluded that a quasi-static model is an appropriate basis to calculate the likelihoods of plate rupture (SAR Section 2.3.4.5.3.3.3).

NRC Staff's Review

The NRC staff reviewed the use of the FLAC3D computer code in the analyses of the drip shield plate capacity. The NRC staff reviewed the information in SNL (2007ap, Section 7.3.3.1) and determined that DOE appropriately compared the FLAC3D model results with an alternative approach used in structural mechanical analyses (i.e., LS-DYNA). In response to the NRC staff's RAI, DOE provided the additional information in DOE (2009bp, Enclosure 8) to address the representation of nonlinear responses of materials.

The NRC staff confirmed that the rock rubble static loads used in the models were consistent with the degraded drift configurations used elsewhere in SAR Section 2.3.4.5 for seismic events, and that the UDEC analysis representation of this rubble resulted in bulking factors that were appropriate for the Topopah Springs lithophysal tuff. In addition, the NRC staff conducted independent confirmatory calculations using an alternative modeling approach to evaluate drip shield deformation from rock loading (Ibarra, et al., 2007aa). The NRC staff compared the deformation patterns determined from the independent model to the deformation patterns DOE determined using the dynamic modeling approach and reviewed DOE responses to the NRC staff's RAI. On the basis of this review, the NRC staff finds that DOE provided support for using FLAC3D to calculate drip shield plate performance. In addition, the results of this comparison supported the DOE conclusion that, because the drip shield framework has lower capacity than the plates, deformation is most likely to occur on the legs of the drip shield and not on the crown. The NRC staff finds that the quasi-static analyses provide a reasonable basis to determine the likelihood of drip shield plate rupture from seismically accelerated rock rubble loads.

Drip Shield Framework Deformation

DOE calculated the likelihood of drip shield framework failure using the same approach as implemented for the drip shield plate analyses (SAR Section 2.3.4.5.3.3.2). These analyses determined that the drip shield framework has approximately half the bearing capacity as compared to the drip shield plates and that buckling of the drip shield legs results from exceeding the bearing capacity. DOE also determined that if the drip shield becomes tilted after the framework buckles, the drip shield connector plate and connector guide provide a physical barrier that will divert seepage from the crown to the sides of the drip shield, as outlined in SNL Section (2007ay, 6.7.3.2).

DOE postulated that if one segment of a drip shield collapsed more extensively than adjacent segments, localized stresses may lead to rupture of the drip shield plates along the crown. DOE considered the likelihood of isolated segment collapse to be low because rubble loads are expected to be relatively uniform, and the rigidity of the drip shield is expected to effectively transfer loads to the adjacent segments (DOE, 2010ac). Thus, DOE expects complete collapse of the drip shield when loads exceed the design capacity. Nevertheless, DOE analyzed stress-strain relationships for a partially collapsed drip shield and determined that plate rupture would occur if vertical displacements between adjacent segments exceeded approximately 19 cm [7.5 in] (DOE, 2010ac). DOE concluded that such displacements between adjacent segments are unlikely to occur, because the structure of the drip shield will effectively transfer stress from a deforming segment onto the adjacent segments. This stress transfer leads to a progressive collapse of adjacent drip shield segments, rather than isolated collapse and potential tearing of a single segment (DOE, 2010ac).

NRC Staff's Review

The NRC staff conducted independent confirmatory calculations using an alternative modeling approach to evaluate drip shield deformation from rock loading (Ibarra, et al., 2007aa). These calculations confirmed that buckling of the drip shield framework is expected in the legs. The NRC staff evaluated the drip shield design DOE provided (e.g., SAR Figure 1.3.4-15). On the basis of the low volumes of seepage water potentially contacting the drip shield, the NRC staff concludes that the drip shield connector plate and connector guide adequately divert seepage water from the crown area if the drip shield is tilted due to framework buckling. Thus, the NRC staff finds that uncertainties in the drip shield framework capacity would not adversely affect the potential for seepage water to contact the waste package through tilting of the drip shield if buckling occurred in the drip shield legs.

The NRC staff concluded that potential underestimation of the drip shield framework capacity could result in early transitions to a damage state (i.e., SAR Section 2.3.4.5.4.1, Idealized Damage State 2) where the waste package is pinned by the collapsed drip shield. As a result, the waste package would be restrained from movement during large magnitude seismic events and have a reduced potential for stress corrosion cracking. For unrestrained motion during seismic events, as would occur when the drip shield is intact, up to 4 percent of the waste package surface area can be damaged sufficiently for stress corrosion cracks (SCC) to develop (SAR Section 2.3.4.5.2.1.4.2). In contrast, a collapsed drip shield localizes the potential damaged area on the waste package and results in an approximate order-of-magnitude decrease in the potential for stress corrosion cracking (SAR Section 2.3.4.5.4.3.2.1). On the basis of these relationships, the NRC staff concludes that potential uncertainties that result in DOE overestimating drip shield framework capacity are not significant to performance, because increasing the ability of the drip shield framework to withstand seismic loads increases the potential for larger amounts of waste package damage, and potential radionuclide releases, through SCC.

The NRC staff reviewed the DOE modeling approach for evaluating plate response during partial drip shield framework collapse and finds that DOE used appropriate physical parameters and geometries to evaluate stress-strain relationships (DOE, 2010ac). The NRC staff independently confirmed that: (i) the DOE modeling approach (using the FLAC3D and LS-DYNA computer programs) is consistent with standard practice for determining stress-strain relationships and (ii) DOE's failure criteria are appropriate and adequately implemented.

The NRC staff finds acceptable DOE's assumption that relatively uniform rock-rubble loads generally occur on a drip shield during seismic events, on the basis of the relatively uniform characteristics of the expected rock rubble and because most drifts would have wholly collapsed during potentially damaging, very low probability seismic events. Nevertheless, the NRC staff recognizes that DOE represented heterogeneity in rubble load for some calculations for evaluating drip shield fragility [e.g., SNL Section 6.4.3.2.2.2 (2007ap)]. The NRC staff also notes that some potential exists for localization of rubble loads on the drip shield during very low probability, large magnitude seismic events. As a consequence, the NRC staff concludes that potential exists for the rubble load to be greater on some segments of a drip shield than on adjacent segments. However, in the context of how these results are used in DOE's performance assessment, the NRC staff determines that a more detailed consideration of the potential nonuniform load distribution is not likely to affect the overall structural performance of the drip shield frame as to significantly affect its functionality as a seepage barrier. Therefore, the NRC staff concludes that such variations have minor effects on the overall seepage barrier performance of drip shields.

On the basis of the review of the drip shield design and the expected response of the drip shield as a composite structure, the NRC staff finds DOE adequately concluded that significant stress redistribution would occur to adjacent drip shield segments, if loading was localized on an individual segment. The NRC staff concludes that differential collapse of the drip shield would most likely involve at least several adjacent segments, which would be sufficient to prevent localized strains from exceeding the failure strain of the drip shield plates. The NRC staff recognizes that there is uncertainty in the amount of vertical displacement that the drip shield plates can accommodate {i.e., 18.2 cm [7.17 in] between two bulkheads} before the failure strain criterion derived from an assumed differential displacement of 19.8 cm [7.8 in] might be exceeded, as described in DOE (2010ac, Enclosure 9). At the same time, the NRC staff notes that the analysis DOE provided likely overestimates failure potential because (i) the boundary conditions on the bulkheads are assumed to be fixed (which overestimates stresses) and (ii) the three longitudinal stiffeners are neglected (which underestimates the overall stiffness of the composite structure).

To evaluate the potential significance of the uncertainty in the barrier capability of the drip shield plates during seismic events, the NRC staff used insights from the TSPA analysis for intrusive igneous events (SAR Section 2.4.2.2.1.2.3). In that analysis, DOE considered that an igneous intrusive event removed all barrier capabilities from the drip shield and the waste package, and made all waste available for dissolution and transport. Using an approximate 10^{-8} average annual probability of occurrence, DOE calculated a probability-weighted igneous intrusive dose equivalent to less than 0.001 mSv/yr [0.1 mrem/yr] for the 10,000-year period (SAR Section 2.4.2.2.1.2.3.1)]. In DOE (2009aa) and SNL (2008ag, Appendix P), DOE also showed that increasing the average annual probability of occurrence to 10^{-7} increases the expected annual dose equivalent to less than 0.006 mSv/yr [0.6 mrem/yr].

In comparison to an igneous intrusive event, seismic events have limited potential to create openings in drip shield plates that are large enough to permit advective water inflow. In addition, the NRC staff notes the following: (i) only a limited range of potential seepage waters has compositions that support potential localized corrosion processes (SER Section 2.2.1.3.3), (ii) in-drift conditions can support potential localized corrosion processes for only a limited period of time, and (iii) potential openings that could result from localized corrosion are small (SER Section 2.2.1.3.1). Thus, the NRC staff reasonably expects that potential releases from uncertainties in drip shield performance under seismic conditions would be appreciably smaller than what could be expected for an igneous intrusive event. Given that an igneous intrusive event (which essentially removes all EBS capabilities) contributes less than a dose equivalent of 0.006 mSv/yr [0.6 mrem/yr] to the total effective dose calculation, uncertainties related to the drip shield barrier performance during seismic events are expected to cause much less variation in the total annual dose. Thus, the NRC staff concludes that DOE has adequately accounted for the performance of the drip shield barrier function in the performance assessment because uncertainties in the DOE evaluation would not affect the results of the performance assessment significantly.

Summary of NRC Staff's Evaluation of Drip Shield Performance

DOE relies on the drip shields as effective barriers to advective water flow or rock rubble impacts on the waste package. The NRC staff reviewed the information DOE presented relevant to the barrier capability of the drip shield and finds the following:

- DOE appropriately identified and adequately considered potential events and processes such as fault displacement and seismically induced drift collapse that may lead to openings in the drip shield that affect barrier capabilities for the initial 10,000 year period, and projected these events and processes beyond the 10,000 year post-disposal period through the period of geologic stability.

- DOE used acceptable models and information to demonstrate that potential openings from horizontal or vertical displacements during seismic events would not affect performance significantly.

- DOE acceptably assumed that fault displacements sufficient to damage a waste package remove all barrier capabilities from the associated drip shield.

- DOE appropriately evaluated the potential for ruptures in the drip shield plates during the first 12,000 years of closure by taking a conservative approach.

- DOE appropriately determined that a small likelihood exists for such ruptures if earthquakes with annual probabilities of exceedance of $<5 \times 10^{-7}$ occur. DOE adequately implemented this likelihood of plate failure in the TSPA.

- DOE adequately demonstrated that uncertainties in this information would not affect the results of the performance assessment significantly.

The NRC staff finds that DOE adequately evaluated the barrier capabilities of the drip shield mechanical disruption due to seismic events and has appropriately incorporated the risk-significant aspects of this evaluation into the performance assessment calculations.

2.2.1.3.2.3.5 Waste Package Mechanical/Structural Performance

DOE classified the waste package as important to waste isolation (SAR Table 2.1-1). DOE provided information on structural response of the waste package to mechanical disruption in SAR Section 2.3.4.5. The objective of this SER section is to evaluate whether adequate technical bases have been provided for waste package abstractions used in DOE's TSPA.

DOE assessed potential waste package mechanical damage by performing detailed structural analyses. The results of these structural analyses were used as inputs to the seismic consequence abstractions (SCA). The SCA simulates mechanical interactions among the waste packages, the drip shield, the emplacement pallet, and/or accumulated rubble as a function of PGV. DOE calculated waste package damage as (i) SCCs that may allow diffusive radionuclide releases and (ii) rupture and puncture areas that may allow advective radionuclide releases (reviewed in SER Section 2.2.1.3.4.3.5).

The results of the seismic consequence abstractions are used as inputs to other process-level models and direct inputs to the TSPA. The waste package corrosion abstraction uses waste package breaches at the process level to initiate double-sided corrosion (reviewed in SER Section 2.2.1.3.1). Note that in this context, a breach is defined as any failure mechanism that penetrates the waste package (i.e., cracks, ruptures, and punctures). Waste package breaches also impact the chemistry inside the waste package (reviewed in SER Section 2.2.1.3.4). Stress corrosion crack area is used in the EBS transport abstraction to model a pathway for diffusive radionuclide release (reviewed in SER Section 2.2.1.3.4). Waste package rupture or puncture area is used in the flux-splitting model to calculate water flux through the waste package (reviewed in SER Section 2.2.1.3.3).

Information presented in SAR Table 2.1-3 suggests that seismic ground motion damage to the EBS components is an important mechanism that affects the EBS capability to perform its intended functions. DOE stated in DOE (2009bl, Enclosure 1) that seismically-induced waste package damage is more significant in early times and that nominal failure processes are more significant at later times. According to DOE, seismically-induced stress corrosion cracking is the most probable waste package damage mechanism. The majority of commercial spent nuclear fuel (CSNF) and CDSP waste package failures due to seismically induced stress corrosion cracking occur prior to drip shield plate/crown failure, as described in DOE (2009bl, Enclosure 1, Figures 5 and 6).

As described in the following list, DOE considered three idealized states of the EBS (SAR Section 2.3.4.5):

1. Structurally stable drip shield state (intact drip shield)—when the waste packages are free to move and may be damaged due to impacts with other components of the EBS during seismic events

2. Drip shield framework failure state (collapsed drip shield)—when the drip shield–waste package interactions during seismic events may damage the waste package outer barrier

3. Drip shield plates failure state—when the waste package is surrounded by and in direct contact with rubble and may be damaged due to waste package–rubble interactions during seismic events

As DOE detailed in DOE (2009bl, Enclosure 1, Figure 1), nominal stress corrosion cracking in a CSNF waste package would initiate between 200,000 and 300,000 years, when the timeframe is dependent on the drip shield performance (reviewed in SER Section 2.2.1.3.2.6). The initiation of stress corrosion cracking would occur after the beginning of Idealized State 2. The CSNF waste packages cannot move as freely in Idealized State 2 as in Idealized State 1, thereby reducing the potential for seismically induced stress corrosion cracking.

For the three idealized states, DOE considered two waste package failure modes.

1. The first failure mode is referred to as "the residual stress failure mode" in this SER section. The waste package damage is expressed in terms of the waste package outer corrosion barrier surface area that may be susceptible to stress corrosion cracking. It is defined as an area with the residual stresses exceeding one of three residual stress threshold values: 90, 100, and 105 percent of the Alloy 22 yield stress (reviewed in SER Section 2.2.1.3.1.3.2.3).

2. The second failure mode is referred to as "the tensile tearing failure mode" in this SER section. DOE used Alloy 22 ultimate tensile strain as a failure criterion to evaluate the waste package outer barrier tensile tearing (rupture and/or puncture) occurrence.

For these two failure modes, DOE developed the abstractions using a three-part approach: (i) the rupture/puncture probability was defined as a function of PGV and the effective tensile stress limits, (ii) the probability of a nonzero damaged area was defined as a function of PGV and the residual stress threshold damage, and (iii) for nonzero damaged area cases, a conditional probability distribution for the magnitude of the conditional damaged area was defined as a function of PGV and the residual stress threshold.

DOE's analyses results indicate greater mechanical damage potential to the waste package during Idealized State 1. However, the NRC staff reviewed the fundamental aspects of damages in all three idealized states and their abstractions. The review presented in this section is organized around these major topics considering the context of DOE's TSPA.

Idealized State 1: Waste Package Structural Response with Structurally Stable Drip Shield

Modeling Assumptions and Approach

In SAR Section 2.3.4.5.2.1, DOE provided information on waste package structural response for the Idealized State 1 where the drip shield is structurally stable. DOE considered that dynamic impacts of the waste package on the rest of the EBS components may lead to waste package damage and rupture of the outer corrosion barrier. DOE evaluated the movement of and damage to waste packages resulting from seismic loads. The following three cases of impacts were considered using numerical models: (i) impacts between waste packages, (ii) impacts between the waste package and the emplacement pallet, and (iii) impacts between the waste package and the drip shield (SAR Section 2.3.4.5.2.1). DOE analyzed the TAD and the co-disposal (CDSP) waste packages for three waste package conditions where the drip shield is expected to remain functional and structurally stable (SAR Section 2.3.4.5.2.1). The three conditions are (i) 23-mm [0.91-in]-thick outer corrosion barrier with intact internals, (ii) 23-mm [0.91-in]-thick outer corrosion barrier with degraded internals, and (iii) 17-mm [0.67 in]-thick outer corrosion barrier with degraded internals. (DOE modeled a waste package with degraded internals as the waste package outer corrosion barrier only.)

DOE did not include the waste package damage potential for impact between the waste package and drip shield in the seismic damage abstractions. DOE's decision was based on the observations of the waste package damage from the analyses of impacts between waste packages (SAR Section 2.3.4.5.2, p. 2.3.4–131). DOE concluded that the waste package areas damaged as a result of a side impact on a flat elastic surface were zero or very small. These damaged areas were significantly less than the damaged areas from end impacts, as described in SNL (2007ay, Table 6-13). DOE stated that the waste package side impacts on a flat elastic surface are representative of the waste package impacts on the drip shield side wall.

DOE also stated that vertical impacts between the waste package and the drip shield would have a small contribution to the total waste package damage (SNL, 2007ay). DOE concluded that the impact loads on the waste package would be distributed over a large contact area of the drip shield bulkheads and stiffeners. DOE further concluded that vertical impacts between the waste package and the drip shield surrounded by rubble would be similar to impacts between waste packages, which also result in small damaged areas. Therefore, DOE concluded that the impact damage between waste packages is representative of the waste package damage from vertical impacts between the waste package and drip shield.

NRC Staff's Review

The NRC staff reviewed DOE's treatment of uncertainty of the waste package conditions, specifically, the three waste package conditions DOE analyzed. As mentioned under idealized State 1, DOE considered 23- and 17-mm [0.91- and 0.67-in] waste package outer corrosion barriers. These represent corrosion thinning of 2.4 and 8.4 mm [0.09 and 0.33 in], respectively, from the initial 25.4-mm [1-in] outer corrosion barrier thickness and correspond to a timeframe of approximately 340,000 and 1.2 million years after emplacement (reviewed in SER Section 2.2.1.3.1.3.2.1). The NRC staff concludes that the time duration DOE considered adequately covers the period of interest (i.e., 1 million years) for the mechanical disruption of engineered barriers. Therefore, the NRC staff concluded that DOE adequately took into account uncertainties in the waste package conditions and environmental effects on the waste package components for the period of interest. Consideration of uncertainty was accomplished through appropriate reductions of the waste package outer corrosion barrier thickness and degradation of the waste package internals.

The NRC staff compared the material properties of the EBS components DOE incorporated into the numerical models with the information available in the open literature (American Society of Mechanical Engineers, 2001aa) and finds that DOE used appropriate ranges of values of the mechanical properties for the EBS components.

The NRC staff also concludes that DOE's observation that the side-on impact on a flat, elastic surface is a good representation for the lateral impacts of the waste packages on the drip shield is reasonable because the inside surface of the drip shield wall is a smooth surface with no protruding bulkheads. On the basis of its review of DOE's analysis, the NRC staff finds that DOE's assumption that the results from impacts between waste packages bound the results for side or lateral impacts between the waste package and the drip shield.

The NRC staff finds acceptable DOE's conclusion that the vertical impact between the waste package and the drip shield would be similar to impacts between the waste packages and the pallets because the collision modes are similar. The NRC staff reviewed information DOE presented on the frequency of the vertical impacts between the waste package and the drip shield, including the representativeness of the damage modes used as input to the TSPA model in SNL (2007ay, Section 6.4.5). On the basis of this information, in 17 realizations of kinematic analyses at PGV levels of 1.05 and 2.44 m/sec [3.44 and 8 ft/sec], the number of impacts between the waste package and the drip shield bulkheads was less than 10, and at PGV levels of 4.07 m/sec [13.35 ft/sec] the number of impacts increased to 48. Note that the APE of a seismic event associated to the PGV of 4.07 m/sec [13.35 ft/sec] is on the order of 10^{-8}. The NRC staff concludes that because the frequency of occurrence for the vertical impacts between the waste package and the drip shield is low, the waste package damage due to these impacts would be small when compared to the waste package damage from waste package to waste package and waste package to pallet impacts. Therefore, the NRC staff finds that the assumption DOE used would not significantly affect TSPA results.

Computational Approach

To estimate waste package damage and rupture potential, DOE developed a two-part calculation process with numerical models developed using the computer code LS-DYNA (Livermore Software Technology Corporation, 2003aa).

First, large-scale kinematic analyses were performed to determine the impact parameters for multiple waste packages in an emplacement drift. The parameters included locations and time of impacts, relative velocity and impact angles, and forces between the impacting bodies. DOE used 17 ground motion time histories at PGV levels of 0.4, 1.05, 2.44, and 4.07 m/sec [1.31, 3.44, 8, and 13.35 ft/sec] in these analyses.

The large-scale kinematic calculations presented in the SAR consider a "string" of multiple waste packages. A combination of TAD and CDSP waste packages in a section of an emplacement drift was considered. For these analyses, DOE considered a partially or fully collapsed emplacement drift. The drip shield was considered to be in a structurally stable condition. Thus, the structurally stable drip shield provided the only restriction to the movement of the waste packages and the pallet. DOE recorded impacts for the central waste packages (three and two central waste packages for the TAD and CDSP configurations, respectively) in the total string of waste packages (SAR Section 2.3.4.5.2.1.3.1).

Second, DOE carried out detailed finite elements analyses for estimating damage and rupture potential. Impacts between individual waste packages and between waste package and pallet were analyzed. DOE evaluated waste package damage over the range of impact parameters, including those determined from the large-scale kinematic analyses. Using the results of the detailed finite element analyses, DOE estimated the waste package damage and rupture potential for the multiple impacts modeled using the large-scale kinematic analyses.

NRC Staff's Review

The NRC staff reviewed the modeling and computational approach DOE employed to evaluate the waste package response to vibratory ground motions while the drip shield is structurally stable. The NRC staff finds the ground motions used are consistent with the values presented on the bounded hazard curve in SAR Figure 2.3.4-18. The NRC staff also finds that DOE followed established industry practice for performing finite element analyses (Bathe, 1996aa) of mechanical/structural components of the waste package. The NRC staff finds reasonable and acceptable the geometric representation of the waste package and its components and associated simplifications made to the waste package geometry as the methodologies used are consistent with standard practice in finite element modeling. The NRC staff finds that the finite element models were appropriately used for characterizing waste package damage as input to the TSPA calculations. Further, the NRC staff finds that use of the results for the central waste packages from a string of waste packages would be representative because they would not be affected by the model boundaries along the emplacement drift direction.

Further Details of Applicant's Damage Analyses for the Case of Intact Drip Shield (Idealized State-1)

DOE stated that the waste package pallet eventually fails as the stainless steel connector tubes lose their structural integrity (SAR Section 2.3.4.1). For the damage analyses, however, DOE made an assumption that the waste package pallet is intact. This assumption, according to DOE, would lead to greater damage to the waste package outer corrosion barrier during vibratory ground motion. As DOE explained, the reason for this conclusion is that higher magnitude stresses are generated when the waste package impacts a "relatively stiff pallet as opposed to the crushed tuff invert" (SAR Section 2.3.4.1, p. 2.3.4–10). However, DOE initially did not consider that loss of connector integrity could result in a different range of conditions for pallet pedestal orientations, impact locations, and impact frequencies. The NRC staff questioned whether the variability in these parameters may have exceeded the range DOE considered. The NRC staff's review indicated that larger uncertainty in the pedestal orientation can potentially affect the calculated results. For example, impact locations, time of impact, relative velocity of the impacting bodies, angle of impacts, and forces between the impacting bodies could be affected. To clarify this question, DOE was requested to supplement the information presented in SAR Sections 2.3.4.5.2 and 2.3.4.5.4 to address whether such uncertainties would affect significantly the characteristics of waste package damage calculated in kinematic analyses.

In response to the NRC staff's RAI, DOE provided additional evaluation in DOE (2009bq, Enclosure 1) to demonstrate that the intact waste package pallet assumption did not underestimate the potential for waste package damage in the kinematic analyses. DOE stated that at lower PGV levels, the waste package and the pallet pedestals would have limited relative motion. Therefore, if the stainless steel connector tubes were to lose structural integrity due to corrosion, the waste package damage would remain bounded by the results of the analyses with the intact waste package pallet. DOE stated that, at higher PGV levels and degraded connector tubes, the impact between the waste package and pallet would be characterized by one of three cases: the waste package impacts both pallet pedestals (Case 1), the waste package impacts one pallet pedestal (Case 2), and the waste package impacts only the invert (Case 3).

For Case 1, DOE stated that the angles and locations of impacts would be similar to those used in the kinematic analyses with an intact pallet. For Case 2, DOE stated that the locations of the impacts would be toward the end of the waste packages, as the waste package would tend to slide off the remaining pedestal and onto the invert. In SNL (2007ap, Tables 6-49 and 6-50), DOE stated that, due to higher waste package stiffness at the waste package lid, the waste package would experience less damage for impacts near the waste package lids than in the middle of the waste package. DOE concluded that for Case 2, the waste package damage would be bounded by the results of the analyses for waste package impacts with an intact pallet. For Case 3, DOE stated that the waste package damage would be bounded by the results of the intact pallet. This result is due to the waste package experiencing less damage from impact forces distributed over a larger waste package area. Therefore, DOE concluded that the results of the analyses with an intact pallet would bound the waste package damage for the case of structural integrity loss from corroded stainless steel connector tubes.

NRC Staff's Review

The NRC staff reviewed DOE's damage analyses and responses to the NRC staff's RAI and on the basis of its review concludes that

- The waste package damage from impacts with an intact waste package pallet would bound the waste package damage from impacts with two separated pallet pedestals because the contact area of impact would not change.

- The waste package damage from angular impacts with an intact waste package pallet would be representative for the waste package damage with a single-pallet pedestal because such an impact causes more damage than other impacts.

- The waste package damage from impacts with an intact waste package pallet would bound the waste package damage with the invert because the area of impact would increase and, as a result, reduce the stresses in the waste package outer corrosion barrier.

DOE considered waste package damage from angular impacts with an intact waste package pallet and concluded that the waste-package-to-pallet impacts are likely to cause more waste package damage than other types of impacts. The NRC staff's review determined that DOE did not sufficiently address the potential change in the frequency of this type of impact due to pallet degradation. However, the NRC staff concluded that the number of waste package angular impacts would not significantly increase because of pallet degradation. This conclusion is because the likelihood of waste package angular impacts on a single pallet pedestal is only feasible at very high PGV levels due to the close proximity of the waste packages to each other and other EBS components within drip shield boundaries. On the basis of its review of the applicant's seismic hazard curves, the NRC staff finds acceptable DOE's view that the APE of a seismic event associated with such high PGV levels is very low. Therefore, the NRC staff finds the modeling and computational approach DOE used acceptable, as it would not significantly affect the results of the TSPA calculations.

Consideration of Residual Stress Failure Mode

To analyze the residual stress failure mode, DOE calculated the total damaged area of the waste package. Total damaged area is defined as the sum of the areas of all outer corrosion barrier elements in which the stress exceeds a threshold stress level at the end of a simulation. The three residual stress threshold values used are 90, 100, and 105 percent of the yield strength. (The NRC staff's evaluation of the residual stress threshold values DOE used to estimate the waste package damaged area is presented in SER Section 2.2.1.3.1.3.2.3.)

DOE used results from the analyses of the impacts between waste packages and between the waste package and the pallet. Inputs for the TSPA calculations were prepared in the form of lookup tables that provided damaged area as a function of the impact parameters. According to the information provided in these lookup tables (SNL, 2007ap), the amount of damage for single impacts is largest for impacts between a waste package and a pallet. The damage increases with a decrease in the outer corrosion barrier thickness. The reported damage area for single impacts ranged from 0.002 to 14.333 percent of the total surface area for the TAD waste package and from 0.002 to 20.106 percent for the CDSP waste package.

For the analyses with multiple waste packages, the amount of reported damage is largest for impacts between a waste package and a pallet. The damage increases with an increase in PGV levels and a decrease in the outer corrosion barrier thickness. The reported damage area ranged from 0.006 to 43.467 percent of the total surface area for the TAD waste package. The range for the CDSP waste package, used by DOE, was from 0.006 to 19.585 percent of its surface area (SNL, 2007ap).

NRC Staff's Review

The NRC staff reviewed the residual stress failure mode results for these analyses and considered DOE's response to an RAI (DOE, 2009br) that addressed whether the intact waste package pallet assumption would not underestimate the potential for waste package damage in the kinematic analyses. The NRC staff finds that DOE followed established industry practice in performing these finite element analyses, incorporated acceptable simplification and defensible assumptions, and used appropriate loading conditions to characterize the waste package damage. Therefore, the NRC staff finds that the waste package damage results for the residual stress failure mode are technically defensible and appropriate for use as input to TSPA analyses.

Consideration of Tensile Tearing Failure Mode

To analyze the tensile tearing failure mode, DOE assessed the rupture condition for a single impact. The maximum effective strain in the waste package outer corrosion barrier for the full time-history analyses was compared with the rupture tensile strain failure criterion (SAR Section 2.3.4.5.2.1.3.2). DOE demonstrated through detailed finite element analyses that the strain for a single impact in the outer corrosion barrier was always below the ultimate tensile strain for Alloy 22 (SAR Section 2.3.4.5.2.1.3.2). For multiple impacts modeled in the large-scale kinematic analyses, DOE stated that if an impact causes "severe" deformation, the additional large impacts to the deformed area have the potential to cause rupture. For both the TAD and the CDSP waste packages with intact internals, DOE stated that the overall deformation of the outer corrosion barrier resulting from multiple impacts was insignificant even at the largest impact velocities. Therefore, DOE concluded that no rupture would occur (SAR Section 2.3.4.5.2.1.3.2).

For the analyses with degraded internals, DOE considered that the deformation from low-velocity impacts {PGV levels less than 1.05 m/sec [3.44 ft/sec]} was not severe enough to lead to rupture after multiple impacts. In addition, the deformation becomes very large as the impact velocity increases. For PGV levels of 1.05 m/sec [3.44 ft/sec] and higher, a second impact of equal or greater magnitude would potentially cause rupture of the outer corrosion barrier. Therefore, for the PGV of 1.05 m/sec [3.44 ft/sec] and higher, which have a mean APE of 10^{-5}, the waste package rupture probability exceeds zero. In some realizations of large-scale models for both the TAD and the co-disposal waste package configurations with degraded internals and PGV levels of 2.44 m/sec [8 ft/sec] and higher, DOE calculated the probability of rupture equal to one.

NRC Staff's Review

The NRC staff concludes that the failure criterion (i.e., ultimate tensile strain) DOE used to evaluate the waste package rupture occurrence from a single impact is consistent with acceptable industry practice and is widely used in the field of mechanical/structural engineering (American Society of Mechanical Engineers, 2001aa). DOE relied on engineering judgment to determine whether multiple impacts to the waste package result in tensile rupture (SAR Section 2.3.4.5.1.4.2). If the degree of deformation from a single impact was judged significant, a second impact of equal or greater magnitude was judged sufficient to cause tensile rupture. However, DOE initially did not describe the magnitude of stress or strain on the outer corrosion barrier, the impact velocities that caused this damage, or the threshold beyond which such damage occurs. The NRC staff determined that the SAR did not explain how variations in these or other indicators of damage were considered in the expert judgment process and therefore requested additional information.

In response to the NRC staff's RAI, DOE provided information in DOE (2009bq, Enclosure 2) to demonstrate the acceptability of the methodology that involves engineering judgment used in the qualitative evaluation of waste package rupture probability for multiple impacts. DOE performed a quantitative evaluation of the waste package rupture probability. The analysis is based on maximum effective strain limit and assessment of tensile strain in the waste package outer corrosion barrier. Because the quantitative approach did not predict waste package rupture, DOE developed a qualitative approach. This approach was based on an assessment of the outer corrosion barrier deformation. The deformation results were used to estimate the waste package rupture probability for multiple impacts. In SNL Figures 6-31 through 6-36 (2007ap), DOE examined deformation shapes of the outer corrosion barrier to determine a deformed state that could cause rupture if a second large impact occurred. For the analyses at an impact velocity of 5 m/sec [16.4 ft/sec], DOE stated that the outer corrosion barrier developed deformations sufficient to cause rupture at a subsequent seismic event. DOE defined this state as a lower bound such that another impact of 5 m/sec [16.5 ft/sec] or higher would cause rupture of the waste package outer corrosion barrier. DOE used impact force values associated with impacts at 5 m/sec [16.4 ft/sec] as a threshold force to define zero probability of waste package rupture due to multiple impacts. DOE defined the force associated with impacts at 7 m/sec [23 ft/sec] as an “upper force peg point” and used this to interpolate and extrapolate probability of waste package rupture between zero and one. DOE concluded that this qualitative method would not underestimate waste package rupture probability, because the force threshold used as a lower bound was derived on the basis of less severe and more frequent waste package deformations at impact velocities of 5 m/sec [16.4 ft/sec] and higher.

The NRC staff finds this threshold value is a reasonable bound such that another impact of 5 m/sec [16.4 ft/sec] or higher would rupture the waste package outer corrosion barrier. The NRC staff's conclusion is based on the following. The NRC staff finds acceptable DOE's conclusion that the waste-package-to-pallet impacts would lead to greater damage to waste packages than the waste-package-to-waste-package or the waste-package-to-drip-shield impacts. Further, for the waste-package-to-pallet impacts, the most damaging scenario is angular impacts at 6° angles into the middle of the TAD waste package with degraded internals (SNL, 2007ap). During its review, the NRC staff could not locate the information on PGV levels that would trigger impact velocities of 5 m/s [16.4 ft/sec] or higher for the waste-package-to-pallet impacts. In response to an NRC staff RAI, DOE identified the needed information on impact velocities of the drip-shield-to-waste-package impacts in SNL (2007ap, Tables 6-148 through 6-150). On the basis of this information, impact velocities of 4 m/sec [13.12 ft/sec] or higher are likely to occur only for seismic events at PGV levels of 4.07 m/sec [13.35 ft/sec] or higher. The NRC staff estimated that for a given PGV level, the waste-package-to-pallet impacts would exhibit impact velocities similar to those observed for the drip-shield-to-waste-package impacts. Therefore, the NRC staff concludes that impact velocities of 5 m/sec [16.4 ft/sec] are likely to occur only for seismic events with an APE of 10^{-8} or lower. Thus, subsequent seismic events capable of triggering these large impact velocities are unlikely and therefore beyond consideration for TSPA analyses. Moreover, the NRC staff also concludes that the waste package should have enough remaining capacity, after the first seismic event with impact velocities of 5 m/sec [16.4 ft/sec], to withstand subsequent seismic events at these impact velocities. This is based on NRC staff's review of DOE-provided numerical results indicating the waste package would not exhibit rupture for a single event at impact velocities of 10 m/sec [32.8 ft/sec], as described in SNL (2007ap, Table 6-63). Therefore, the NRC staff finds that the qualitative methodology used to evaluate waste package rupture probability for multiple impacts based on a 5-m/sec [16.4-ft/sec] impact velocity threshold is acceptable and would not significantly affect the results of TSPA analyses.

The NRC staff determined that DOE defined, in SNL (2007ap, Section 6.3.2.2.5), the maximum effective strain limit for the waste package rupture condition as 0.57 for uniaxial tension and 0.285 for biaxial tension. For realizations where the maximum effective strain was less than 0.285, DOE considered that rupture was not credible. When the maximum effective strain exceeded 0.285, the strain limit was multiplied by the triaxiality factor, resulting in an effective strain limit between 0.285 and 0.57. Finally, DOE evaluated the rupture condition on the basis of the newly computed strain limit. For some realizations, for which DOE concluded that the waste package did not rupture, the NRC staff noted that the computed maximum effective strains exceeded the effective strain limit [e.g., SNL (2007ap, Table 6-92)]. In response to the NRC staff's RAI, DOE stated in DOE (2009bq, Enclosure 3) that for all realizations with computed effective strain in the outer corrosion barrier greater than the uniaxial tensile strain limit of 0.57, as outlined in SNL (2007ap, Tables 6-60, 6-90, and 6-92), the stress state is compressive. Therefore, according to DOE, under these conditions, the waste package rupture would not occur. The NRC staff reviewed this information DOE submitted and finds acceptable the exclusion of the waste package rupture for these realizations of kinematic analyses because a compressive state of stress would not lead to waste package rupture. Therefore, the NRC staff finds that DOE has demonstrated that the waste package damage results for the tensile tearing failure mode are appropriate inputs for abstraction in its TSPA model.

Idealized State 2: Waste Package Structural Response under Collapsed Drip Shield

Modeling Assumptions and Approach

DOE provided information on waste package structural response for the Idealized State 2 with a collapsed drip shield framework (SAR Section 2.3.4.5.4.3.2). DOE assessed deformations and stresses in the outer corrosion barrier of a TAD waste package loaded by a collapsed drip shield and the accumulated rubble. Outer corrosion barriers that were 17 and 23 mm [0.67 and 0.91 in]-thick with intact and degraded internals were assessed. DOE's model represents the intact internals by the inner vessel, the TAD canister, and the fuel baskets with plates inside the canister. DOE assigned properties of Type 316 stainless steel to all internal components. The internals, which are assumed to be completely degraded, were represented by a material that can be considered to be similar to a weakly cohesive soil with no significant strength. This material fills the interior volume of the outer corrosion barrier to limit volume change to 50 percent.

DOE performed numerical analyses to assess the waste package structural response under a collapsed drip shield using the FLAC3D finite element models (SAR Section 2.3.4.5.4.3.2). In these analyses, the drip shield was not explicitly modeled and was represented by bulkhead flanges that contact the waste package after collapse of the drip shield framework. DOE conducted these quasi-static analyses by applying vertical static loads to the drip shield bulkheads. The vertical loads were monotonically increased until pressures ranging from 500 to 1,500 kPa [10,400 to 31,300 psf] were reached. DOE considered that the average vertical static pressure from lithophysal rockfall for a complete drift collapse exerted onto the drip shield is 127 kPa [2,652 psf] (SAR Table 2.3.4-35). For the drip shield average vertical loading demand of 127 kPa [2,652 psf] (SAR Section 2.3.4.5.4.3.2.1), the maximum quasi-static pressures applied to the waste package are equivalent to PGAs in the range of about 3 to 9 g ("g" is acceleration due to gravity). DOE monitored the structural deformations and the residual stresses induced in the outer corrosion barrier as a function of the average vertical pressure exerted on the outer corrosion barrier by the drip shield bulkhead flanges.

NRC Staff's Review

The NRC staff evaluated the modeling approach for Idealized State 2 that DOE employed to evaluate the waste package response under quasi-static loading under the collapsed drip shield. The NRC staff concludes that for characterizing waste package damage, DOE followed established industry practice for mechanical/structural performance assessment using finite element methods (Bathe, 1996aa). Further, the NRC staff finds that DOE used the modeling calculations and represented the waste package and the drip shield and their component geometries, including geometry simplifications, appropriately.

The NRC staff finds that representing the drip shield by bulkhead flanges is acceptable because the damage-causing contact between the collapsed drip shield and the waste package is likely to occur between the drip shield bulkhead flanges and the waste package outer corrosion barrier. For Idealized State 2 analyses, DOE assumed that drip shield components have zero contact angles (i.e., lie flat) on the waste package outer corrosion barrier when vertical loads are applied. However, DOE initially did not provide a basis to support the conclusion that the drip shield components would have a zero contact angle with the waste package if the drip shield framework collapses. In addition, the initial information in the SAR did not address how uncertainties in contact angle that result from differential deformation of the drip shield (e.g., partial framework collapse) or tilting of the waste package (e.g., due to the waste package emplacement pallet degradation) could affect the analyses for waste package damage. The NRC staff also considered that localization of stress from angular impacts may affect the localization of tensile strain in the outer corrosion barrier and, thereby, increase the likelihood of puncture or rupture (e.g., SAR Section 2.3.4.5.4.4.2).

In response to the NRC staff's RAI that addressed these issues, DOE provided additional information in DOE (2009br, Enclosure 1) to demonstrate the adequacy of its modeling approach. DOE provided waste package damage estimates that bounded waste package damage for angular impacts of the drip shield onto the waste package outer corrosion barrier. DOE stated that a partially collapsed drip shield could result in angular contact between the waste package outer corrosion barrier and the drip shield bulkhead. According to DOE, a partially collapsed drip shield does not completely lose its load-bearing capacity. DOE stated that a modeling approach that allows the drip shield to fully collapse onto the waste package (i.e., a modeling approach that produces a zero contact angle between the waste package outer corrosion barrier and the drip shield bulkheads) would overestimate the total load transferred to the waste package and, therefore, would overestimate waste package damage.

The NRC staff reviewed this information and concluded that the overall modeling approach DOE used is acceptable for the following two reasons. First, for the residual stress failure mode, although an angular impact of the drip shield onto the waste package outer corrosion barrier could result in localization of stresses, the stress concentration areas would also be reduced. This would reduce the waste package outer corrosion barrier surface area that may be susceptible to stress corrosion cracking. Second, for the tensile tearing failure mode, DOE's analyses that do not consider an angular impact of the drip shield may underestimate the tensile tearing stresses. However, tensile rupture of the outer corrosion barrier would not occur for Idealized State 2, even for nonzero contact angles. This conclusion is based on the results of independent studies the NRC staff performed (Ibarra, et al., 2007aa, ab; Pomerening, et al., 2007aa). These independent studies showed that given the high ductility of Alloy 22, the waste package outer corrosion barrier would not be breached for the loads considered in the Idealized State 2 and nonzero angular impact. Therefore, the NRC staff finds that DOE's modeling approach is acceptable, because it would not significantly affect the results of TSPA calculations.

Consideration of Residual Stress Failure Mode

For the residual stress failure mode, DOE calculated the total damaged area as the sum of areas of all outer corrosion barrier elements (including interior and exterior surfaces) in which a single residual stress threshold of 90 percent of the Alloy 22 yield stress is exceeded (SAR Section 2.3.4.5.4.3.2). The NRC staff's confirmatory evaluation of the residual stress threshold values to estimate the waste package damaged area is presented in SER Section 2.2.1.3.1.3.2.3. For the analyses with a 17- or 23-mm [0.67- or 0.91-in]-thick outer corrosion barrier with intact internals, DOE made the following observations:

- The damaged area was less than 0.025 percent of the total outer corrosion barrier surface area for average vertical pressure up to 1,200 kPa [25,062 psf].

- The maximum damaged area was approximately 0.3 percent or less of the total outer corrosion barrier surface area for the highest evaluated vertical pressure of 1,500 kPa [31,328 psf].

- For the analyses with degraded internals, the vertical pressure of about 660 and 1,000 kPa [13,784 and 20,885 psf] may lead to a fully damaged waste package for 17- and 23-mm [0.67- and 0.91-in]-thick outer corrosion barriers, respectively.

- For vertical pressure of less than or equal to 350 kPa [7,309 psf], the waste package damaged area was less than 0.1 percent of the total area (SAR Figure 2.3.4-93).

NRC Staff's Review

The NRC staff, on the basis of its review of the residual stress failure mode, finds that DOE followed established industry practices in performing these finite element analyses, made reasonable assumptions, incorporated reasonable simplifications, and used appropriate loading conditions to characterize the waste package damage. The NRC staff further finds that the results for the waste package damage DOE presented are technically defensible and, therefore, acceptable as input to the TSPA calculations. DOE's results are consistent with earlier studies by the NRC staff related to deformation shapes, strains, and stresses (Ibarra, et al., 2007ab).

Consideration of Tensile Tearing Failure Mode

For the tensile tearing failure mode, DOE provided information on the maximum stresses in the waste package outer corrosion barrier for three vertical pressure levels: 486, 807, and 1,483 kPa [10,150, 16,854, and 30,972 psf]. According to this information, the maximum stresses in the outer corrosion barrier did not exceed 420.4 MPa [8,779,959 psf], (SAR Figures 2.3.4-91 and 2.3.4-92), which is below the Alloy 22 ultimate tensile strength of 786 MPa [16,415,431 psf], as detailed in SNL (2007ap, Table 4-3).

NRC Staff's Review

The NRC staff reviewed the tensile tearing failure mode results for three vertical pressure levels analyses using YMRP Section 2.2.1.3.2 and concluded, on the basis of the reasonableness of DOE's assumptions related to material behavior, establishment of initial and boundary conditions for the abstraction models and comparison with NRC staff's independent studies (Ibarra, et al., 2007ab), that the results are technically defensible. The NRC staff concluded that DOE adequately demonstrated that rupture of the waste package outer corrosion barrier would not occur at the three vertical load levels selected for analyses (SAR Section 2.3.4.5.4.3.2) because the maximum stresses were below the ultimate tensile strength of Alloy 22 by a comfortable margin.

Assessment of Collapsed Drip Shield Condition (Idealized State 2)

For Idealized State 2, with a collapsed drip shield framework, DOE concluded that

- Idealized State 2 bounds the case with intact waste package internals [tensile strain calculations from dynamic loads due to rock rubble after drip shield plate failure (Idealized State 3, which is reviewed in the section to follow)]

- Idealized State 1 bounds State 2 for the case with degraded internals (the kinematic analyses for TAD waste packages)

However, DOE initially did not present the model results for tensile strains of the waste package after drip shield collapse (SAR Section 2.3.4.5.4.4.1). In addition, DOE initially did not discuss how free interactions between the waste package and drip shield, or dynamic interactions with rock rubble, appropriately bound localized tensile strains that could occur between a collapsed drip shield and the waste package. In response to the NRC staff's RAI, DOE provided additional information intended to demonstrate that its results were bounding (DOE, 2009bs). DOE discussed how the free interactions between the waste package and drip shield and dynamic interactions with rock rubble appropriately bound localized tensile strains that could occur between a collapsed drip shield and the waste package.

DOE performed a quantitative comparison of the maximum effective plastic strain results of (i) the kinematic analyses for impacts between the waste package and the pallet with degraded internals and (ii) the quasi-static analyses for the waste package with degraded internals loaded by a collapsed drip shield. On the basis of this comparison, DOE concluded that the maximum effective plastic strains from the kinematic calculations of impacts between a waste package and a pallet with degraded internals were greater. Thus, DOE concluded that the results bounded the effective plastic strains for the waste package with degraded internals loaded by a collapsed drip shield. In addition, DOE performed a quantitative comparison of the maximum effective plastic strain results of the kinematic analyses for the waste package surrounded by rubble and the quasi-static analyses for the waste package with its intact internals loaded by a collapsed drip shield. DOE concluded that the effective plastic strains from the calculations for the waste package surrounded by rubble were greater and, therefore, bounded the effective plastic strains for the waste package with intact internals loaded by a collapsed drip shield.

NRC Staff's Review

The NRC staff reviewed the information DOE provided and compared quantitative results of the effective plastic strain DOE provided, and on the basis of this, confirmed that the effective plastic strain results of both kinematic analyses bound the results of quasi-static analyses. Therefore, the NRC staff finds acceptable DOE's technical bases that demonstrate that free interactions between the waste package and drip shield and dynamic interactions with rock rubble appropriately bound localized tensile strains that could occur between a collapsed drip shield and the waste package. Further, the NRC staff finds acceptable DOE's waste package performance for the Idealized State 2 (waste package loaded by a collapsed drip shield framework) as representative input in the TSPA evaluation.

Idealized State 3: Waste Package Structural Response in Direct Contact with Rubble

Modeling Assumptions and Approach

DOE provided information on waste package structural response for Idealized State 3 where the waste package is in direct contact with rock rubble (SAR Section 2.3.4.5.4.3.1). DOE considered the loads produced by the weight of the rock rubble and the amplification of these loads during vibratory ground motion. These loads may lead to waste package damage through stress-corrosion cracking, or rupture and puncture of the outer corrosion barrier. To examine the waste package damage potential, DOE performed mechanical/structural analyses of the TAD waste package in direct contact with the rubble. Two waste package outer corrosion barrier thicknesses of 17 and 23 mm [0.67 and 0.91 in] with degraded internals were considered. The system was subjected to static loads and dynamic amplification from ground motions with PGV levels of 0.4, 1.05, 2.44, and 4.07 m/sec [1.31, 3.44, 8, and 13.35 ft/sec] (SAR Section 2.3.4.5.4.1). These PGV values correspond to the ground motions presented on the bounded hazard curve in SAR Figure 2.3.4-18.

DOE conducted two-dimensional seismic analysis of the waste package surrounded by rubble using the computer code UDEC. The UDEC model initially represented an intact emplacement drift containing a waste package and pallet resting on the invert. The drift was allowed to collapse onto the waste package. Once static equilibrium was established, the model was subjected to ground motions and equilibrium was reestablished. DOE used a complete drift collapse simulation similar to the one used to assess potential drip shield framework buckling and drip shield plate rupture (SAR Section 2.3.4.5.3.2.1). The results included residual tensile stresses and effective tensile strains in the outer corrosion barrier. General observations on the deformed shapes of the outer corrosion barrier were also provided. DOE did not include the inner vessel, the TAD canister, or the fuel baskets in the waste package representation and only considered the degraded state of the waste package internals. DOE represented the degraded internals as a material similar to a weak cohesive soil with no significant strength. DOE stated that, for the geometrical representation used, the results of the TAD waste package provided a reasonable estimate of damage for the CDSP waste package as well. Therefore, separate models were not developed for the TAD and CDSP waste packages.

DOE used a two-dimensional plane strain representation of the waste package and its components for dynamic analyses under rubble loads, as outlined in SNL (2007ap, pp. 6–216). This simplification assumes that the waste package extends infinitely in the direction normal to the calculation plane and that the structural response of the waste package is not affected by its boundaries in that direction (i.e., waste package lids). In SNL (2007ap, Appendix D), DOE compared results of two-dimensional and three-dimensional stress analyses, using uniform static loadings that are not representative of the dynamic loads associated with seismic events. Because of the higher rigidity of the waste package lid area, the NRC staff considered that the outer corrosion barrier area in the vicinity of the waste package lid potentially could be more susceptible to tensile tearing than an open cylinder.

In response to the NRC staff's RAI, DOE provided additional information in DOE (2009bt) to demonstrate that the use of a two-dimensional waste package representation in seismic analyses of the waste package surrounded by rubble did not underestimate waste package damage. DOE stated that the two-dimensional waste package representation had reduced stiffness because the waste package lids that provide additional structural support were not included. The two-dimensional waste package representation was chosen because it maximizes structural deformation of the outer corrosion barrier. Further, DOE stated that three-dimensional waste package analyses were performed to investigate the potential for failure of waste package lids and connections between the waste package lids and the waste package wall. DOE concluded that these analyses demonstrated that tensile rupture of the outer corrosion barrier would only occur when the outer corrosion barrier collapses due to the waste package wall buckling, as described in SNL (2007ap, Appendix D). DOE stated that because the two-dimensional waste package representation underestimates the loading demands needed for an outer corrosion barrier collapse, DOE concluded this representation would not underestimate the potential for the waste package outer corrosion barrier tensile failure.

NRC Staff's Review

The NRC staff reviewed the analyses and concluded that DOE provided acceptable technical bases to demonstrate that the two-dimensional waste package representation in seismic analyses of the waste package surrounded by rubble did not underestimate the waste package damage for the residual stress failure mode and the waste package puncture probability. The NRC staff's conclusion is based on the following. The waste package damage for the residual stress failure mode and waste package puncture probability are functions of waste package deformations. The NRC staff finds that this is a conservative approach to analyzing the performance because DOE used a two-dimensional waste package representation, and the kinematic analyses of the waste package surrounded by rubble would overestimate the waste package deformations. As a result, the NRC staff concludes that the waste package damage for the residual stress failure mode and waste package puncture probability would also be overestimated.

The NRC staff reviewed the kinematic analyses of the waste package surrounded by rubble analyses and concluded that this modeling approach may underestimate the tensile tearing stress in the waste package outer corrosion barrier near the waste package lid. However, the NRC staff concluded that tensile rupture of the outer corrosion barrier in these locations is not likely to occur for the Idealized State 3 loading scenario. This conclusion is based on the following. In Idealized State 3, DOE performed kinematic analyses for the same set of ground motion time histories used for the Idealized State 1 evaluation. However, in Idealized State 3, the waste package is surrounded by rubble and the dynamic impacts would be distributed over a larger waste package contact area than in Idealized State 1. Redistribution of impact loads would reduce the potential for high strain/stress concentration regions and thus result in subsequent reduction in waste package damage. The NRC staff independently verified that the largest distributed impact forces on the waste package in Idealized State 3 do not exceed the maximum forces evaluated in the kinematic analyses for Idealized State 1. Moreover, in Idealized State 1, DOE adequately demonstrated that waste package rupture is not likely to occur. As a result, the NRC staff finds acceptable DOE's assertion that waste package rupture is not likely to occur in Idealized State 3. Thus, the NRC staff finds acceptable the modeling approach DOE used because the model does not underestimate waste package damage for TSPA abstractions.

Consideration of Residual Stress Failure Mode

For the residual stress failure mode, DOE concluded that the damaged area was generally a small percentage of the total waste package surface area. For the residual stress threshold of 90 percent of the yield stress, the damaged area resulted in 0.2 percent of the total waste package outer corrosion barrier surface area. For the residual stress threshold of 105 percent of the yield stress, the damaged area was about 3 percent of the total outer corrosion barrier surface area. DOE stated that the increase in damaged area correlated with an increase in PGV levels and thinning of the outer corrosion barrier.

NRC Staff's Review

The NRC staff reviewed the results of the residual stress failure mode in DOE's analyses and DOE's response to the NRC staff's RAI (DOE, 2009bt) that demonstrated that two-dimensional waste package representation maximizes structural deformation of the outer corrosion barrier. Because higher waste package deformations would lead to higher residual stresses in the waste package outer corrosion barrier, the NRC staff finds acceptable the results for the residual stress failure mode because they do not underrepresent waste package damage. Therefore, the NRC staff finds that DOE appropriately represented the waste package damage results for the residual stress failure mode as input to the TSPA code.

Consideration of Tensile Tearing Failure Mode

For the tensile tearing failure mode, DOE concluded in SNL (2007ap, Section 6.5.1.4.1) that the probability of rupture for the TAD and CDSP waste packages surrounded by rubble for the 17- and 23-mm [0.67- and 0.91-in]-thick outer corrosion barrier with degraded internals is zero. DOE's conclusion was based on the observation that, for all simulations, the maximum effective plastic strain was below the ultimate tensile strain of Alloy 22.

For this idealized state, in addition to rupture probability, DOE calculated puncture probability of the waste package outer corrosion barrier. DOE considered that a severely deformed outer corrosion barrier may be punctured by the sharp edges of fractured or partially degraded internal components. DOE calculated a potential for puncture of the outer corrosion barrier. The calculation considered the reduction in the final cross-sectional area of a severely deformed outer corrosion barrier, as identified in SNL (2007ap, Section 6.5.1.4.1). DOE assumed that the probability of outer corrosion barrier puncture is zero until deformation of the waste package outer corrosion barrier is such that the diameter is reduced by 10 cm [4 in], as outlined in SNL (2007ap, pp. 6–234). According to DOE, the puncture of the waste package outer corrosion barrier increased with the increase in PGV and with decrease in the outer corrosion barrier thickness. Reported rupture probability ranges were from 0.01 to 0.82 for the 23-mm [0.91-in]-thick outer corrosion barrier with degraded internals and from 0.05 to 1.00 for the 17-mm [0.67-in]-thick outer corrosion barrier.

NRC Staff's Review

The NRC staff reviewed DOE's analysis of the tensile tearing failure mode. On the basis of its review, the NRC staff finds that DOE used a failure criterion (i.e., ultimate tensile strain to evaluate the waste package rupture occurrence) that is consistent with accepted industry practice and one that is widely used in the field of mechanical/structural engineering (American Society of Mechanical Engineers, 2001aa). In SNL (2007ap, Section 6.5.1.2.2), DOE assessed the effective plastic stresses and strains of the final waste package configuration after reestablishing equilibrium. However, in the SAR, DOE did not explain whether effective stresses and strains were assessed at intermediate steps during the dynamic loading simulations. Because of the reversal of dynamic loading during modeled seismic events, the NRC staff questioned whether the effective plastic stresses and strains of final waste package configurations, after reestablishment of equilibrium, are consistent with the maximum effective plastic stresses and strains that occur during dynamic simulations.

In response to the NRC staff's RAI, DOE provided additional information in DOE (2009br, Enclosure 2) to demonstrate that using stresses and strains computed at the end of dynamic analysis is appropriate and does not underestimate damage to the waste package. DOE stated that, in the dynamic analyses of the waste package surrounded by rubble, the code cumulatively computes the effective plastic strain, and the plastic strains increase during the analyses' time history. In addition, DOE stated that the effective plastic strain is larger than the effective strain for the analyses with strain reversals and loading/unloading transitions. Therefore, DOE concluded that the use of effective plastic strain value obtained at the end of dynamic analyses is appropriate to evaluate the waste package damage. DOE stated that this approach would not underestimate the waste package rupture probability. The NRC staff reviewed this information, and on the basis of the reasonableness of the modeling assumptions and initial and boundary conditions used in DOE's analyses, finds DOE's technical bases acceptable. Specifically, the NRC staff finds that the evaluation of the stresses and strains at the end of dynamic analysis would not underestimate waste package damage. Thus, the NRC staff finds acceptable the analyzed waste package damage results for the tensile tearing failure mode.

Treatment of Puncture Probability

In calculating the waste package puncture probability, DOE assumed that the probability of the waste package outer corrosion barrier puncture is zero until deformation reaches a preset value. A waste package diameter reduction of 10 cm [4 in] was selected as the preset limit, as identified in SNL (2007ap, pp. 6–234). Use of this assumption implies that the cross-sectional area of the outer corrosion barrier must decrease by 11 percent before the probability of puncture exceeds zero. This percentage decrease can be calculated from the ratio of the design basis waste package outer diameter to the waste package outer diameter reduced by 10 cm [4 in]. For the highest PGV level used, DOE calculated the probability of the 17-mm [0.67-in]-thick outer corrosion barrier puncture to be 0.20. This implies that 20 percent of the waste packages would be punctured during a seismic event at a 4.07-m/sec [13.35-ft/sec] PGV level.

NRC Staff's Review

The NRC staff reviewed information DOE provided on treatment of puncture probability. On the basis of its review, the NRC staff determines that if DOE assumed that the waste package puncture probability exceeds zero for any deformation of the outer corrosion barrier, then the number of punctured waste packages during a seismic event at a 4.07-m/sec [13.35-ft/sec] PGV level would only increase by 2 percent. Moreover, because the APE for a 4.07-m/s [13.35-ft/sec] PGV level is 10^{-8}, this difference in the waste package puncture probability during the postclosure period would reduce even further. Therefore, the NRC staff finds acceptable DOE's conclusion that the assumption would not significantly affect TSPA results.

In SAR Section 2.3.4.5.4.3.1.2, DOE stated that for a residual stress threshold of 90 percent of the yield stress, the damage area resulted in 0.2 percent of the total waste package outer corrosion barrier surface area. For a residual stress threshold of 105 percent of the yield stress, the damaged area was 3 percent. The NRC staff noted inconsistencies between this information and that provided in SAR Figure 2.3.4-89 and sought clarification. DOE provided the following clarification in its response to the NRC staff's RAI, as outlined in DOE (2009bq, Enclosure 4) that addresses these inconsistencies:

"The DOE agrees that the percentages cited in the fourth and fifth sentences in SAR Section 2.3.4.5.4.3.1.2 are inconsistent with SAR Figure 2.3.4-89. The DOE will correct the numerical values in the fourth and fifth sentences of SAR Section 2.3.4.5.4.3.1.2 to read as follows:

If the residual stress threshold (RST) is 90% of the yield strength, the average damaged area is less than 1.2% of the total outer corrosion barrier surface area. If the RST is 105% of the yield strength, the average damaged area is less than 0.1% of the surface area.

These numerical values are consistent with SAR Figure 2.3.4-89 and with the data in Mechanical Assessment of Degraded Waste Packages and Drip Shields Subject to Vibratory Ground Motion (SNL 2007ap, Table 6-163)."

Summary of the NRC Staff's Evaluation of Waste Package Mechanical/ Structural Performance

The NRC staff reviewed DOE's information related to the mechanical disruption of engineered barriers (MDEB) abstractions for the waste package performance assessment and makes the following conclusions:

- For Idealized State 1, where the drip shield is structurally stable, DOE concluded that (i) dynamic impacts of the waste package with the rest of the EBS may cause damage to the waste package from end-to-end impacts between waste packages and between waste package and pallet; and (ii) the extent of waste package damage for TSPA abstractions is a function of the waste package type, the state of waste package internals, PGV levels, and the outer corrosion barrier thickness. The NRC staff finds that DOE's conclusions are consistent with the analyses presented in the SAR and other supporting documents and that DOE adequately represented waste package performance for Idealized State 1 in the TSPA analysis.

- For Idealized State 2, with a collapsed drip shield framework, DOE concluded that (i) for the case with intact waste package internals, the waste package damage estimated for the Idealized State 3 is bounding; and (ii) for the case with degraded internals, the waste package damage estimated for the Idealized State 1 is bounding. The NRC staff finds DOE's conclusions acceptable because DOE has demonstrated, by comparing the results for all three idealized states, that the extent of waste package damage for Idealized State 2 bounds the waste package damage for Idealized States 1 and 3.

- For Idealized State 3, where the waste package is in direct contact with rock rubble, DOE concluded that (i) a waste package with a 23-mm [0.91-in]-thick outer corrosion barrier and degraded internals will not be damaged under seismic events with peak ground velocities below 2.44 m/sec [8 ft/sec], and (ii) the waste package damage depends on the waste package outer corrosion barrier thickness and the PGV levels. The NRC staff finds these conclusions to be consistent with the analyses presented in the SAR and other supporting documents. The NRC staff finds DOE's waste package performance acceptable for Idealized State 3 in the TSPA abstractions.

The NRC staff further concludes that the technical bases for TSPA waste package abstractions presented in the SAR are adequately supported because

- DOE adequately considered processes and events such as outer corrosion barrier thinning and mechanical damage due to collisions during seismic events, in analyzing the degree of mechanical disruption of waste packages during the initial 10,000 year period, and projected these processes and events beyond the 10,000 year post-disposal period through the period of geologic stability.

- DOE adequately addressed uncertainties in the waste package conditions and the environmental effects on the waste package components

- For characterizing waste package damage, DOE followed established practice for mechanical and structural performance assessment

- DOE used appropriate seismic loading conditions that are consistent with the values presented on the bounded hazard curve

- To evaluate the waste package damage, DOE used failure criteria that are consistent with accepted industry practice and/or widely used criteria in the field of mechanical and structural engineering

- For calculating the residual stress and establishing tensile tearing failure modes, DOE used analytical and numerical methods that are appropriate for the types of analyses

In summary, the NRC staff finds acceptable DOE's technical bases for the waste package abstractions it used and that it adequately represented waste package performance in the TSPA abstractions.

2.2.1.3.2.4 Evaluation Findings

The NRC staff has reviewed SAR Section 2.3.4 and other information submitted in support of the license application, which includes information required by 10 CFR 63.21(c)(1)–(3), (9), (10), (15), and (19) related to mechanical and structural performance of EBS components, and finds, with reasonable expectation, that the relevant requirements of 10 CFR 63.114 and 63.342 are satisfied regarding the abstraction of mechanical disruption of engineered barriers in the performance assessment. In particular, the NRC staff finds that

- The inputs (such as mechanical properties of rocks, drip shield and waste package components, geological characteristics of the region surrounding the EBS, seismic inputs like ground motion and fault displacements and their associated probabilities of exceedance) DOE used in the process-level models are based on appropriate and sufficient data from the site and surrounding region and consider uncertainties and variabilities in parameter values, taking into account alternative conceptual models and analytical techniques for analyzing the mechanical disruption of engineered barriers, in compliance with 10 CFR 63.114(a)(1-3)

- Technical bases are provided by DOE for either the inclusion or exclusion of FEPs, including effects of degradation, deterioration, or alteration processes of engineered barriers that would adversely affect performance of the natural barriers, consistent with the limits on performance assessment in 10 CFR 63.342; and evaluations are of sufficient detail for those processes that would significantly affect repository performance [10 CFR 63.114(a)(4-6)]

- Adequate technical bases have been provided by DOE for process-level models propagated through abstractions and used in the TSPA, as required by 10 CFR 63.114(a)(7).

- DOE used performance assessment methods for the period of geologic stability consistent with the methods used to demonstrate compliance for the initial 10,000 years following permanent closure, in compliance with 10 CFR 63.114(b)

- DOE included in the performance assessment for the period of geologic stability those FEPs used for the initial 10,000-year period, consistent with specific limits, in compliance with 10 CFR 63.342(c)

2.2.1.3.2.5 References

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CHAPTER 6

2.2.1.3.3 Quantity and Chemistry of Water Contacting Engineered Barriers and Waste Forms

2.2.1.3.3.1 Introduction

Safety Evaluation Report (SER) Section 2.2.1.3.3 provides the U.S. Nuclear Regulatory Commission (NRC) staff's evaluation of the U.S. Department of Energy's ("DOE" or the "applicant") abstraction of the repository drift system that may alter the chemical composition and volume of water contacting the drip shield and waste package surfaces (NRC, 2005aa). DOE described this abstraction in Safety Analysis Report (SAR) Sections 2.3.5 and 2.3.7 (DOE, 2008ab). The NRC staff's evaluation focuses on key features, events, and processes (FEPs) that address the following topics: (i) the chemistry of water entering the drifts, (ii) the chemistry of water in the drifts, and (iii) the quantity of water in contact with the engineered barrier system (EBS). These three abstraction topics provide the input needed to model the features and performance of the EBS (e.g., drip shield and waste package) and their contributions to barrier functions. For example, in its SAR, DOE relied on corrosion tests that were conducted on waste package and drip shield materials under a range of geochemical environments. The range of aqueous testing environments the applicant used was based on a range of potential starting water compositions and from knowledge of near-field and in-drift processes that alter these compositions. Finally, the abstraction of other key FEPs related to this section is addressed in other sections of this SER. FEPs that address thermal-hydrologic processes affecting seepage rates are reviewed in SER Section 2.2.1.3.6, those that address corrosion processes affecting the drip shield and waste packages are reviewed in SER Section 2.2.1.3.1, and those that address the quantity and chemistry of water inside breached waste packages and the invert are reviewed in SER Section 2.2.1.3.4. Also, the review of the rationale for key FEPs that DOE has excluded from these abstractions is covered in SER Section 2.2.1.2.1.

2.2.1.3.3.2 Regulatory Requirements

The applicant is required to provide information related to the abstraction of the quantity and chemistry of water contacting engineered barriers and waste forms as specified in 10 CFR 63.21(c)(1), (9), (10), and (15). The requirements in 10 CFR 63.114 (Requirements for Performance Assessment) and 10 CFR 63.342 (Limits on Performance Assessments) include postclosure performance assessment to demonstrate compliance with 10 CFR 63.113 (Performance Objectives for the Geologic Repository after Permanent Closure). Compliance with 10 CFR 63.113 is reviewed in SER Section 2.2.1.4.1.

The regulations in 10 CFR 63.114 require, in part, that a performance assessment

- Include data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the surface and subsurface from the site and the region surrounding Yucca Mountain [10 CFR 63.114(a)(1)]

- Account for uncertainty and variability in the parameter values [10 CFR 63.114(a)(2)]

- Consider and evaluate alternative conceptual models [10 CFR 63.114(a)(3)]

- Provide technical bases for either the inclusion or exclusion of FEPs, including effects of degradation, deterioration, or alteration processes of engineered barriers that would adversely affect performance of the natural barriers, consistent with the limits on performance assessment in 10 CFR 63.342, and evaluate in sufficient detail those processes that would significantly affect repository performance [10 CFR 63.114(a)(4–6)]

- Provide a technical basis for the models used in the performance assessment to represent the 10,000 years after disposal [10 CFR 63.114(a)(7)]

The NRC staff's evaluation of DOE's inclusion or exclusion of FEPs is in SER Section 2.2.1.2.1. Requirements for performance assessment for the initial 10,000 years following disposal are specified in 10 CFR 63.114(a). Requirements for the performance assessment methods for the time from 10,000 years through the period of geologic stability, defined in 10 CFR 63.302 as 1 million years following disposal, are in 10 CFR 63.114(b) and 10 CFR 63.342. These sections provide that through the period of geologic stability, with specific limitations, the applicant

- Use performance assessment methods consistent with the performance assessment methods used to demonstrate compliance for the initial 10,000 years after disposal [10 CFR 63.114(b)]

- Include in the performance assessment those FEPs used in the performance assessment for the initial 10,000-year period [10 CFR 63.342]

The NRC staff's review of the SAR and supporting information follows the guidance in the Yucca Mountain Review Plan (YMRP), NUREG–1804 (NRC, 2003aa), Section 2.2.1.3.3, the Quantity and Chemistry of Water Contacting Engineered Barriers and Waste Forms, as supplemented by additional guidance for the period beginning 10,000 years after permanent closure (NRC, 2009ab). The YMRP acceptance criteria for model abstractions that provide guidance for the NRC staff's review of the applicant's abstraction of the quantity and chemistry of water contacting engineered barriers and waste forms follow:

1. System description and model integration are adequate.
2. Data are sufficient for model justification.
3. Data uncertainty is characterized and propagated through the abstraction.
4. Model uncertainty is characterized and propagated through the abstraction.
5. Model abstraction output is supported by objective comparisons.

The NRC staff used a risk-informed approach and guidance provided by the YMRP, as supplemented by NRC (2009ab), in its review of the SAR and supporting information to evaluate aspects of the quantity and chemistry of water contacting engineered barriers and waste forms important to repository performance. The NRC staff considered all five YMRP acceptance criteria in its review of information provided by DOE. In the context of these criteria, only those aspects of the model abstraction that substantively affect the performance assessment results, as determined by the NRC staff, are discussed in detail in this section. The NRC staff's determination is based both on risk information provided by DOE and on the NRC staff's knowledge gained through experience and independent analyses.

2.2.1.3.3.3 Technical Review

2.2.1.3.3.3.1 Chemistry of Water Entering Drifts

The abstraction for the chemistry of water entering drifts uses site-specific and literature-derived information as inputs to the applicant's near-field chemistry model, which simulates chemical interactions of minerals in the Yucca Mountain host rocks with pore waters that percolate downward toward the repository. The model calculates (i) a water–rock interaction parameter that is used to predict initial seepage water compositions important to drip shield and waste package corrosion; (ii) radionuclide solubility [key parameters are pH, ionic strength (I), and concentrations of chloride (Cl^-), nitrate (NO_3), and fluoride (F^-)]; and (iii) the range of in-drift carbon dioxide partial pressures (pCO_2). Important processes related to developing these parameters are discussed later in this section under the heading "Conceptual Model."

DOE used the results of its near-field chemistry model as inputs to other process-level models and direct inputs to the Total System Performance Assessment (TSPA) model. Potential seepage water compositions are used by the in-drift chemical and physical environment abstraction (reviewed in SER Section 2.2.1.3.3.3.2) and the waste package and drip shield corrosion abstraction at the process level (reviewed in SER Section 2.2.1.3.1). The near-field chemistry model predicts a range of in-drift pCO_2 values that are used to generate a lookup table in the TSPA model, which is sampled to provide inputs to the waste form degradation and mobilization abstraction (reviewed in SER Section 2.2.1.3.4). SER Section 2.2.1.3.6 reviews the abstraction that addresses thermal-hydrologic processes affecting seepage rates, SER Section 2.2.1.3.1 reviews the abstraction that addresses corrosion processes affecting the drip shield and waste packages, and SER Section 2.2.1.3.4 reviews the abstraction that addresses the quantity and chemistry of water inside breached waste packages and the invert.

In SAR Table 2.1-3 and DOE (2009an), DOE notes that the chemistry of water flowing into drifts is recognized as important to the barrier capability of the emplacement drift, one component of the EBS. Water chemistry is important to performance because some seepage waters can have compositions that affect the corrosion of engineered materials. Localized corrosion of the waste package may occur if seepage waters of appropriate chemistry contact the waste package when the waste package temperature exceeds 23.4 °C [74.1 °F] (i.e., the ambient drift temperature prior to waste emplacement). The NRC staff's review in SER Section 2.2.1.3.1 considers the conditions under which localized corrosion is not expected to occur. In DOE's proposed system of engineered barriers, titanium drip shields prevent seepage water from contacting the waste package. DOE predicts that the drip shields will maintain their capability until compromised by mechanical or corrosive failure. The NRC staff's review of SAR Section 2.3.6.8 concludes (see SER Section 2.2.1.3.1.3.1) that appreciable fluoride in the seepage water is needed to chemically compromise (by corrosion) the integrity of the drip shield. At DOE's predicted fluoride concentrations for nondisruptive scenarios, the drip shields will not appreciably corrode and will remain intact, as barriers to seepage contacting the waste package, during the first 250,000 years following closure. In DOE's model, as long as the drip shields remain intact, only slow general corrosion of waste packages occurs and only diffusional release of radionuclides is possible. With intact drip shields, significant amounts of seepage water are unlikely to contact the waste packages during the first 40,000 years following closure, when temperatures and relative humidity values are expected to exceed the conditions prior to waste emplacement (see SAR Figures 2.3.5-33 and 2.3.5-34). After 40,000 years, seepage water chemistries are predicted by DOE to return to compositions with dilute concentrations of dissolved components consistent with environmental conditions prior to waste emplacement.

The NRC staff determines that these predicted environmental conditions are consistent with NRC staff's independent analyses (CNWRA, 2007aa). The NRC staff's review concludes (SER Section 2.2.1.4) that these conditions will not significantly affect the mobility of radionuclides released from the waste package into the invert.

Mechanical processes are the other means by which drip shield performance may be compromised. DOE excluded early drip shield failure due to partial or complete collapse of drifts as a result of thermal effects (FEP 2.1.07.02.0A) on the basis of "low consequence," as explained in SER Section 2.2.1.2.1.3.2. The adequacy of the rationale for excluding this specific FEP is also reviewed in SER Section 2.2.1.2.1.3.2. DOE's performance assessment and the NRC staff's review conclude (SER Sections 2.2.1.3.1.3.1 and 2.2.1.3.2.6) that few drip shields are expected to suffer mechanical failure within 12,000 years after repository closure. The NRC staff considers this conclusion important because DOE calculated that conditions in the drift (e.g., temperature, pH, seepage water chemistry) may support localized corrosion of the waste package if the drip shield fails and allows seepage water to contact the waste package within approximately 12,000 years after repository closure, as described in DOE (2009dg, Enclosure 11). DOE calculated that beyond 12,000 years after repository closure, there is a low probability for conditions in the drift to support localized corrosion of the waste package, if the drip shield fails and allows seepage water to contact the waste package. This is the case even given a somewhat elevated temperature of the waste package. DOE's model calculated that both the pH and nitrate-to-chloride ratio of water that may contact the waste package will likely be too high to initiate localized corrosion beyond 12,000 years after repository closure.

The NRC staff reviewed the applicant's model for chemistry of water entering drifts in SAR Section 2.3.5.3, references therein, and applicant responses to requests for additional information (RAIs) (DOE, 2009an, 2009dg, 2009ck). Because of the limited potential for the chemistry of water entering the drifts to affect performance significantly, the NRC staff conducted a simplified review that focused on the fundamental aspects of this abstraction. This approach is consistent with YMRP guidance for conducting a risk-informed, performance-based review. Thus, the NRC staff's review focused on (i) the conceptual model, (ii) the initial range of pore water chemistries, (iii) the range of seepage water chemistries the near-field chemistry model predicted, and (iv) abstraction and integration. The review presented in this section is organized around these major topics and presented within the context of DOE's performance assessment evaluation. SER Section 2.2.1.3.3.3.2 provides an assessment of the chemistry of seepage water that may contact the waste package or enter the invert during the time period when conditions (e.g., temperature, pH, seepage water chemistry) in the drift are affected by heat generated from radioactive decay of the waste.

Conceptual Model

The NRC staff reviewed the information provided in SAR Section 2.3.5.3 and references therein to evaluate the adequacy of the conceptual model of the chemistry of water entering the drifts, including the description and model integration into the TSPA. A conceptual model, also referred to as an abstraction, is a collection of concepts and factors that describe and affect a certain process or outcome.

SAR Table 2.2-1 contains the FEPs that DOE believes are potentially relevant to the chemistry of water entering drifts. DOE evaluated and included the following FEPs in this abstraction: (i) Chemical characteristics of groundwater in the unsaturated zone (FEP 2.2.08.01.0B), (ii) Chemistry of water flowing into the drift (FEP 2.2.08.12.0A), and

(iii) Chemical effects of magma and magmatic volatiles (FEP 1.2.04.04.0B). DOE evaluated and excluded (on the basis of low probability or low consequence) the following FEPs from this abstraction: (i) Hydrothermal activity (FEP 1.2.06.00A), (ii) Altered soil or surface water chemistry (FEP 1.4.06.01.0B), (iii) Rind (chemically altered zone) forms in the near-field (FEP 2.1.09.12.0A), and (iv) Re-dissolution of precipitates directs more corrosive fluids to waste packages (FEP 2.2.08.04.0A). The exclusion of these FEPs from this abstraction is reviewed in SER Section 2.2.1.2.1.3.2. Furthermore, DOE's treatment of FEPs during the initial 10,000 years following permanent closure in this abstraction continues unchanged through the period of geologic stability (defined as 1 million years in 10 CFR 63.302).

DOE's conceptual model describes the chemical evolution of water as it percolates vertically toward the repository drifts. In the model, the water flows through the Topopah Spring repository host rock, a homogeneous unit that is 200 m [656.2 ft] thick, with average rock and hydrologic properties derived from measurements from equivalent units in the Yucca Mountain vicinity. Pore waters percolating through the unsaturated zone are modeled as chemically evolving by dissolution of alkali feldspar, which makes up about 60 percent of the host rock. Because of alkali feldspar's abundance, DOE assumed its dissolution represented host rock dissolution processes. The rate of feldspar dissolution increases as pore waters encounter elevated host rock temperatures that result from heat generated from radioactive waste decay.

After pore waters flow through the Topopah Spring rock to a location above the repository, the model calculates a chemical composition by combining one of four initial pore water compositions with an amount of feldspar predicted to have dissolved, and assuming chemical equilibrium with the minerals calcite and amorphous silica. Cation exchange onto clays or zeolites is considered implicitly. Gas phase carbon dioxide (CO_2) concentrations are controlled by contributions from the CO_2 present in the local aqueous phase, CO_2 released from the evaporation of water (containing dissolved CO_2), and the partial pressure of CO_2 ($10^{-3.5}$ atmospheres) in the atmosphere. Calcite precipitation and feldspar dissolution influence the aqueous phase concentration of CO_2. Temperature also has a strong effect on CO_2 because this gas partitions more strongly to the gas phase at elevated temperatures.

The NRC staff compared DOE's description of the near-field chemistry conceptual model with the NRC staff's understanding of primary mineral dissolution and secondary mineral precipitation processes that control the chemical evolution of pore water as it percolates downward through the Yucca Mountain natural system. On the basis of that comparison, the NRC staff concludes that while the included FEPs (SAR Table 2.2-1) are broad in nature, DOE included an appropriate level of detail regarding the important chemistry-affecting processes and provided adequate technical bases for their inclusion in the conceptual model, as further evaluated in these subsequent SER subsections: (i) Initial Range of Pore Water Chemistries, (ii) The Range of Seepage Water Chemistries Predicted by the Near-Field Chemistry Model, and (iii) Abstraction and Integration.

In addition, DOE considered the seismic and igneous intrusive scenarios in the abstraction of seepage water chemistry. The conceptual model for seepage water chemistry in the seismic scenario is the same as for the nondisruptive scenario described previously. For the igneous intrusive scenario, in which basaltic magma similar in composition to dikes found in the Yucca Mountain area fills much of the drifts, DOE considered the composition of seepage waters contacting the waste to be consistent with water that has reacted with basalt [BSC 2005ad, Section 6.3.1.3.5(a)]. DOE considered three basalt-influenced water compositions from large fractured basalt reservoirs (SAR Tables 2.3.7-10 and 2.3.7-11).

On the basis of the review of this information and knowledge of likely basalt–water interactions, the NRC staff concludes that the three basalt groundwaters considered in the SAR have chemical compositions that span the range of variation that could potentially enter a breached waste package following an igneous event and are therefore acceptable for use in model simulations. Additional discussion of basaltic magma and its influence on seepage water can be found in SER Section 2.2.1.3.10. DOE discussed the limited sensitivity of in-package chemistry to incoming water composition, whether from seismic scenarios or the igneous intrusive scenario in BSC [2005ad, Sections 6.5(a) and 6.6(a)]. Consequently, NRC staff concludes that the chemistries affected by igneous events, seismically induced ground motion, or fault displacement are adequate for their intended use in the SAR.

Initial Range of Pore Water Chemistries

This section addresses the characterization and propagation of data uncertainty through the model abstraction, and how the model abstraction output is supported by objective comparisons. DOE described input parameter development and parameter uncertainty in SAR Sections 2.3.5.3.2.2.1 and 2.3.5.3.2.2.2. The NRC staff reviewed the information provided in the SAR, references therein, and applicant responses to RAIs (DOE, 2009ck) to evaluate whether the model inputs for the chemistry of water entering the drift abstraction are adequate. This evaluation focused on evaluating the uncertainty in the range of initial pore water chemistry [especially pH, ionic strength (I), and concentrations of chloride (Cl^-) and nitrate (NO_3^-)] and carbon dioxide partial pressures (pCO_2). In SAR Section 2.3.5.5 DOE identified these parameters as important inputs to the abstractions that deal with drip shield and waste package corrosion.

DOE's near-field chemistry model considers four representative initial pore water compositions as inputs. DOE assumed these four compositions, which were derived from a compositional analysis of pore water samples, represent the range of potential pore water compositions expected for the entire Yucca Mountain repository. To determine the representative waters, a multistep screening process, based on charge balance and partial pressure of carbon dioxide, was used to evaluate 90 pore water sample compositions from Yucca Mountain cores that DOE deemed to be sufficiently complete for use in the near-field chemistry model. Thirty-four pore water sample compositions were identified as meeting the charge balance criteria (± 10 percent) and as having been minimally affected by microbial alteration (thus suitable for further consideration). The applicant performed a statistical cluster analysis on the sample compositions. Clustering analysis is a standard statistical method for finding clusters of data that are similar in some sense to one another. The members of a cluster are more like each other than they are like members of other clusters. The most typical case in a cluster is referred to as the "centroid." The applicant identified 4 distinct clusters, or groupings, from the 34 sample compositions. The sample with the composition closest to the centroid of each cluster was selected as representative of each cluster.

The 56 pore water sample compositions that DOE's screening process eliminated from consideration had a median chloride-to-nitrate ratio five times greater than the 34 samples found to be acceptable by DOE. DOE attributed the high chloride-to-nitrate ratios of the screened-out samples to the loss of nitrate by microbial activity during sample storage. The NRC staff evaluated DOE's support for the criteria used to screen the initial pore water compositions used as inputs to the near-field chemistry model (DOE, 2009ck). The NRC staff notes that while microbial activity during storage could result in high chloride-to-nitrate ratios, no evidence of such activity was presented by DOE. As a result, the NRC staff assessed the risk significance of any uncertainty that may have been introduced by excluding samples from the data set DOE used. The NRC staff compared the range and uncertainty in pore water compositions represented by DOE's 34 included pore water compositions with the NRC staff's understanding of the Yucca Mountain natural system obtained from independent analyses of unsaturated zone geochemical processes and field observations at Yucca Mountain (Pabalan, 2010aa). The independent analysis considered 156 pore water samples that the U.S. Geological Survey (USGS) collected from the unsaturated zone of Yucca Mountain. These 156 samples did not meet all of DOE's screening criteria for inclusion in the near-field chemistry model. However, the NRC staff concludes the data for these samples are sufficiently complete for this analysis, because the samples were characterized for pH, ionic strength, and chloride-to-nitrate concentration ratio. Thirty-three samples were selected to represent the range and distribution of the 156 pore water composition data set. This sample set represents a larger spatial distribution than DOE's samples and also bounds and enlarges the range of composition and concentration that DOE's 34 pore water compositions covered. The NRC analysis used StreamAnalyzer 2.0 and OLIAnalyzer 3.0 (Gerbino, 2006aa) thermodynamic software to simulate the evaporative evolution of pore seepage waters. The StreamAnalyzer software uses a different electrolyte solution thermodynamic model and a more comprehensive thermodynamic dataset than the EQ3/6 code DOE used. The results of the analysis indicated that evaporation of initially dilute pore waters forms brines with compositions that do not support localized corrosion of the waste package and general corrosion of the drip shield. On the basis of this analysis, the NRC staff concludes that considering a range of pore water compositions broader than the range DOE used in its near-field chemistry abstraction does not significantly affect the performance of the drip shield and waste package. Consequently, the NRC staff concludes that DOE's range of initial pore water compositions represented by the 34 pore water samples, while not bounding, is adequate for its intended use in DOE's TSPA model.

The Range of Seepage Water Chemistries Predicted by the Near-Field Chemistry Model

The NRC staff reviewed the information provided in SAR Section 2.3.5.3, references therein, and applicant responses to RAIs (DOE, 2009ck) to evaluate whether the implementation of the conceptual model of the chemistry of water entering the drifts is adequate. The NRC staff's review included an assessment of the sufficiency of data for model justification, and the characterization and propagation of model uncertainty through the model abstraction. As explained in SER Sections 2.2.1.3.1.3.1 and 2.2.1.3.2.6, the NRC staff finds acceptable DOE's determination that few drip shields will fail within 12,000 years after repository closure as a result of corrosion or mechanical failure. For the period 12,000 to 40,000 years after repository closure, the NRC staff expects that heat generated from radioactive decay of the waste would continue to affect conditions in the repository (i.e., temperature, relative humidity, pH, and chemical composition of in-drift waters). SER Section 2.2.1.3.3.3.2 assesses the chemistry of seepage water during the time period 12,000 to 40,000 years after repository closure, when conditions in the repository are affected by heat generated from radioactive decay of the waste. As discussed previously, after 40,000 years, the repository is expected to return to environmental conditions similar to those that existed prior to waste emplacement. Consequently, the repository environment is not expected to alter the chemical compositions of seepage waters the near-field chemistry model predicted would enter the drift. As a result, this review focuses on the range of chemistries the near-field chemistry model predicted after 40,000 years.

DOE used the near-field chemistry model to determine the potential chemical compositions of seepage waters entering the drifts during the thermal period and when conditions return to those prior to waste emplacement. The model uses a decoupled approach where hydrological and thermal processes are calculated independently. Chemistry is partly coupled to the thermal and hydrological processes through the dissolution of alkali feldspar. The chemical composition of potential seepage waters is calculated at a location above the repository and at the bottom of the model domain using the geochemical speciation and reaction path code EQ3/6.

The NRC staff reviewed the information provided in SAR Sections 2.3.5.3.3.3 and 2.3.5.3.3.5, references therein, and applicant responses to RAIs (DOE 2009ck) to evaluate the adequacy of model support for the range of seepage water compositions the near-field chemistry model predicted. The NRC staff reviewed DOE's comparison of the range of pH, chloride-to-nitrate ratio, and ionic strength values the near-field chemistry model predicted with the calculated values of pH, chloride-to-nitrate ratio, and ionic strength for the 34 pore water sample compositions included in its abstraction. The comparison demonstrated that the range of compositions the near-field chemistry model predicted for 40,000 years and beyond, when the repository environment is expected to return to conditions prior to waste emplacement, is not significantly different than the range of the compositions of the 34 starting pore water samples DOE included in its near-field chemistry model. The NRC staff also compared the range of compositions the near-field chemistry model predicted against the 56 samples DOE screened out from its near-field chemistry model and also against the 33 pore water compositions that NRC staff selected to represent the 156 USGS Yucca Mountain unsaturated zone pore water samples. The results of this comparison demonstrate that the range of seepage water pH, chloride-to-nitrate ratio, and ionic strength values that the DOE near-field chemistry model predicted encompasses most (greater than 92 percent) of the pore water pH, chloride-to-nitrate ratio, and ionic strength values represented by the 34 screened in, 56 screened out, and 33 representative USGS samples. On the basis of this analysis, the NRC staff concludes that the range of seepage water pH, chloride-to-nitrate ratio, and ionic strength values that DOE's near-field chemistry model predicted under conditions prior to waste emplacement is adequate for its intended use in the TSPA analysis.

Abstraction and Integration

The NRC staff reviewed the information and description provided in SAR Section 2.3.5.3.4 and references therein to evaluate whether the model integration and abstraction into the TSPA model of the chemistry of water entering the drifts are adequate. None of the results from the abstraction of the chemistry of water entering the drifts are directly used in the TSPA model. The near-field chemistry model provides inputs to the EBS physical and chemical environment abstraction model. Results from the EBS physical and chemical environment abstraction model are abstracted into the TSPA model. Both the EBS physical and chemical environment abstraction model and the abstraction of results into the TSPA model are evaluated in SER Section 2.2.1.3.3.3.2.

The near-field chemistry model calculates the thermal field using the same modeling approach and assumptions as other unsaturated zone thermal-hydrologic models. The NRC staff has evaluated the information provided in SAR Section 2.3.5.3.3.2.6 and compared it with the multiscale thermal-hydrologic modeling (SAR Section 2.3.5.4.1) and the in-drift condensation modeling (SAR Section 2.3.5.4.2). The NRC staff concludes that the modeling approach DOE used in the SAR is consistent in these related abstractions, including the assumptions and parameters used, and is therefore acceptable.

Summary

Because of the limited potential for the chemistry of water entering drifts to affect drip shield and waste package performance significantly and the extent of DOE's corrosion testing programs, the NRC staff's review focused on the fundamental aspects of the near-field chemistry abstraction of water entering the drifts. The NRC staff reviewed the description of the near-field environment, the assumptions incorporated in this near-field chemistry model abstraction, the range of initial pore water compositions, the range of predicted seepage water compositions, and integration with other model abstractions. For the period beyond 40,000 years after repository closure, the repository environment is expected to return to conditions prior to waste emplacement. The NRC staff concludes that the applicant provided adequate information in the license application consistent with acceptance criteria in YMRP Section 2.2.1.3.3.3 and therefore satisfies the applicable regulatory requirements in 10 CFR 63.114. Also, NRC staff concludes that DOE's treatment of FEPs during the initial 10,000 years following permanent closure in this near-field chemistry abstraction continues unchanged through the period of geologic stability (defined as 1 million years in 10 CFR 63.302). Therefore, the applicant's treatment of FEPs through the period of geologic stability is consistent with 10 CFR 63.114(b) and 63.342(c), and is acceptable.

In particular, considering the risk to performance, the applicant adequately described (i) the range of input data to the near-field chemistry model, (ii) important processes such as feldspar dissolution, and (iii) integration and consistency with other related model abstractions. The NRC staff further concludes that the range of ambient temperature information passed to other abstractions, as well as DOE's corrosion testing program [e.g., pH, ionic strength (I) and concentrations of chloride (Cl) and nitrate (NO_3), and gas phase partial pressures of carbon dioxide (pCO_2)], is adequate.

2.2.1.3.3.3.2 Chemistry of Water in the Drifts

The abstraction for the chemistry of water in the drifts receives information on input gas and water compositions from the near-field chemistry model. The main purpose of DOE's in-drift water chemistry abstraction is to predict the range of chemical compositions for seepage on the waste package or in the invert for a given set of temperature, relative humidity, and pCO_2 conditions. This abstraction is implemented in the TSPA model in the form of lookup tables. These lookup tables enable the TSPA model to provide the parameters (and their uncertainties) needed to represent the chemical environment for the corrosion of waste package surfaces and for radionuclide transport in the invert.

The in-drift chemistry abstraction is not used to provide input to drip shield corrosion modeling. Instead, DOE modeled general corrosion of the titanium drip shield using two corrosion rate values based on weight-loss data determined from long-term corrosion tests. The NRC staff's evaluation of DOE's drip shield general corrosion model abstraction is discussed in SER Section 2.2.1.3.1.3.1.1. In addition, SER Section 2.2.1.3.6 reviews the abstraction for thermal-hydrologic processes affecting seepage rates, SER Section 2.2.1.3.1 reviews the abstraction for corrosion processes affecting the drip shield and waste packages, and SER Section 2.2.1.3.4 reviews the abstraction for the quantity and chemistry of water inside breached waste packages and the invert.

According to the applicant's SAR Table 2.1-3 and DOE (2009an), the chemistry of water in the drifts is important to the performance capability of the emplacement drift (a barrier in the EBS). For example, the TSPA uses lookup tables with a distribution of values for water chemistry. Incorrect or inadequate representation of the chemistry of the waters in the drift may influence the calculation of waste package corrosion and radionuclide transport in the invert and may thus lead to incorrect dose estimates. Key risk information considered in determining the adequacy of this abstraction includes (i) input data to the in-drift precipitates/salts model; (ii) consideration of processes that strongly affect water chemistry, such as evaporation, condensation, and salt precipitation; and (iii) uncertainty propagation through the model abstraction.

The NRC staff considers the following specific observations from DOE's performance assessment to be important in evaluating this abstraction of chemistry of water in the drifts:

- During much of the thermal period, to about 12,000 years after repository closure, the drip shield is expected to prevent seepage water from contacting the waste package surface and greatly reduce the possibility of localized corrosion of the waste package.

- With no seepage water contacting the waste package within 12,000 years after repository closure and relatively limited expected waste package corrosion, only diffusive, not advective, release of radionuclides from the waste package is considered possible by DOE.

- For the period following 12,000 years after repository closure, DOE calculated that there is a low probability for the repository environment (i.e., pH and chemical composition of in-drift waters) to support localized corrosion of the waste package even if the drip shield fails and allows seepage water to contact the waste package.

- After 40,000 years, the temperature and relative humidity in the drifts is expected to have returned to conditions consistent with those prior to waste emplacement. Similarly, seepage water chemistry is also expected to have returned to compositions consistent with those prior to waste emplacement with dilute concentrations of dissolved components. As a result of the low temperature, high relative humidity, and dilute seepage water chemistry, localized corrosion is not an important contributor to waste package degradation after 40,000 years.

Through its FEP screening process, DOE excluded several processes that might have been important to the chemical evolution of water in the drifts, such as corrosion due to deliquescence (FEPs 2.1.09.28.0A and 2.1.09.28.0B). The adequacy of the rationale for excluding specific FEPs from the in-drift water chemistry abstraction is reviewed in SER Section 2.2.1.2.1.3.2.

The NRC staff reviewed the applicant's model for chemistry of water in the drifts contained in SAR Section 2.3.5.5, references therein, and the applicant's responses to RAIs (DOE, 2009an, 2009cv, 2009cw). The NRC staff's review focused on evaluating whether (i) the conceptual model includes important processes, (ii) data and model justification are adequate, (iii) data and model uncertainty are adequate, and (iv) model abstraction support is adequate.

Conceptual Model and Important Processes

The NRC staff reviewed the information and system description that DOE provided in SAR Section 2.3.5.5, references therein, and applicant responses to RAIs (DOE, 2009cv, 2009cw) to evaluate whether the conceptual model used to characterize the in-drift chemical environment is adequate. The NRC staff's evaluation focused on the (i) conceptual model and (ii) important processes.

SAR Table 2.2-1 contains the features, events, and processes (FEPs) that DOE identifies as potentially relevant to the chemistry of water in the drifts. DOE evaluated and included the following FEPs in the in-drift water chemistry abstraction: (i) Chemical characteristics of water in drifts (FEP 2.1.09.01.0A); (ii) Reduction-oxidation potential in drifts (FEP 2.1.09.06.0B); (iii) Reaction kinetics in drifts (FEP 2.1.09.07.0B); and (iv) Thermal effects on chemistry and microbial activity in the engineered barrier system (FEP 2.1.11.08.0A). DOE evaluated and excluded, on the basis of low probability or low consequence, the following FEPs in this abstraction:

- Chemical effects of excavation and construction in the engineered barrier system (FEP 1.1.02.00.0A)
- Undesirable materials left (in the repository) (FEP 1.1.02.03.0A)
- Seismic-induced drift collapse alters in-drift chemistry (FEP 1.2.03.02.0E)
- Chemical properties and evolution of backfill (FEP 2.1.04.02.0A)
- Erosion or dissolution of backfill (FEP 2.1.04.03.0A)
- Chemical effects of rock reinforcement and cementitious materials in the engineered barrier system (FEP 2.1.06.01.0A)
- Chemical degradation of invert (FEP 2.1.06.05.0D)
- Chemical effects at engineered barrier system component interfaces (FEP 2.1.06.07.0A)
- Gas explosions in the engineered barrier system (FEP 2.1.12.08.0A)
- Radiolysis (FEP 2.1.13.01.0A)
- Complexation in the engineered barrier system (FEP 2.1.09.13.0A)
- Microbial activity in the engineered barrier system (FEP 2.1.10.01.0A)
- Gas generation (CO_2, CH_4, H_2S) from microbial activity (FEP 2.1.12.04.0A)
- Radiological mutation of microbes (FEP 2.1.13.03.0A)
- Chemical effects of excavation and construction in the near-field (FEP 2.2.01.01.0B)

The adequacy of the rationale for excluding these specific FEPs from the in-drift water chemistry abstraction is reviewed in SER Section 2.2.1.2.1.3.2. DOE's pre-10,000-year treatment of FEPs in this abstraction continues unchanged beyond the 10,000-year postdisposal period through the period of geologic stability.

DOE explained its conceptual model for in-drift chemistry as follows: early in the postclosure period, drift wall temperatures higher than the boiling point of water will prevent seepage from occurring. After the drift wall temperatures fall below the boiling point of water and the rewetting process begins, seepage may occur, as local hydrologic conditions allow. Because waste package surface temperatures will still be elevated, seepage water falling on the drip shield, and on the waste package in the event of drip shield failure, will evaporate and concentrate. As waste package temperatures continue to decrease, relative humidity will increase to the point that wet conditions persist. Over time, further increases in relative humidity will suppress evaporation and result in progressively more dilute aqueous solutions on the waste package surface or in the invert.

DOE's model considers how the chemistry of seepage water will evolve after it enters the repository drifts. In its conceptual model, DOE considered the effects of seepage water evaporation, condensation, gas–water interaction, precipitation and dissolution of salts, and salt separation. DOE's conceptual model describes in general terms how each of these processes influences the chemistry of in-drift water. For example, seepage evaporation will cause the most soluble components to concentrate in the aqueous phase and minerals to precipitate. With precipitation, the relative concentrations of components remaining in solution will change. Salt separation may occur when seepage water flows downward over the drip shield or waste package surface while evaporation is occurring. During this process, spatial separation of chemical components could occur, transporting the more soluble aqueous components (e.g., NO_3^-) and leaving behind as precipitates the less soluble constituents (e.g., Cl^-, as NaCl precipitate). Condensation of water, which would dilute the aqueous phase concentration, could occur when the in-drift relative humidity is high enough. To model the in-drift water chemistry, DOE used the in-drift precipitates/salts model, which is a process-level geochemical model that accounts for the effects of in-drift processes. The in-drift precipitates/salts model was implemented using the EQ3/6 geochemical code and a Pitzer thermodynamic database that was developed for use in the in-drift precipitates/salts model. The NRC staff concludes that the DOE model appropriately considered the most risk- and performance-significant processes based on the NRC staff's independent understanding of processes affecting the evolution of in-drift water chemistry (e.g., Murphy, 1994aa; Browning, et al., 2004aa; Yang, et al., 2011aa).

In natural systems, the chemical evolution of evaporating water generally is controlled by the high solubility of chloride and nitrate salt minerals relative to the moderate solubility of calcium sulfate and the low solubility of calcium carbonate minerals—a mechanism referred to as a chemical divide (Hardie and Eugster, 1970aa). Thus, evaporation of initially dilute natural waters at the Earth's surface, such as in saline lakes, typically leads to the formation of one of three brine types, depending on the initial composition of the system: calcium chloride brine, alkaline carbonate brine, or high sulfate brine. DOE concluded that the same brine types could occur within the drifts because in-drift brines would be produced by processes similar to those that occur at the Earth's surface. The specific processes considered by DOE are discussed below.

DOE used several assumptions in its abstraction of in-drift water chemistry. For example, all aqueous and gas constituents are assumed to achieve and maintain local equilibrium. The NRC staff considers this assumption appropriate because the chemical reactions considered in the abstraction are fast relative to the modeling timeframe. Also, the seepage waters on the waste package surface are assumed to reach equilibrium with the relative humidity on the waste package surface. The NRC staff concludes that this assumption is appropriately conservative because the temperature would be highest and the relative humidity would be lowest at the waste package surface, which would maximize seepage water evaporation and result in the highest brine concentration. In addition, DOE assumed that an aqueous solution is present for all temperature and relative humidity conditions once seepage onto a waste package occurs. The NRC staff considers this an appropriate and conservative assumption given that an aqueous phase is required for corrosion of engineered barriers. This assumption is also supported by laboratory experiments in which dryout was not observed at temperatures up to 190 °C [374 °F], particularly for brines with Na–K–Cl–NO_3 compositions (Rard, et al., 2006aa). DOE also assumed the chemical compositions of drift wall condensation and of condensation that penetrates a failed drip shield to be the same as seepage composition and to be benign. The NRC staff concludes that this assumption is appropriate because water that condenses on the drift wall or other surfaces likely will be dilute, and any increase in concentration due to chemical interaction with drift wall surfaces, in-drift gases, and dusts deposited on waste package and drip shield surfaces will be small relative to increases in concentration due to evaporation.

In the salt separation abstraction, DOE assumed that the solution that forms during the salt separation process is chloride rich. The NRC staff considers this an appropriate and conservative assumption because a chloride-rich solution is more corrosive to the waste package material than a chloride plus nitrate solution.

With the in-drift precipitates/salts model, DOE conducted a series of seepage evaporation/dilution analyses at discrete temperature, relative humidity, and pCO_2 values. The analyses used the water compositions derived from the near-field chemistry model as input. The analyses used 11 water–rock interaction parameters for each of the 4 representative pore water compositions, for a total of 44 water compositions. DOE selected 3 temperatures for the analyses—30, 70, and 100 °C [86, 158, and 212 °F]—to cover the temperature range of interest while minimizing interpolation errors. At each temperature, the 44 waters were evaporated at 3 pCO_2 values: 10^{-2}, 10^{-3}, and 10^{-4} bar; these pCO_2 values were selected on the basis of the results from the near-field chemistry model. In a second set of EQ3/6 simulations, the waters were diluted by a factor of 100. The oxygen partial pressure (pO_2) in all the simulations was set equal to the atmospheric value to represent oxidizing conditions in the drift. The NRC staff considers this assumption appropriate given that the ranges for expected conditions are consistent with those predicted by DOE's performance assessment model and the NRC staff's independent modeling (e.g., Murphy, 1994aa; Browning, et al., 2004aa). The NRC staff concludes that the pCO_2 values and pore water composition selection are appropriate, as discussed in SER Section 2.2.1.3.3.3.1.

The seepage evaporation/dilution analysis results form the basis for DOE's in-drift water chemistry abstraction, which was implemented in the TSPA code in the form of 396 lookup tables (simulation results for 4 representative pore waters × 11 water–rock interaction parameter values × 3 temperatures × 3 pCO_2 values). The lookup tables represented the range of chemical compositions that potentially could be generated by evaporation or dilution of drift wall seepage or condensation, or by waters imbibed into the invert. The lookup tables enabled the TSPA code to provide the following parameters and their uncertainties for a given set of temperature, relative humidity, and pCO_2 conditions: pH, ionic strength, Cl^- and NO_3^- concentrations, and the NO_3^-/Cl^-ratio. These parameters are used in the TSPA model to represent the chemical environment for the corrosion of waste package surfaces and for radionuclide transport in the invert.

To determine which set of lookup tables is used for the in-drift water composition (SAR Section 2.3.5.5.4.3), the TSPA model used the following four inputs: the starting water identity (Groups 1, 2, 3, or 4); the water–rock interaction parameter derived from the near-field chemistry model; the pCO_2, which was derived from the near-field chemistry model; and the waste package surface temperature derived from the multiscale thermal-hydrologic model. The specific water composition in the table was selected on the basis of the relative humidity at the waste package surface, which in turn was derived from the multiscale thermal-hydrologic model. For water–rock interaction parameters, temperatures, and pCO_2 values that fell between the values listed in the lookup tables, DOE interpolated values from adjacent tables.

In SAR Section 2.3.5.5.4.3, DOE indicated that brine compositions resulting from seepage evaporation are most sensitive to the degree of water–rock interaction and to pCO_2; temperature had a comparatively smaller effect. The degree of water–rock interaction (the amount of feldspar dissolved) had the greatest effect on pH. With increasing amounts of feldspar dissolved, all the waters DOE considered in the analysis evolved into carbonate-type brines because feldspar dissolution and secondary mineral precipitation consume calcium and magnesium ions and raise the pH and bicarbonate concentration. DOE observed that the relationship between the degree of water–rock interaction and brine type is important because carbonate-type brines typically have chloride and nitrate concentrations that are not conducive to localized corrosion of the Alloy 22 waste package outer barrier material. DOE concluded that corrosive calcium and magnesium-chloride brines are not expected to form in the potential repository.

The NRC staff compared DOE's conceptual model with the NRC staff's understanding of the Yucca Mountain natural system and independent analysis of in-drift processes (Browning, et al., 2003aa, 2004aa). On the basis of this understanding and risk-informed independent analysis, the NRC staff concludes that DOE incorporated the appropriate physical processes and couplings in its system description for the in-drift water chemistry abstraction. The included FEPs and associated physical processes and couplings in the abstracted model are consistent with independent geochemical modeling and consider relevant processes and parameters.

DOE's model calculated that for time periods beyond 12,000 years after repository closure, there is a low probability for the repository environment (i.e., temperature, pH, and chemical composition of in-drift waters) to support localized corrosion of the waste package even if the drip shield fails and allows seepage water to contact the waste package. DOE's model calculated that the pH and nitrate-to-chloride ion ratio in in-drift water will generally be too high to initiate localized corrosion in this time period. The NRC staff conducted independent analysis of in-drift water that may contact the waste package under the temperature and relative humidity conditions that may exist in the drift at 12,000 years after repository closure or later (Pabalan, 2010aa). The NRC analysis used StreamAnalyzer 2.0 and OLIAnalyzer 3.0 (Gerbino, 2006aa) thermodynamic software to simulate the evaporative evolution of seepage waters, using as input the compositions of USGS pore water samples discussed in SER Section 2.2.1.3.3.3.1. The results of the NRC staff's analysis illustrate that brines resulting from evaporation of initially dilute pore waters do not support localized corrosion of the waste package at in-drift temperature and relative humidity conditions 12,000 years after repository closure or later. On the basis of this independent analysis, the NRC staff concludes that DOE's results that indicate a low probability for localized corrosion initiation beyond 12,000 years after repository closure are acceptable, because the repository environment (i.e., temperature, pH, and chemical composition of seepage waters) would not support the initiation of localized corrosion (SER Section 2.2.1.3.1.3.2.2).

Data and Model Justification

The NRC staff reviewed information that DOE provided in SAR Section 2.3.5.5, references therein, and applicant responses to RAIs (DOE, 2009cw) to evaluate whether the data and model justification used to characterize the in-drift chemical environment are adequate.

This evaluation focused on (i) adequacy of the thermodynamic database and (ii) support of the model by laboratory experiments and other corroborating sources.

As indicated in the preceding section, the in-drift precipitates/salts model was implemented using the EQ3/6 geochemical code and a Pitzer thermodynamic database. The parameters in the database were obtained from a variety of thermodynamic data and solubility measurements reported in the scientific literature and synthesized into an internally consistent data set. DOE evaluated the principal temperature-dependent Pitzer parameters in the synthesized data set for their ability to reproduce the original source information. The NRC staff evaluated the comparisons of measured data and model results in SNL (2007ao, Appendix I) and concludes that the in-drift precipitates/salts model adequately reproduces the data used to construct the thermodynamic database.

DOE also used several chemical data sets to support its parameter values and to build confidence in the in-drift precipitates/salts model. The data sets included (i) laboratory experiments designed to investigate the effects of evaporation on the chemical evolution of water compositions and environmental conditions relevant to the potential repository; (ii) a natural analog for evaporative concentration of seawater at the Morton Bahamas solar salt production facility on Great Inagua Island in the Bahamas; and (iii) compilations of solubility measurements in single, binary, and ternary salt systems from handbooks and published sources. DOE compared these data with results from the in-drift precipitates/salts model in SAR Section 2.3.5.5.3.3 and referenced documents. The NRC staff evaluated these comparisons of measured data and model results and verified that, in general, the parameters used in the Pitzer thermodynamic database are adequately supported by laboratory experiments, natural analog research, process-level modeling, and scientific literature (e.g., Linke,1958aa,1965aa; McCaffrey, et al.,1987aa; Wolf, et al.,1989aa; Rosenberg, et al., 1999aa, 1999ab; Alai, et al., 2005aa). However, the NRC staff notes that not all of these model simulation results, such as those for individual and multisalt solutions, are consistent with the experimental data presented by the applicant. For example, although DOE reported that in-drift precipitates/salts model results for single, binary, and ternary salt saturation points and deliquescence relative humidities are generally within 20 percent of literature values, several in-drift precipitates/salts model results differ by 20 percent or more compared with the experimental data. Although DOE asserted that the comparisons between literature and calculated values are favorable for individual and multisalt systems, figures provided for the single and multisalt systems show a mismatch between some calculated and experimental values, including a lack of similar trending between data sets. Furthermore, figures from SNL (2007ao) not included in the SAR show greater uncertainties than those included in the SAR. These figures show that as the complexity of the system increases, the dissimilarity between the in-drift precipitates/salts model-calculated values and literature data appears to increase. Nevertheless, the NRC staff concludes that model justification is adequate for this review on the basis of propagation of uncertainty throughout the in-drift water chemistry abstraction (see next section), and because localized corrosion is not considered a contributor to waste package degradation after 12,000 years, because the in-drift water chemistry will not support localized corrosion of the waste package.

The NRC staff also evaluated the sufficiency of DOE’s baseline data to justify the in-drift water chemistry abstraction. The NRC staff concludes that the thermal, hydrological, and geochemical values used by DOE are adequately justified. For example, the abstraction uses temperature and pCO_2 values that are technically defensible and reasonably account for uncertainties and variability in those parameters. Based on the discussion set forth in this subsection, the NRC staff also concludes that DOE adequately described how measured pore water chemistry was used, interpreted, and synthesized into the in-drift water chemistry abstraction. However, the NRC staff notes that DOE may not have considered the full range of natural system characteristics when DOE established the initial and boundary conditions for the in-drift water chemistry model. In particular, the four starting water compositions DOE used may not entirely represent the uncertainty in Yucca Mountain pore water composition. Nevertheless, as discussed in SER Section 2.2.3.3.3.1, and because of adequate consideration of uncertainty, NRC staff concludes, on the basis of its understanding of near-field processes and from the results of its independent analyses, that incorporating a wider range of starting water compositions in the in-drift water chemistry abstraction will not significantly affect repository performance. Thus, the NRC staff concludes DOE's baseline data are sufficient to justify the in-drift water chemistry abstraction given their intended use in DOE's TSPA model.

Data and Model Uncertainty

The NRC staff reviewed information that DOE provided in SAR Section 2.3.5.5, references therein, and applicant responses to RAIs (DOE, 2009cw) to evaluate whether the data and model uncertainties used to characterize the in-drift chemical environment are adequate. This evaluation focused on (i) inclusion of uncertainty propagation in input data and (ii) uncertainty propagation throughout the in-drift precipitates/salts model.

In SAR Section 2.3.5.5, DOE identified that uncertainties in the in-drift precipitates/salts model result in uncertainties in the TSPA code values of pH, ionic strength, concentrations of Cl^- and NO_3^-, NO_3^-/Cl^- ratio, and deliquescence relative humidity. DOE evaluated these uncertainties using model–data comparisons. Uncertainty in pH was given particular consideration due to variances in methods of measuring pH (whether true activity of the hydrogen ion is taken into account) and because there is significant experimental error when measuring the pH of concentrated brines. For solutions with water activities approximately 0.75 or higher, pH uncertainty was determined indirectly through the uncertainty in total inorganic carbon concentration, which reasonably reflects uncertainty in pH for the near-neutral range. This carbon concentration was evaluated using data from evaporation experiments and on calcite or CO_2 solubility. For more concentrated solutions with lower water activities, pH uncertainty was estimated on the basis of comparisons of calculated and measured pH in concentrated solutions. DOE evaluated the uncertainty in ionic strength by comparing values calculated using the in-drift precipitates/salts model with those derived from evaporation experiments. Uncertainties in the Cl^- and NO_3^- concentrations and in the NO_3^-/Cl^- ratio were evaluated by comparing in-drift precipitates/salts model results with data from evaporation experiments and solubility measurements. DOE assessed the uncertainty in deliquescence relative humidity by comparing in-drift precipitates/salts model results with deliquescence relative humidity values reported in the literature that DOE referenced.

The NRC staff evaluated DOE's characterization and propagation of uncertainty in the in-drift water chemistry abstraction. The NRC staff concludes that uncertainties in the DOE conceptual model are adequately considered because they are consistent with available laboratory experiments and chemical data in published literature (e.g., McCaffrey, et al., 1987aa; Alai, et al., 2005aa). Specifically, uncertainties in pH, ionic strength, deliquescence relative humidity, and ionic concentrations derived from the in-drift precipitates/salts model are reasonably supported by laboratory evaporation experiments, solubility data, and deliquescence relative humidity data.

Model Abstraction Support

The NRC staff reviewed information that the DOE provided in SAR Section 2.3.5.5, references therein, and applicant responses to RAIs (DOE, 2009cv) to evaluate whether the support for the model abstraction used to characterize the in-drift chemical environment is adequate. This evaluation focused on (i) consistency with process-level modeling and (ii) consistency of output data ranges with independent data.

In SAR Section 2.3.5.5.4.2.2, DOE described the approach it used to build confidence in the in-drift water chemistry model abstraction. For example, DOE abstracted the range of in-drift water chemistry in the form of lookup tables at discrete temperature and $p\mathrm{CO_2}$ values. DOE supported the abstraction approach by demonstrating that the results derived by interpolation between lookup tables are within the stated model uncertainties for in-drift precipitates/salts model simulations at the actual temperature and $p\mathrm{CO_2}$ conditions tested. DOE provided additional support to its in-drift water chemistry model abstraction in DOE (2009cv). The NRC staff evaluated the DOE information and concludes that the TSPA abstraction of in-drift water chemistry is consistent with process-level modeling. The NRC staff verified that the in-drift water chemistry abstraction is based on the same assumptions and approximations demonstrated by DOE to be appropriate for process-level models of closely analogous natural or experimental systems.

The NRC staff also verified that DOE used accepted and well-documented procedures to construct and test the numerical model that simulates the evolution of in-drift water chemistry. The thermodynamic database used in the in-drift precipitates/salts model was developed by DOE from a variety of literature sources (see SAR Section 2.3.5.5.4.2.2) and synthesized into an internally consistent data set, which was evaluated by the NRC staff for its ability to reproduce the original source information. Therefore, the NRC staff concludes that the in-drift precipitates/salts model was appropriately supported by comparison with laboratory and natural analog information. Further, the NRC staff compared the ranges of in-drift water chemistry (e.g., pH, Cl^- and NO_3^- concentrations, ionic strength) DOE tabulated in lookup tables with the ranges derived from an alternative modeling approach (Leslie, et al., 2007aa). The NRC modeling approach used thermodynamic calculations to simulate the evaporation of seepage waters. The calculations were implemented using StreamAnalyzer 2.0 and OLIAnalyzer 3.0 (Gerbino, 2006aa) and used as input chemical compositions of Yucca Mountain unsaturated zone pore water samples the USGS reported (Yang, et al., 2003aa, 1998aa, 1996aa). The ranges in pH, Cl^- and NO_3^- concentrations, and NO_3^-/Cl^- ratio derived from the NRC approach are consistent with those derived from DOE's in-drift water chemistry abstraction. Thus, the NRC staff concludes that DOE's abstraction output is supported by objective comparisons.

Summary

The NRC staff reviewed the in-drift chemistry abstraction (including the description of the in-drift environment), the assumptions incorporated in the in-drift precipitates/salts model abstraction, the Pitzer database for the in-drift precipitates/salts model, supporting data and experiments, and uncertainty propagation through the in-drift precipitates/salts model. The NRC staff concludes that the applicant provided sufficient and adequate information in the license application consistent with acceptance criteria in YMRP Section 2.2.1.3.3.3 and therefore satisfies the applicable regulatory requirements in 10 CFR 63.114. Also, NRC staff concludes that DOE's treatment of FEPs during the initial 10,000 years following permanent closure in this abstraction continues unchanged through the period of geologic stability (defined as 1 million years in 10 CFR 63.302). Therefore the applicant's treatment of FEPs through the period of geologic stability is consistent with 10 CFR 63.114(b) and 63.342(c), and is acceptable.

In particular, on the basis of risk insight information, the applicant adequately and sufficiently (i) described input data to the in-drift precipitates/salts model; (ii) considered important processes such as evaporation, condensation, and salt precipitation; and (iii) propagated uncertainty through the model abstraction. The NRC staff concludes that, for temperatures consistent with environmental conditions prior to waste emplacement, the range of chemistry tabulated in the TSPA lookup tables (i.e., relative humidity; pCO_2 conditions of pH, ionic strength, and Cl^- and NO_3^- concentrations; and the NO_3^-/Cl^- ratio) adequately represents the potential chemistry of water contacting the surface of waste packages and radionuclide transport in the invert. The NRC staff also concludes, on the basis of its independent analysis, that in-drift water chemistry is unlikely to initiate waste package localized corrosion in the time following 12,000 years after repository closure.

2.2.1.3.3.3.3 Quantity of Water in Contact With the Engineered Barrier System

Abstracting the quantity of water in contact with the EBS (i) determines seepage flux rates (the amount of water flowing through a pathway per unit time) through and around breached or intact waste packages and the drip shield and (ii) provides an estimate for partitioning of radionuclides exiting the EBS between unsaturated zone fractures and in the rock matrix. The EBS flow abstraction receives seepage flux approaching the drift wall from the drift seepage abstraction (BSC, 2004aa), condensation on the drift walls from the In-drift Natural Convection and Condensation model abstraction (BSC, 2004aw), capillary wicking (imbibition flux) into the invert from the Multiscale Thermal-Hydrologic Model abstraction (BSC, 2005ah), and the size and evaluation of corrosion openings on the waste packages from the WAPDEG corrosion model abstraction (BSC, 2004bs). SER Section 2.2.1.3.6 reviews the abstraction that addresses thermal-hydrologic processes affecting seepage, SER Section 2.2.1.3.1 reviews the abstraction that addresses corrosion processes affecting the drip shield and waste packages, and SER Section 2.2.1.3.4 reviews the abstraction that addresses the quantity and chemistry of water inside breached waste packages and the invert.

The NRC staff review of this abstraction is important because missing, discontinuous, or misrepresented flow paths in the EBS and inadequate representation of water flow through and capillary diversions around the breached drift and waste packages, may result in incorrect dose estimates. Key risk information used to determine the adequacy of this abstraction includes (i) seepage flux rate at the drift wall, (ii) timing of the failure of the drip shield, and (iii) fraction of the breached patch area on waste packages. As discussed in the previous paragraph, this key risk information, which affects the distribution and flux rate throughout the EBS, is computed outside of this flow abstraction and then passed into this flow abstraction. Furthermore, the relative magnitude of the imbibition flux from the host rock matrix into the invert, which is computed outside the EBS flow abstraction, and the flux into the invert (that, in turn influences the fraction of radionuclides released from the EBS into unsaturated fractures and the rock matrix), which is computed by the EBS flow abstraction, comprise the key risk information propagated to the unsaturated zone transport abstraction in the TSPA model.

The NRC staff reviewed the applicant's model for EBS flow in SAR Section 2.3.7.12, references therein, and applicant responses to RAIs (DOE, 2009av, 2009an, 2009ay, 2009am). The NRC staff's review focused on evaluating whether (i) the conceptual model for flow paths and flux splitting throughout the intact and failed EBS components under both nominal and disruptive events is adequate; (ii) model integration of the EBS flow abstraction with other abstractions in the TSPA model, as well as information exchanges between the EBS flow abstraction and other abstractions, is transparent and adequately described; (iii) model parameters are adequately supported by available experimental data, and data uncertainties are adequately propagated within the EBS flow abstraction and into other abstractions in the TSPA code; and (iv) model uncertainties are adequately analyzed through alternative model abstractions.

Conceptual Model for the Engineered Barrier System (EBS) Flow Paths and Flux Splitting

The NRC staff reviewed the information and description provided in SAR Sections 2.3.7.12 and 2.2.1.2.1, references therein, and applicant responses to RAIs (DOE, 2009ab, 2009av, 2009an) to evaluate whether the model integration and conceptual model of the quantity of water in the EBS are adequate. This evaluation focused on (i) the continuity and integration of flow paths and (ii) impacts of intact and breached EBS components on the EBS flow in the nominal case and disruptive events.

SAR Table 2.2-1 contains the FEPs that DOE believes are potentially relevant to the quantity of water in contact with the EBS. The applicant evaluated and included the following FEPs in this abstraction: (i) Capillary effects (wicking) in engineered barrier system (FEP 2.1.08.06.0A) and (ii) Unsaturated flow in the engineered barrier system (FEP 2.1.08.07.0A). The applicant evaluated and excluded, on the basis of low probability or low consequence, the following FEPs from this abstraction: (i) Advection of liquids and solids through cracks in the waste package (FEP 2.1.03.10.0A), (ii) Advection of liquids and solids through cracks in the drip shield (FEP 2.1.03.10.0B), (iii) Saturated flow in the engineered barrier system (FEP 2.1.08.09.0A), and (iv) Condensation on underside of drip shield (FEP 2.1.08.14.0A). Note that the adequacy of the rationale for excluding these specific FEPs from this abstraction is reviewed in SER Section 2.2.1.2.1.3.2. Furthermore, DOE’s treatment of FEPs during the initial 10,000 years following permanent closure in this abstraction continues unchanged beyond initial 10,000 years following permanent closure through the period of geologic stability.

The applicant’s EBS flow abstraction is based on a mass-conserving, flux-splitting algorithm involving eight potential unsaturated flow pathways in the EBS. The flow pathways and fluxes along these pathways are labeled F1 through F8 in SAR Figure 2.3.7-8 (DOE, 2009av). The upper wall of the drift forms the top boundary (along the F1 flow path), and the bottom part of the invert forms the lower boundary (along the F8 flow path). The EBS flow abstraction calculates time-variant flux rates along unsaturated flow pathways across the EBS for the nominal case and disruptive events.

The applicant described the flow pathways and flux rates (F1–F8) in the EBS flow abstraction as follows (Figure 6-1): the F1 flow path accounts for the total dripping flux from a drift wall. The total dripping flux is the sum of the seepage flux from the drift wall above and the condensed water on drift walls (SER Section 2.2.1.3.6.3.4); these are direct inputs to the EBS flow abstraction. The F2 flow path accounts for the flux through partially failed patches of the drip shield formed by general corrosion. Localized corrosion of the drip shield is excluded from the performance assessment, as discussed in SER Section 2.2.1.2.1.3. The F3 flow path accounts for the diversion of flux around the drip shield (computed as F3 = F1 – F2), which will drain directly into the invert. Although the diversion of flux around the drip shield is included in the construction of the EBS flow model, the applicant did not implement the flux-splitting algorithm for drip shields in TSPA simulations, because the drip shields were modeled to be either all intact or failed, as described in SNL (2007aj, Section 6.1.1). The F4 flow path accounts for the flux through patches, formed as a result of general corrosion of the outer barrier of the waste packages. Localized corrosion of the outer barrier of waste packages is not considered

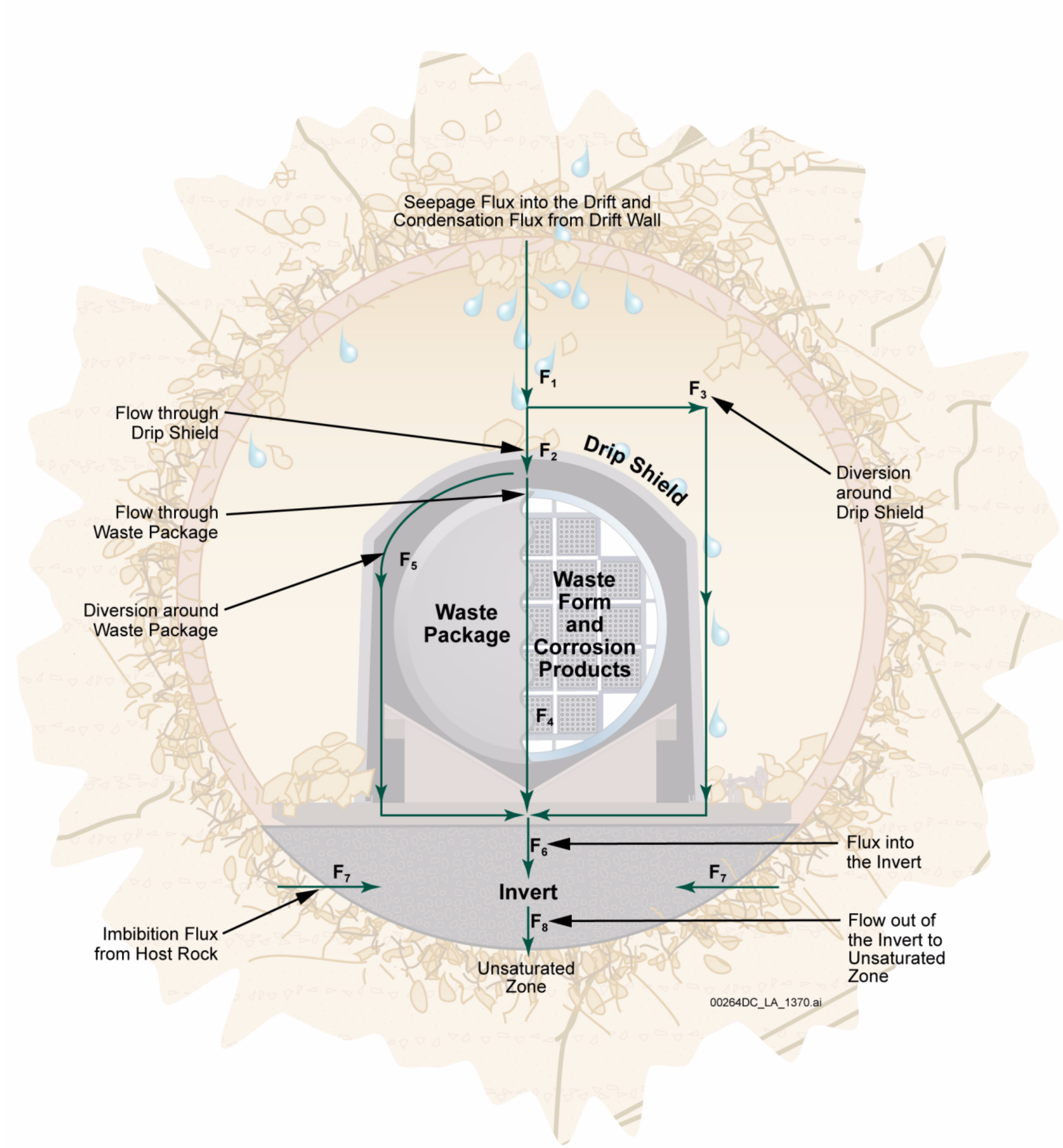

Figure 6-1. Potential Flow Pathways in the Engineered Barrier System (SAR Figure 2.3.7-8; DOE, 2008ab)

important by DOE for this abstraction, because the probability of waste package breach by localized corrosion is low in DOE's model (SER Section 2.2.1.3.1.3.1). The F5 flow path accounts for the diversion of flux around waste packages (computed as F5 = F3 – F4), which will drain directly into the invert. The F6 flow path is the total flux entering the invert (computed as F6 = F4 + F3 + F5). The F7 flow path accounts for the imbibition flux from the host rock matrix into the invert and is a direct input to the EBS flow abstraction. The F8 flow path is the total flux from the invert to the unsaturated zone (computed as F8 = F6 + F7). Thus, the magnitude of fluxes in the EBS is determined by the flux rates at the drift wall, flow exchanges between the invert and the surrounding unsaturated fractured domain, and the size of corrosion patches on the drip shield and waste packages, which are externally calculated. The rest of the fluxes in the EBS are computed on the basis of the mass-conserving, flux-splitting algorithm.

In addition to the nominal case, the applicant addressed both the igneous intrusion and the seismic ground motion cases. The NRC staff confirmed that the applicant demonstrated, on the basis of TSPA simulation results, that these two disruptive modeling cases are the most significant contributors to the total dose for the 10,000- and 1-million-year simulations, as shown in SAR Figure 2.4-18 (DOE, 2009av). Because the contribution of the other modeling cases (including drip shield early failure, waste package early failure, seismic fault displacement, and volcanic eruption) to the mean annual dose for the 10,000- and 1-million-year simulations are at least an order of magnitude smaller than the contributions to the mean annual dose by the igneous intrusion and the seismic ground motion modeling cases, the other modeling cases are discussed in SER Section 2.2.1.4.1, but are not discussed in this section. For the igneous intrusion scenario, the drip shield and waste packages entirely lose their integrity instantaneously at the time of the intrusive event, and all seepage water approaching the drift wall flows through the waste package, as described in SNL (2007aj, Section 6.1.1). For the seismic ground motion scenario, after the drip shield fails, the water flow rate through a damaged waste package depends on the expected fraction of the waste package surface (which increases with time) that is breached by corrosion patches, as shown in SNL [2008ag, Figures 8.3-11(a) and 8.3-12(a)] and DOE (2009an).

The NRC staff reviewed the model conceptualization, mass-balance equations, and underlying assumptions of the EBS flow model abstraction and other relevant abstractions with which the EBS flow model abstraction exchanges data and information to determine whether the applicant adequately described the EBS flow model abstraction and the underlying mass-conserving, flux-splitting algorithm. The NRC staff concludes that the applicant adequately identified and described potential flow pathways and flow factors in estimating the quantity of water that could contact the EBS and waste forms on the basis of the mass-conserving, flux-splitting algorithm in the EBS flow abstraction. The NRC staff concludes that the EBS flow abstraction is adequate because (i) the potential unsaturated flow pathways that the applicant identified encompass all potential major flow pathways within the EBS, and between the EBS and the surrounding unsaturated fractured rocks; (ii) the impact of drip shield and waste package failures is adequately addressed in the EBS flow abstraction by explicitly incorporating the failed or intact mode of the drip shield and incorporating externally computed temporal variations in the number of corrosion patches on waste packages; (iii) the impacts of the transient nature of dripping flux and flux exchange between the unsaturated zone and the invert are adequately carried into the EBS flow abstraction through the F1 flow path and the F7 flow path, respectively; and (iv) loss of barrier capability of the EBS following an igneous event and gradual increase of the expected fraction of surface patches on waste packages, which account for increased water fluxes due to seismic ground motion (DOE, 2009an), are physically reasonable, and they are more risk significant as compared with the nominal case.

Model Integration and Information Flow

The NRC staff reviewed the information provided in SAR Section 2.3.7.12, references therein, and applicant responses to RAIs (DOE, 2009am, 2009ay) to evaluate whether the model integration and information exchange with the other abstractions are adequate. This evaluation focused on (i) integration and continuity of flow components in the EBS flow abstraction and (ii) transparency and adequacy of information on input to and output from the EBS flow abstraction.

Input to the EBS flow abstraction includes seepage flux into the drift from the drift seepage abstraction (BSC, 2004aa); condensation on the drift walls from the In-drift Natural Convection and Condensation Model abstraction (BSC, 2004aw), which makes up the F1 flow path; imbibition flux into the invert; the F7 flow path from the Multiscale Thermal-Hydrologic Model abstraction (BSC, 2005aa); and patch size and its evolution from the WAPDEG corrosion model (BSC, 2004bs), which is used for calculating the F4 flow path. The F4 flow path determines the seeping or nonseeping condition in the waste package. Information on the seeping or nonseeping condition is used in the EBS radionuclides and colloid abstraction for determining the rate constant for irreversible attachment of plutonium and americium onto mobile corrosion product colloids (DOE, 2009ay).

DOE's EBS Unsaturated Zone Interface Model (SNL, 2007aj) uses water fluxes along the F6 and F7 flow paths to calculate advective flow rates of radionuclides and colloid suspensions to be used in the unsaturated zone transport abstraction. The F6 flow path determines the water flux rate for radionuclides and colloid suspensions from the invert into the unsaturated zone fractures in a seeping environment. In a nonseeping environment, advective flux from the invert to unsaturated zone fractures is zero unless drift wall condensation is greater than zero. The F7 flow path provides the water flux for advective transport of radionuclides and colloid suspensions from the invert into the unsaturated zone matrix. Imbibition along the F7 flow path could provide a small advective flux into the unsaturated zone matrix in both seeping and nonseeping environments (DOE, 2009am).

To determine whether the applicant adequately described the integration of the EBS flow model abstraction and the information exchange with other abstractions in the TSPA model, the NRC staff reviewed the model conceptualization, the mass-balance equations, and the underlying assumptions of the EBS flow model abstraction and other relevant abstractions with which the EBS flow model abstraction exchanges data and information. The NRC staff concludes that the applicant adequately described model integration and information flow between the EBS flow abstraction and other abstractions in the TSPA model because the applicant adequately (i) identified and integrated input to (the F1 and F7 flow paths) and output from (the F6 and F8 flow paths) the EBS flow abstraction, (ii) computed fluxes internally across breached EBS components (the F2 and F4 flow paths) on the basis of time-variant information from other abstractions, and (iii) computed fluxes internally without requiring direct input from other abstractions. The applicant adequately identified the upstream abstractions (drift seepage abstraction and In-drift Natural Convection and Condensation model), in-parallel abstractions (Multiscale Thermal-Hydrologic Model and WAPDEG corrosion model), and the downstream abstractions (EBS radionuclide transport abstraction), as well as the information exchanged among them through the F1, F2, F4, F6, F7, and F8 flow paths, under both seeping and nonseeping conditions.

Data Support and Uncertainties

The NRC staff reviewed the information provided in SAR Section 2.3.7.12, references therein, and applicant responses to RAIs (DOE, 2009av) to evaluate whether the supporting data and the characterization of uncertainties for the EBS flow abstraction are adequate, including propagation of data uncertainty through the model abstraction. This evaluation focused on (i) the adequacy of experiments to address uncertainties associated with the number of drip points and flow rates, (ii) whether the experimental data are adequately used to bound uncertainties associated with the EBS fluxes, and (iii) whether the uncertainties are adequately propagated within the EBS flow abstractions and into the other abstractions in the TSPA code.

DOE relied on experimental data from the Breached Drip Shield experiments, as outlined in SNL (2007aj, Section 6.3.2.3) and BSC (2003ag), to derive an equation to estimate flux through a breached drip shield (the F2 flow path). The equation is a function of the lateral spread angle of the rivulet flow on the drip shield, the number of corrosion breaches on the drip shield, the length of the breaches, and a sampled uncertainty factor. The uncertainty factor was bounded by DOE using data from the Breached Drip Shield experiments. The applicant adopted the same equation to estimate the flux through a breached waste package (the F4 flow path). The applicant identified that the only difference in implementing the equation for the drip shield and waste forms is that (i) the radius of the curvature of the waste package is less than that of the drip shield and (ii) the nominal patch size is smaller for a waste package than for the drip shield in the WAPDEG corrosion model (BSC, 2004bs). These differences affect the bounds for the uncertainty factor established for the drip shield and the waste package. DOE supported the abstraction for the flow through a breached drip shield and waste packages on the basis of data from the Breached Drip Shield experiments. However, in the TSPA code, the data support is used only for breaches on the waste package because the drip shields are modeled to be either all intact or failed, as detailed in SNL (2007aj, Section 6.1.1).

The applicant bounded the uncertainty factor used, as described in SNL (2007aj, Sections 7.1.1.1 and 7.1.1.2), for the calculations of the F4 flow path on the basis of data from the breached drip shield experiments (BSC, 2003ag). The applicant assumed a uniform distribution for the uncertainty factors because there were insufficient data available to define any other distribution. The NRC staff finds this assumption to be appropriate because uniform distributions are typically used where the uncertainty is difficult to quantify based on the available data. Consequently, the NRC staff concludes that uncertainties associated with the seepage flow arriving at the drift wall are appropriately propagated into the EBS abstraction through the F1 flow path, and uncertainties associated with unsaturated flow in the host rock matrix are propagated into the EBS flow abstraction through the F7 flow path.

The NRC staff reviewed the Breached Drip Shield experiments to determine whether (i) the experiments were adequately designed to support the flux-splitting model conceptualization and (ii) the experimental data were adequately used to bound uncertainties in the EBS flow processes. The NRC staff concludes that the applicant adequately used the data from the Breached Drip Shield experiments, in conjunction with simplified geometrical interpretations, as outlined in SNL (2007aj, Section 6.5.1.1.2), to characterize and bound the uncertainties associated with the F4 flow path (flux through a breached waste package). This conclusion is based on the following: (i) flow on the drip shield occurred as rivulets in the experiments, as expected from a physical standpoint, as a result of drips and splashes from a number of discrete drip points onto the drip shield; (ii) the experiments were run with the flow rate varying over 2 orders of magnitude {0.2–20 m^3/yr [52.8–5,283.4 gal/yr]}, covering a wide range of uncertainty in the flow rate; (iii) the experiments involved a sufficiently wide range for drip points (1 to 90 drip points) directly above or away from patches to address uncertainties associated with drip locations (SAR Section 6.5.1.1.2.4); and (iv) the experiments provided the range for the splash angle and the effective drip shield, which were used for calculating uncertainties associated with the F4 flow path. During its review, the NRC staff calculated weighted seepage rates per waste package to be 0.01, 0.04, 0.05, and 0.08 m^3/yr [2.6, 10.6, 13.2, and 21.1 gal/yr] using the information in the SAR (DOE, 2009av, Figure 2.3.3-47) for the mean seepage rate per waste package during the present-day, monsoon, and glacial climate states, and from 10,000 to 1 million years, respectively. These computed seepage rates are lower than the seepage rates used in DOE's Breached Drip Shield experiments; however, the NRC staff concludes that the lower DOE seepage rates in the Breached Drip Shield experiments are conservative from the perspective of radionuclide transport in the EBS because higher water flux rates would result in lower radionuclide mass concentrations.

Although upscaling and real-world repository conditions may introduce additional uncertainties, due to reasoning provided in (i) through (iv) in the previous paragraph, the NRC staff concludes that the Breached Drip Shield experiments adequately captured physical processes associated with the flow through the breached drip shield and waste packages. Hence, the NRC staff concludes that the use of the data from these experiments to bound uncertainties associated with the F4 flow path is adequate. Because the applicant implemented a mass-conserving flux-splitting algorithm in the EBS flow abstraction, and in light of the discussion in the previous section on Model Integration and Information Flow (last paragraph), the NRC staff concludes that (i) uncertainties associated with the drip flux and condensed water are adequately propagated into the EBS flow abstraction through the F1 flow path, (ii) uncertainties associated with the number of patches on a breached drip shield and a waste package are adequately propagated into the EBS flow abstraction through the F4 flow path, and (iii) uncertainties associated with flow conditions in the unsaturated zone around the invert are adequately propagated into the EBS flow abstraction through the F7 flow path. Finally, the uncertainties associated with the EBS flow abstraction and data are adequately propagated into the EBS radionuclide transport abstraction through the F6 flow path and the F8 flow path. Hence, the NRC staff concludes that data uncertainty within the EBS flow abstraction and between the EBS flow abstraction and other abstractions in the TSPA model is adequately propagated.

Model Support and Uncertainties

The NRC staff reviewed the information provided in SAR Section 2.3.7.12.3.5, references therein, and applicant responses to RAIs (DOE, 2009an, 2009av) to evaluate whether model support and uncertainties for the EBS flow abstraction are adequate, including characterization and propagation of model uncertainties. This evaluation focused on (i) the adequacy of the alternative conceptualizations, (ii) the adequacy of the justification for the inclusion or exclusion of the alternative conceptualizations, and (iii) the need or adequacy for comparison of model output with the results from other process-level models.

The applicant presented two alternative conceptualizations relevant to the EBS flow abstraction to characterize and propagate uncertainty through the model abstraction. These alternative conceptualizations included the bathtub flow model and the dual-continuum invert flow model (SAR Section 2.3.7.12.3.5).

DOE’s EBS flow abstraction is based on a flux-splitting algorithm that assumes a nonponding (no water accumulation) condition in the engineered barrier system. The applicant alternatively tested a ponding condition through a bathtub model that allows retaining and accumulating water in the waste package before releasing it to the EBS. DOE identified that a flow-through (nonponding) model is bounding for the bathtub (ponding) model in calculations of concentration and mass releases of radionuclides from the EBS due to the delays in releases in the bathtub case, when (i) radionuclides are solubility rate limited or dissolution rate limited and the inflow rate is time invariant or (ii) radionuclides are dissolution rate limited and there is a step change in the inflow rate. The applicant identified that the flow-through model is not bounding for the bathtub model when the inflow rate increases, because the flow-through model (with an increased volumetric water flow rate) would result in lower mass concentrations than the bathtub model (with a fixed, completely mixed bathtub storage volume). However, the total mass releases (unlike the mass concentrations) passed from the EBS model abstraction to the unsaturated transport abstraction would be identical for the flow-through and bathtub models.

DOE discussed another alternative conceptualization in which the flow domain in the invert is characterized as a dual-continuum model, as opposed to the single-continuum model, in the EBS flow abstraction. In this alternative model, the flow domain is divided into intergranular and intragranular flow domains. As a result, the F8 flow path is redefined as the flux from the intragranular invert continuum to the unsaturated zone. The applicant introduced an additional flow path, F9, which accounts for flux from the intergranular invert continuum to the unsaturated zone. The applicant did not include this conceptualization in the TSPA model due to insufficient experimental data to validate diffusion when, in transport simulations, the water content is very low.

The NRC staff reviewed DOE's proposed alternative model conceptualizations to determine whether the rationale for their inclusion or exclusion in DOE's TSPA model is adequate. The NRC staff concludes that the exclusion of the bathtub model from the TSPA code is appropriate because the flow-through model, as implemented in the TSPA code, is bounding for the bathtub model when flow rates are constant. On the other hand, the applicant noted that when the inflow rate increases and radionuclides are solubility rate limited, the difference in the performance of the bathtub model and the flow-through model is not critical to performance. The NRC staff concludes that the applicant's conclusion is reasonable because mass releases, rather than concentrations, of radionuclides are passed from the EBS to the unsaturated zone in the TSPA model and the mass of mobilized radionuclides (as a result of dissolution of waste forms) computed from the bathtub and flow-through models is identical in this case. The NRC staff further concludes that the exclusion of the dual-continuum model from the TSPA code is acceptable because the spatial distribution of flow in the invert, or flow between and within the invert materials, is not significant in determining radionuclide releases from the EBS into the unsaturated fractures and unsaturated matrix in the TSPA model construction.

Further, DOE demonstrated that the EBS flux-splitting algorithm tends to overestimate the fraction of drift flow that enters the breached mock-up drip shield (F2/F1) in the Breached Drip Shield experiments, as shown in SNL (2007aj, Table 6.5-2 and Figure D–12). On the basis of experimental data, the applicant estimated that the fraction of drift flow that entered the breached drip shield ranged from 0.013 to 0.275 with a median value of 0.049. DOE also used the results from the Breached Drip Shield experiments to estimate the fraction of drift flow that enters breached waste packages. Using the flux-splitting model, the applicant calculated that when approximately 4 percent of the waste package surface area is breached by general corrosion, 10, 90, and 100 percent of the seepage flux approaching the (failed) drip shield from above enters into a breached waste package for the 5^{th}, 50^{th}, and 95^{th} percentiles, respectively. In the TSPA model implementation, the average fraction of the breached waste package surface area in 1 million years is on the order of 10^{-3} for the nominal and disruptive modeling cases. The applicant estimated that 0–11 percent (with a mean/median value of 5.5 percent) and 0–12 percent (with a mean/median value of 6 percent) of the seepage flux above the (failed) drip shield entered into a breached commercial spent nuclear fuel waste package for the nominal and seismic ground motion cases, respectively (DOE, 2009an). The NRC staff conducted independent analyses to predict the number of breached waste packages that could be contacted by seepage flux (CNWRA and NRC, 2008aa). These analyses predicted on average that 2 percent and 4 percent of breached waste packages would be contacted by seepage flux for the nominal and seismic ground motion cases, respectively. The rest of the seepage flux was diverted around the breached waste packages.
Based on the discussion in the previous paragraph, the NRC staff concludes that the average 5–11 percent of seepage flux entering into a damaged commercial spent nuclear fuel or codisposal waste package used by DOE in TSPA calculations is consistent with breach flux rates (which had a median value of 5 percent of the inflow rate) obtained from Breached Drip

Shield experiments. Thus, NRC staff concludes that the applicant provided adequate modeling support using experimental data and TSPA model output because DOE's calculations for average percentage of seepage flux are consistent with staff's independent analyses.

Summary

The NRC staff reviewed the model conceptualization, mass-balance equations, the underlying assumptions of the EBS flow model abstraction, and other relevant abstractions with which the EBS flow model abstraction exchanges data and information. The NRC staff also reviewed the Breached Drip Shield experiments the applicant used to bound data uncertainties and the alternative model conceptualizations the applicant used to analyze model uncertainties. The NRC staff concludes that the applicant provided adequate information in the license application consistent with acceptance criteria in YMRP Section 2.2.1.3.3.3 and therefore satisfies the applicable regulatory requirements in 10 CFR 63.114. Also, NRC staff concludes that DOE's treatment of FEPs during the initial 10,000 years following permanent closure in this abstraction continues unchanged through the period of geologic stability (defined as 1 million years in 10 CFR 63.302). Therefore, the applicant's treatment of FEPs through the period of geologic stability is consistent with 10 CFR 63.114(b) and 63.342(c), and is acceptable.

The NRC staff concludes that the applicant adequately described (i) the EBS flow abstraction involving failed and intact EBS components under the nominal case and disruptive events, (ii) input to and output from the EBS flow abstraction and the information exchange between the EBS flow abstraction and other abstractions in the TSPA code, (iii) data support for bounding uncertainties associated with fluxes through a breached drip shield and a waste package and their propagation across the EBS flow abstraction and into other abstractions in the TSPA code, and (iv) alternative model conceptualizations for analyzing model uncertainties.

Finally, the NRC staff concludes that the igneous intrusion and seismic ground motion scenarios are the most significant contributors to the total dose for the 10,000- and 1-million-year simulations. In the TSPA code implementation, the fraction of seepage water at the end of 1 million years represents the maximum seepage flux that entered into damaged waste packages. TSPA calculations indicated that at the end of 1 million years, on average 5.5 percent of the seepage water approaching the (failed) drip shield entered into breached commercial spent nuclear fuel waste packages under nominal and seismic scenarios. Similarly, at the end of 1 million years, on average 6 and 11 percent of the seepage water approaching the (failed) drip shield entered into a breached codisposal waste package under nominal and seismic scenarios, respectively. Thus, on average, only up to 11 percent of the seepage flux above the (failed) drip shield would be available for the advective transport of radionuclides and colloids in the waste form and corrosion products domains in the EBS radionuclide transport abstraction. In this abstraction, a zero or nonzero value of the flux through a failed waste package determines the type of the transport mechanism for the radionuclides and colloids in the waste form and corrosion products domains (diffusive transport if the flux is zero; advective transport otherwise). Because, on average, not more than 11 percent of the seepage water can enter a waste package under any circumstances, the NRC staff concludes that breached waste packages consistently divert a large fraction (more than 89 percent) of drift seepage. However, as shown in Figure 6-1, this diverted flux around the failed waste package enters into the invert and is used to calculate F6, which determines the advective transport of radionuclides and colloids in the invert domain of the EBS radionuclide transport abstraction. Therefore, the NRC staff concludes that this implementation is acceptable, because advective transport of radionuclides in the invert is controlled by water fluxes in the invert.

2.2.1.3.3.4 Evaluation Findings

The NRC staff has reviewed DOE's SAR and other information submitted in support of the license application, which includes information required by 10 CFR 63.21(c)(1), (9), (10) and (15), and finds, with reasonable expectation, that the relevant requirements of 10 CFR 63.114 and 63.342 are satisfied regarding the abstraction of the quantity and chemistry of water contacting engineered barriers and waste forms. In particular, the NRC staff finds that DOE

- Included field data related to the geology, hydrology, and geochemistry of the surface and subsurface from the site and the region surrounding Yucca Mountain to define parameters and conceptual models used in the performance assessment calculation, in compliance with 10 CFR 63.114(a)(1)

- Accounted for uncertainty and variability in the parameter values used to model the quantity and chemistry of water contacting engineered barriers and waste forms, in compliance with 10 CFR 63.114(a)(2)

- Considered and evaluated alternative conceptual models that are consistent with currently available data and scientific understanding, in compliance with 10 CFR 63.114(a)(3)

- Provided a technical basis for the inclusion of FEPs affecting the quantity and chemistry of water contacting engineered barriers and waste forms, consistent with the limits on performance assessment in 10 CFR 63.342, and evaluated in sufficient detail those processes that would significantly affect repository performance, in compliance with 10 CFR 63.114(a)(4–6)

- Provided technical bases for the models of the quantity and chemistry of water contacting engineered barriers and waste forms used in the performance assessment to represent the 10,000 years after disposal, in compliance with 10 CFR 63.114(a)(7)

- Used performance assessment methods for the period of geologic stability consistent with the methods used to demonstrate compliance for the initial 10,000 years following disposal, in compliance with 10 CFR 63.114(b)

- Included in the performance assessment for the period of geologic stability those FEPs used for the initial 10,000-year period following disposal, consistent with specific limits, in compliance with 10 CFR 63.342(c)

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CHAPTER 7

2.2.1.3.4 Radionuclide Release Rates and Solubility Limits

2.2.1.3.4.1 Introduction

Safety Evaluation Report (SER) Section 2.2.1.3.4 contains the U.S. Nuclear Regulatory Commission (NRC) staff's evaluation of the U.S. Department of Energy's ("DOE" or "applicant") analytic models used in its Total System Performance Assessment (TSPA) computer program to evaluate the processes that could result in water transport of radionuclides out of the engineered barrier system, including the waste package and the invert, and into the unsaturated zone (the rock mass directly below the repository horizon and above the water table). As used in this SER, the term "abstraction" refers to the representation of site characterization data; process-level models for features, events, and processes (FEPs); uncertainty and variability; and their overall integration (in a simplified manner) in the TSPA code. These abstractions were described in Safety Analysis Report (SAR) Section 2.3.7 (DOE, 2009av) and in supporting documents, including the applicant's responses to the NRC staff's requests for additional information. The objective of this review is to evaluate whether or not the applicant's models for radionuclide release rates out of the engineered barrier system are acceptable.

The engineered barrier system and the transport pathway within the drift (repository tunnel) are the initial barriers to aqueous radionuclide release. If a waste package is breached and water enters the waste package, the radionuclides contained in the package may be transported from the engineered barrier system. The processes that could lead to this radionuclide release are affected by the chemical characteristics of the water, which in turn are affected by the materials that interact with the water. Therefore, as required by 10 CFR 63.113 and 10 CFR 63.114, the performance assessment analysis models radionuclide release rates from the engineered barrier system because these rates would significantly affect the timing and magnitude of transport for any radionuclide released from the repository.

The applicant identified five models it considered important for abstracting radionuclide releases from the engineered barrier system. The five models the applicant identified and the associated sections in this SER Section that address them are

1. The in-package chemical and physical environment model (SER Section 2.2.1.3.4.3.1) used to establish the conditions under which waste forms degrade and radionuclides are mobilized

2. The waste form degradation model (SER Section 2.2.1.3.4.3.2) used to calculate the rate at which the waste form degrades and the radionuclides become available for release

3. The concentration limits model (SER Section 2.2.1.3.4.3.3) used to apply chemically based upper limits on dissolved concentrations of some radionuclides

4. The availability and effectiveness of colloids model (SER Section 2.2.1.3.4.3.4) used to calculate the stabilities and concentrations of various types of colloids (small suspended particles that may mobilize radionuclides in water)

5. The engineered barrier system radionuclide transport model (SER Section 2.2.1.3.4.3.5) used to simulate radionuclide transport from the waste form, through the waste package, and out of the engineered barrier system

The FEPs that DOE identified as relevant to radionuclide release rates and solubility limits are listed in the applicant's SAR Section 2.3.7.2 and Table 2.3.7-1. The NRC staff evaluates the rationales for excluding FEPs from the performance assessment model in SER Section 2.2.1.2.1.3.2. In this SER section, the NRC staff finds acceptable the applicant's bases for the list of FEPs considered and excluded from the TSPA code analysis that are relevant to waste form behavior, solubility limits, colloidal transport, and radionuclide release rates.

In addition, the NRC staff finds that the applicant's identification and screening of scenario classes considered all credible processes and events that could lead to radionuclide release (see SER Sections 2.2.1.2.1.3.3 and 2.2.1.2.1.3.4). Evaluations of those FEPs included in the performance assessment are discussed under the five topical areas in this SER Section.

This SER Section relies on the following information as inputs for its analysis: (i) design details of the waste package, waste form, and internal components of the waste package (SER Sections 2.1.1.2.3.4 and 2.1.1.2.3.5); (ii) context for consideration of the barrier capabilities of the waste package and the drift (SER Sections 2.2.1.1.3.2.2, 2.2.1.1.3.2.3, 2.2.1.1.3.2.4, and 2.2.1.1.3.2.5); (iii) information on corrosion (SER Section 2.2.1.3.1) and mechanical failure (SER Section 2.2.1.3.2) of the drip shield and waste package, which may allow water into the waste package; and (iv) information on the rate of delivery of water to the waste package surface and the chemical characteristics of water that may enter the waste package (SER Section 2.2.1.3.3 and 2.2.1.3.6).

The output from the model of radionuclide release rates and solubility limits is used as input to the model for radionuclide transport in the unsaturated zone. The information the unsaturated zone model requires for calculating the movement of the radionuclides includes the rates and magnitudes of radionuclide release from the drift, including the characteristics of dissolved and colloidal species. Information from this model is also used for evaluating the barrier capability of the waste package interior, the waste form, and the drift below the waste package (e.g., the invert) and for supporting the scenario analysis for the engineered barrier system.

2.2.1.3.4.2 Regulatory Requirements

The applicant is required to provide information specified in 10 CFR 63.21(c)(3),(9), and (15) related to radionuclide release rates and solubility limits. The requirements in 10 CFR 63.114 (Requirements for Performance Assessments) and 10 CFR 63.342 (Limits on Performance Assessments) include postclosure performance assessment to demonstrate compliance with 10 CFR 63.113 (Performance Objectives for the Geologic Repository after Permanent Closure). Compliance with 10 CFR 63.113 is reviewed in SER Section 2.2.1.4.1.

The regulations for performance assessment in 10 CFR 63.114 require, in part, that a performance assessment

- Include data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the surface and subsurface from the site and the region surrounding Yucca Mountain [10 CFR 63.114(a)(1)]

- Account for uncertainty and variability in the parameter values [10 CFR 63.114(a)(2)]

- Consider and evaluate alternative conceptual models [10 CFR 63.114(a)(3)]

- Provide technical bases for either the inclusion or exclusion of FEPs, including effects of degradation, deterioration, or alteration processes of engineered barriers that would adversely affect performance of the natural barriers, consistent with the limits on performance assessment in 10 CFR 63.342, and evaluate in sufficient detail those processes that would significantly affect repository performance [10 CFR 63.114(a)(4-6)]

- Provide a technical basis for the models used in the performance assessment to represent the 10,000 years after disposal [10 CFR 63.114(a)(7)]

The NRC staff's evaluation of inclusion or exclusion of FEPs is provided in SER Section 2.2.1.2.1. Requirements for performance assessment for the initial 10,000 years following disposal are specified in 10 CFR 63.114(a). Requirements for the performance assessment methods for the time from 10,000 years through the period of geologic stability, defined in 10 CFR 63.302 as 1 million years following disposal, are specified in 10 CFR 63.114(b) and 10 CFR 63.342. These sections provide that through the period of geologic stability, with specific limitations, the applicant

- Use performance assessment methods consistent with the performance assessment methods used to demonstrate compliance for the initial 10,000 years following permanent closure [10 CFR 63.114(b)]

- Include in the performance assessment those FEPs used in the performance assessment for the initial 10,000-year period (10 CFR 63.342)

For this model abstraction of radionuclide release rates and solubility limits, 10 CFR 63.342(c)(1) further provides that the applicant assess the effects of seismic and igneous activity on repository performance, subject to the probability limits in 10 CFR 63.342(a) and 10 CFR 63.342(b). Specific constraints on the analysis required for seismic and igneous activity are provided in 10 CFR 63.342(c)(1)(i) and 10 CFR 63.342(c)(1)(ii), respectively.

The NRC staff's review of the SAR and supporting information follows the guidance in Yucca Mountain Review Plan (YMRP) Section 2.2.1.3.4 (NRC, 2003aa), Radionuclide Release Rates and Solubility Limits, as supplemented by additional guidance for the period beyond 10,000 years after permanent closure (NRC, 2009ab). The YMRP acceptance criteria for model abstractions that provide guidance for the NRC staff's review of the applicant's abstraction of radionuclide release rates and solubility limits are

1. System description and model integration are adequate
2. Data are sufficient for model justification
3. Data uncertainty is characterized and propagated through the abstraction
4. Model uncertainty is characterized and propagated through the abstraction
5. Model abstraction output is supported by objective comparisons

In its review of the SAR and supporting information, the NRC staff used a risk-informed approach for aspects of radionuclide release rates and solubility limits important to repository performance. The NRC staff considered all five criteria provided in the YMRP in its review of information provided by DOE. In the context of these criteria, only those aspects of the model abstraction that substantively affect the performance assessment results, as determined by the

NRC staff, are discussed in detail in this SER section. The NRC staff's determination is based both on risk information provided by DOE, and on the NRC staff's knowledge gained through experience and independent analyses.

2.2.1.3.4.3 Technical Review

2.2.1.3.4.3.1 In-Package Chemical and Physical Environment

This section details the NRC staff's review of the applicant's abstraction and the TSPA implementation of in-package chemistry, as described in SAR Section 2.3.7.5 and references cited therein. The in-package chemistry model estimates the water chemistry inside breached waste packages and generates abstractions for pH, ionic strength, and fluoride concentration. Water chemistry inside the waste package (especially pH and ionic strength) is important to repository performance because it controls waste form degradation, radionuclide solubilities, and the suspension stabilities of colloids.

The NRC staff's review focused on aspects of the in-package chemistry model considered significant to risk, including conceptual design and implementation, data inputs, model limitations, sensitivity to environmental conditions, and model abstraction and support. Primary data inputs to the in-package chemistry model include (i) the compositions, surface areas, and degradation rates of waste forms and metal components in the waste package; (ii) incoming water chemistries; and (iii) the thermodynamic data used to calculate the stabilities of aqueous- and gas-phase species in the waste package. The applicant used the sensitivity of the model to variations in environmental conditions (e.g., liquid influx rate, $p\mathrm{CO_2}$, and temperature) to determine the potential effects of disruptive events (e.g., seismic or igneous activity) on model outputs.

Conceptual Design and Implementation

The applicant's in-package chemistry conceptual model consists of a batch reactor system composed of water, oxygen, carbon dioxide, waste forms, and metal alloys. The batch reactor system is in equilibrium with atmospheric conditions, and reactants degrade in the presence of water according to a rate determined by the physical properties and exposed surface area of each reactant. During the reactions, secondary mineral phases and metal (hydr)oxide corrosion products precipitate and water changes in composition and mass. The model simulates two water ingress conditions: (i) vapor influx, under which water vapor (simulated as pure water) is assumed to condense and react with internal waste package components; and (ii) liquid influx, under which seepage or "dripping" water (simulated as typical groundwater or drift wall condensate) enters a breached waste package, reacts with internal components, and then exits by advection. Vapor influx is included in the model because water films generated by vapor influx promote radionuclide diffusion, which is simulated in the engineered barrier system using a diffusion model. The NRC staff's evaluation of the diffusion model can be found in SER Section 2.2.1.3.4.3.5.

The applicant's model considers both commercial spent nuclear fuel (SNF) and codisposal waste packages. Commercial SNF waste packages in the model contain 21 pressurized water reactor (PWR) fuel assemblies (21-PWR). Codisposal waste packages in the model contain two DOE multicanister overpacks and two defense high-level waste canisters (2-MCO/2-DHLW). The applicant's model divides the waste packages into two domains: the waste form domain (Cell 1) and the corrosion products domain (Cell 2). Cell 1 of the codisposal waste packages is further divided into Cell 1a, represented by two high-level waste glass pour canisters (2-DHLW), and Cell 1b, represented by two multicanister overpack units containing N-Reactor fuel (2-MCO). Adsorption reactions are not simulated in the waste form cells because the amount of iron corrosion products is low compared to Cell 2 (i.e., the corrosion products domain). Adsorption reactions within Cell 2 are simulated in the engineered barrier system flow and transport model; the NRC staff's evaluation of this model can be found in SER Section 2.2.1.3.4.3.5. The applicant excluded other FEPs that could potentially impact in-package chemistry, such as in-package criticality, oxide-wedging, radiolysis, and microbial activity on the basis of low probability or low consequence. In SER Section 2.2.1.2.1.3.2, the NRC staff evaluates and finds acceptable the applicant's rationales for excluding these processes.

The applicant used the geochemical reaction path equilibrium modeling code EQ6 to simulate interaction of water and materials in commercial SNF Cell 1 and codisposal Cells 1a and 1b (referred to collectively as waste form cells). For vapor influx, water is added to the waste form cells as one of the reactants at a rate corresponding to the maximum diffusion rate of vapor through openings in a breached waste package. For liquid influx, the solid-centered flow-through option of the EQ6 code is used to simulate the flow of source water into and through a constant-volume, well-mixed batch reactor. Under the flow-through option, an amount of source water is added to the reactor displacing an equal amount of water already equilibrated with the solid phases in the reactor. The water in the reactor then mixes completely, and the water, solids, and gases within the reactor equilibrate again. Kinetically controlled reactants are also added to the reactor prior to equilibrium to capture the case where the residence time within the reactor is sufficiently short that equilibrium cannot be reached with slowly degrading constituents (e.g., metal degradation). The ratio of water to reactants, which depends on liquid influx rate, is treated as a variable in the EQ6 model. At high liquid influx rates, the ratio is such that the materials of the water package are in contact with a volume of water equal to that of the void space. This case is referred to as the "bathtub" model and has the highest ratio of water to waste package materials. At low liquid influx rates, the ratio is such that the volume of water in contact with waste package materials is less than the void space. The applicant examined the impact of varying the water-to-reactants ratio in a sensitivity analysis to evaluate the effects of differing flow conditions on in-package water chemistry BSC [2005ad, Section 6.6.1(a)]. The sensitivity analysis indicated that flow conditions had a negligible effect on pH but had a distinct effect on ionic strength (i.e., as the ratio of water to reactants is decreased to simulate low flow conditions, the ionic strength of the solution increases).

NRC Staff's Review

The NRC staff evaluated the modeling approach and the representation of the commercial SNF and codisposal waste packages used in the in-package chemistry conceptual design. The NRC staff finds that the applicant's use of a thermodynamic equilibrium chemical approach of a flow-through, well-mixed batch reactor is acceptable because it reasonably represents in-package chemistry processes with sufficient flexibility to project the range of chemical conditions expected inside a breached waste package. Kinetics effects on chemical conditions inside the waste package are adequately captured in the model by simulating a range of experimentally derived degradation rates for each waste form and metal component in the waste form cells. The effects of disruptive events on model outputs are adequately evaluated through a sensitivity analysis to determine the effects of differing flow conditions on in-package water chemistry. The NRC staff finds that the representation of commercial SNF and codisposal waste packages in the in-package chemistry model is consistent with waste package designs documented in SAR Section 1.5.2.

The NRC staff evaluated the EQ6 modeling code. On the basis of this evaluation, the NRC staff finds that the mathematical representation of geochemical processes in the EQ6 modeling code is acceptable because it is consistent with generally accepted approaches for simulating geochemical interactions among fluids, gases, and solid materials. The NRC staff also finds that the EQ6 code is acceptable for predicting the chemistry of in-package fluids because it addresses and simulates the geochemical processes that are important to in-package chemistry, including (i) interactions between codisposed waste, (ii) chemical effects of void space in the waste package, (iii) chemical characteristics of water in the waste package, (iv) oxidation-reduction potential in the waste package, (v) reaction kinetics in the waste package, and (vi) chemistry of water flowing into the waste package (SAR Table 2.3.7-1).

Waste Form and Metal Alloy Compositions, Surface Areas, and Degradation Rates

The applicant derived input data for the in-package chemistry model from existing government design documents, standard reference material specification documents, and open literature information. The sources of input data to the in-package chemistry model were justified and documented in BSC [2005ad, Sections 4.1 and 4.1(a), Tables 4.1 and 4.1(a)]. The input data included the compositions, surface areas, and degradation rates of waste forms (e.g., PWR fuel, high-level waste glass, and N-Reactor fuel) and material components of the waste form cells (e.g., stainless steels and aluminum alloys).

NRC Staff's Review

The NRC staff evaluated the input data the applicant used to define the surface areas and compositions of waste forms and material components in the waste form cells by reviewing existing government documents and open literature information, as discussed and referenced in this paragraph. The NRC staff verified that for each solid reactant in the waste form cells, the surface area available to react was calculated from dimensions of the internal components of a 21-PWR or 2-MCO/2-DHLW package (SNL, 2007bm,bn). Also, the NRC staff reviewed the composition of the PWR fuel used for commercial SNF (Cell 1) that was based on an initial enrichment of 2 wt% to 5 wt% U-235 and a burnup of 0 to 50 GWd/MTU (gigaWatt-days per metric ton of uranium) (BSC, 2003af). These enrichments bound the typical PWR fuel available for disposal. The composition of N-Reactor fuel (Cell 1b) was taken from DOE (2000aa), which contains detailed information on multicanister overpack compositions and dimensions. The NRC staff reviewed the composition of high-level waste glass (Cell 1a) that was derived from qualified data on the composition of high-level waste glasses from the Savannah River laboratory (Allison, 2004aa). The compositions of material components in the waste form cells (i.e., stainless steels and aluminum alloys) were based on the American Society for Testing and Materials standards. On the basis of the NRC staff's evaluation of the applicant's input data, the NRC staff finds the applicant's data used to define the surface areas and compositions of each solid reactant in the waste form cells acceptable because they are consistent with published data, and are appropriate for use in process-level in-package chemistry model simulations.

The NRC staff also reviewed the degradation rates the applicant used. With the exception of the N-Reactor fuel, the applicant selected degradation rates for waste forms and material components in the waste form cells on the basis of experimental measurements (BSC, 2004ae,ah,ai,ao). The NRC staff finds this acceptable because the applicant appropriately derived minimum, maximum, and base case degradation rates for each solid reactant and captured the full range (i.e., uncertainty) of potential degradation rates in its analytic model. For the N-Reactor fuel, the applicant assumed that the N-Reactor fuel degraded instantaneously upon waste package breach (SAR Section 2.3.7.8), which is conservative because this approach would maximize the effect of waste form degradation on in-package chemistry. The NRC staff finds that the range of waste form and material degradation rates is adequate for establishing initial and boundary conditions for in-package chemistry model simulations because these ranges are consistent with experimental measurements or are based on conservative assumptions.

Incoming Water Chemistry

The applicant incorporated a range of Yucca Mountain pore water and basalt water chemistries in developing the in-package chemistry model abstractions. For seepage water input for the nominal and seismic scenarios, the applicant selected four Yucca Mountain pore water compositions (SAR Table 2.3.7-9). The applicant also included J–13 well water chemistry as a potentially relevant seepage water because its composition is generally representative of water compositions in the saturated and unsaturated zones in the vicinity of Yucca Mountain (Harrar, et al., 1990aa). For seepage water input for the igneous intrusion case, the applicant selected three groundwaters from large, fractured basalt reservoirs (SAR Tables 2.3.7-10 and 2.3.7-11).

NRC Staff's Review

The NRC staff evaluated the chemistry of incoming waters used in the in-package chemistry model to simulate seepage water input. The NRC staff independently verified that the chemical compositions of the pore waters and the J–13 well water span the range of predominant water types found in the Topopah Spring welded tuff at Yucca Mountain, as described in SNL (2007ak, Section 6.6.5). The NRC staff finds that the applicant adequately limited uncertainty in the initial chemistry of water entering the waste package by incorporating a range of Yucca Mountain pore water and basalt water chemistries in developing the in-package chemistry model abstractions. In BSC [2005ad, Sections 6.5(a) and 6.6(a)] and in SAR Figures 2.3.7-13 through 2.3.7-18, the applicant demonstrated limited sensitivity of in-package chemistry to the incoming water composition. Therefore, the NRC staff finds that the chemistry of incoming waters the applicant used to simulate seepage water input are acceptable for developing the in-package chemistry model abstractions.

Thermodynamic Database

The applicant used the thermodynamic database *data0.ymp.R5* to execute EQ6 simulations (SAR Section 2.3.7.5.2). This database allows for the calculation of mineral and gas solubilities, the chemical state of dissolved species, and the dissolution rates of solids. Uncertainty in the *data0.ymp.R5* thermodynamic database is implicit because it was constructed from the accumulation of a large number of experimental measurements or model estimations, each with its own associated uncertainty. To minimize this uncertainty, the applicant evaluated experimental data and observations from natural analogs, as identified in BSC [2005ad, Section 7.4.3(a)], and the results of sensitivity analyses, as detailed in BSC (2005ad, Section 6.6.11), to select appropriate secondary phase formation for use in process-level model simulations.

NRC Staff's Review

The NRC staff reviewed the thermodynamic database for the in-package chemistry model. The NRC staff finds that the thermodynamic database *data0.ymp.R5* is appropriate because it includes the elements that constitute the waste package, waste form, incoming seepage water, and gas compositions at the temperature expected in the drifts {25 °C [77 °F]} and the thermodynamic data on secondary mineral phases important to the in-package chemistry model (SNL, 2007at; SER Section 2.2.1.3.3). The NRC staff also finds that the applicant appropriately added thermodynamic data for several new mineral phases that could potentially affect model outputs. These data included nickel carbonate, nickel molybdate, and several uranium minerals that precipitate in UO_2 degradation experiments and occur in the vicinity of natural UO_2 ore deposits. The NRC staff reviewed and evaluated the experimental data, evidence from natural analogs, and results of sensitivity analyses used to guide secondary phase selection for use in process-level model simulations. On the basis of this review and evaluation, the NRC staff finds that the applicant's approach to limiting uncertainty in the EQ6 simulations associated with the *data0.ymp.R5* database is acceptable because appropriate secondary minerals were chosen for use in developing the in-package chemistry abstractions.

Model Limitations

The applicant addressed model limitations associated with the accumulation of water inside the waste package and the evolution of material component surface area and void space due to corrosion product buildup by implementing the following assumptions in the process-level EQ6 simulations: (i) when a waste package is breached, the entire contents of the waste package are instantly exposed to oxygen, carbon dioxide, and water, and (ii) secondary mineral formation and buildup inside the waste package do not reduce available void space in the waste package and do not reduce exposure of waste package internals to atmospheric gases and water (i.e., void volume and internal component surface areas are fixed and do not vary with time).

NRC Staff's Review

The NRC staff evaluated the assumptions the applicant used to address model limitations. The NRC staff finds that instantly exposing the entire contents of the waste package to atmospheric gases and water upon breach is acceptable because it increases the potential for the model to predict enhanced radionuclide release. The NRC staff finds the applicant acceptably supported the model by applying this conservative assumption. Reducing exposure of waste forms to gases and water by filling void spaces with degradation products enhances waste isolation by limiting the impact of waste form degradation on solution chemistry. Therefore, the NRC staff finds that fixing void volume and waste package component surface areas is acceptable because it is a conservative analysis that results in faster waste form degradation, which enhances the potential for radionuclide release and transport. On the basis of these reviews, the NRC staff finds that the assumptions the applicant used to address model limitations are acceptable because they will not result in an underestimation of risk.

Environmental Conditions and Sensitivity Analyses

The applicant developed the abstractions for in-package chemistry by analyzing the results of process-level model simulations applied over the following range of environmental conditions: (i) a pCO_2 range of 10^{-4} to $10^{-1.5}$ bars, (ii) a liquid influx rate of 0.1 to 1,000 L/yr [0.026 to 260 gal/yr], (iii) a temperature range of 25 to 100 °C [77 to 212 °F], and (iv) a relative humidity range for vapor influx of 95 to 100 percent. The applicant performed sensitivity analyses to evaluate the effects of uncertain thermal-hydrologic-chemical input parameters on model outputs and approximate model uncertainty for propagation into the TSPA model. Parameters with significant effects on model outputs (e.g., pCO_2 for pH and liquid influx rate and relative humidity for ionic strength) were incorporated as independent variables in the model abstractions. Within the TSPA code, the values of these independent variables are provided by other TSPA submodels (e.g., the engineered barrier system thermal-hydrologic environment submodel provides relative humidity, the engineered barrier system chemical environment submodel provides pCO_2, and the engineered barrier system flow submodel provides liquid influx rate) (SNL, 2008ag). Parameters with smaller effects on model outputs (specifically, material degradation rates) were used to quantify model uncertainty for propagation to the TSPA model.

NRC Staff's Review

The NRC staff reviewed and evaluated the range of environmental conditions that was applied to process-level model simulations to develop the abstractions for in-package chemistry. The NRC staff finds that the applied range of pCO_2 will bound the range of pCO_2 that could potentially exist within emplacement drifts at Yucca Mountain. The NRC staff finds that the applied range of liquid influx rate is appropriate because it bounds the range of flow rate used to characterize the uncertainty in liquid flow through a breached waste package (SER Section 2.2.1.3.3.3.3). Because liquid water, which is required for the modeled reactions to take place, is precluded in the applicant's abstraction from entering the drift at temperatures above 100 °C [212 °F; the boiling point of liquid water], the NRC staff finds that the applicant's model temperature range of 25 to 100 °C [77 to 212 °F] reasonably accounts for the temperature range expected inside breached waste packages. The NRC staff finds the range of relative humidity the applicant applied over which vapor influx conditions are simulated is acceptable because relative humidity less than 95 percent will result in few interconnected surface water films and negligible diffusion of radionuclides, consistent with SNL (2007aj). On the basis of these findings, the NRC staff concludes that the in-package chemistry model reasonably accounts for the range of environmental conditions that could reside inside breached waste packages under liquid and vapor influx conditions for the nominal and disruptive event scenario classes.

The NRC staff reviewed the sensitivity analyses used to evaluate the effects of uncertain thermal-hydrologic-chemical input parameters on model outputs. The NRC staff verified that pCO_2 had a significant effect on model outputs for pH, as detailed in BSC [2005ad, Section 6.6.3(a)], and the liquid influx rate and relative humidity had significant effects on model outputs for ionic strength, as described in BSC [2005ad, Sections 6.6.4(a) and 6.5.2(a)]. The NRC staff finds that the approach of incorporating these parameters as independent variables in the model abstractions (i.e., pCO_2 for pH and liquid influx rate and relative humidity for ionic strength) is acceptable because it ensures adequate integration and coupling of thermal-hydrologic-chemical processes in the model abstractions. On the basis of review and evaluation of the applicant's sensitivity analyses, the NRC staff finds that, after pCO_2, relative humidity, and liquid influx rate, the material degradation rates had the greatest effect on model outputs, as outlined in BSC [2005ad, Section 6.6.5(a)]. Therefore, NRC staff finds that using the results of material degradation rate sensitivity analyses to quantify model uncertainty for propagation to the TSPA model is appropriate.

Abstractions for pH

The applicant's in-package chemistry abstractions for pH provide parameter distributions in the form of lookup tables for the TSPA code. Lower and upper pH limits for liquid and vapor influx in each waste form cell were quantified by simulated acid and base titrations over a range of pCO_2 and ionic strength (SAR Figures 2.3.7-19 to 2.3.7-21). Abstracted pH ranges were defined by secondary oxides, the presence of which limits the range of in-package pH through solubility reactions. The lower pH limit was set by dissolution of trevorite ($NiFe_2O_4$), which accumulates as the steel degrades. The upper pH limit was set by dissolution of schoepite ($UO_3•2H_2O$), which precipitates as UO_2 fuel degrades and CO_2 reaches equilibrium conditions. To capture the uncertainty, the pH values at any given $p$$CO_2$ and ionic strength were assumed to be uniformly distributed between the pH limits established by the titration calculations. The applicant supported estimated ranges for pH in the waste form cells by comparing predicted secondary mineral phases and pH ranges to observations from natural soils and groundwater, natural analogs, and/or laboratory experiments.

NRC Staff's Review

The NRC staff evaluated the modeling approach and information the applicant used to generate and support the in-package chemistry abstractions for pH. The NRC staff finds that the solubility limit approach used to quantify lower and upper pH limits is consistent with accepted geochemical principles. The NRC staff evaluated information provided in DOE (2009ax, Enclosure 4) and finds that the choice of waste package design the model used (e.g., a 5-DHLW/DOE codisposal waste package containing five high-level waste glass containers versus the 2-MCO/2-DHLW codisposal waste package containing two high-level waste glass canisters) does not affect established pH limits because the pH limits are based on buffering reactions that are not influenced by the total volumes and surface areas of material components in the waste form cells. In addition, on the basis of its review of information provided in BSC [2005ad, Sections 6.5(a), 6.6.3(a), and 6.6.5(a)], the NRC staff also finds that lower and upper pH limits defined for each waste form cell in the pH abstractions are appropriate because they are within the pH trends observed in time-dependent basecase EQ6 simulations at different incoming water chemistries and in sensitivity analyses at varying $p$$CO_2$ values and material degradation rates. On the basis of open literature reviews, the NRC staff finds that the phases the applicant predicted to form and control pH in the waste form cells (i.e., trevorite and schoepite) are consistent with the phases reported as alteration products in steel corrosion and UO_2 degradation experiments, as well as in phases observed at natural analogs (Wang, et al., 2001aa; Da Cunha Belo, et al., 1998aa; BSC, 2004ah; Wronkiewicz, et al., 1996aa; Langmuir, 1997aa; Pearcy, et al., 1994aa).

The NRC staff finds that the applicant's abstracted pH ranges are consistent with the pH values measured in qualitatively similar soils and groundwaters and to pH ranges observed in UO_2 degradation experiments (Hem, 1995aa; Wronkiewicz, et al., 1996aa). On the basis of the NRC staff's evaluation of information the applicant provided, the NRC staff finds that the solubility-controlling secondary oxide phases selected to quantify lower and upper pH limits in the applicant's pH abstractions are acceptable and that abstracted pH limits are acceptable.

Abstractions for Ionic Strength

As with pH, the applicant's in-package chemistry abstractions for ionic strength provide parameter distributions in the form of lookup tables for the TSPA code. However, the manner by which ionic strength is abstracted differs under liquid and vapor influx conditions.

Under liquid influx conditions, the applicant derived abstractions for ionic strength from a series of time-dependent EQ6 simulations at different liquid influx rates. The applicant approximated uncertainty in ionic strength on the basis of variation in ionic strength observed in material degradation rate sensitivity analyses. At low liquid influx rates, the model generates high ionic strengths in the waste form cells (SAR Figures 2.3.7-22 to 2.3.7-24) because low flow rates provide sufficient residence time for the buildup in solution of waste form and metal alloy degradation products. The applicant supported high ionic strength predictions in the waste form cells by comparing predicted ionic strengths to observations from natural groundwater and laboratory experiments.

For vapor influx conditions, the applicant abstracted ionic strength as a function of relative humidity. At ionic strengths of 1 molal or less (relative humidity above ~98.5 percent), vapor influx was simulated using EQ6 and the B-dot equation to calculate activity coefficients to derive linear relationships between relative humidity and ionic strength (SAR Figure 2.3.7-25). When the ionic strength exceeded 1 molal (relative humidity at or below 98.5 percent), Pitzer calculations for simple salt solutions from the in-drift precipitates/salts model (SNL, 2007ao) were used to approximate the relationship between relative humidity and ionic strength, as described in BSC [2005ad, Section 6.10.2.2(a)]. Uncertainty in ionic strength was derived from ionic strength variations observed in the Pitzer calculations.

NRC Staff's Review

For liquid influx, the NRC staff evaluated the modeling approach and technical support the applicant used to generate ionic strength abstractions and makes the following findings. The NRC staff finds that the derived ionic strength ranges, from output of time-dependent EQ6 simulations using the *data0.ymp.R5* thermodynamic database, are acceptable because the ranges are calculated from a complete and appropriate set of aqueous species at equilibrium. The NRC staff reviewed the applicant's sensitivity analysis results of liquid influx rate and material degradation rate on in-package ionic strength in BSC (2005ad, Sections 6.6.4[a] and 6.6.5[a]). The variation in the liquid influx rate entering the waste package has a significant effect on ionic strength (i.e., ionic strength significantly increases as liquid influx rate decreases and significantly decreases as liquid influx rate increases). Therefore, the NRC staff finds that deriving ionic strength as a function of liquid influx rate is appropriate because it provides a means for bounding in-package ionic strength over the entire range of flow conditions expected to enter a breached waste package. The variation in material degradation rates has a smaller effect on ionic strength. Therefore, the NRC staff finds that approximating uncertainty in ionic strength as a function of material degradation rates is acceptable because it provides a means for bounding ionic strength on the basis of material components within the waste package. The NRC staff finds appropriate the applicant's supporting information showing that the relationship between low liquid influx rate and high ionic strength is consistent with the evolution of deep groundwater brines in Canadian Shield granite (Appelo and Postma, 1994aa) and the results of UO_2 degradation experiments (Wronkiewicz, et al., 1996aa).

For vapor influx, the NRC staff reviewed and evaluated the modeling approach used to generate the ionic strength abstractions and makes the following findings. The NRC staff finds acceptable the applicant's approach of deriving the ionic strength abstractions for vapor influx as a function of relative humidity because, under vapor influx conditions, the water activity is controlled by relative humidity and ionic strength is strongly related to the water activity. At ionic strength values of 1 molal or less, deriving ionic strength from the output of EQ6 simulations using the *data0.ymp.R5* thermodynamic database is acceptable because the values are calculated from a complete and appropriate set of aqueous species at equilibrium. However, for solutions with ionic strengths greater than 1 molal, deriving ionic strength from the output of EQ6 simulations using the B-dot activity coefficient is inappropriate because precipitates may form at high ionic strength resulting in large uncertainties in calculated pH and ionic strength. The NRC staff evaluated the applicant's in-drift precipitates/salts model (SNL, 2007ao), which uses a Pitzer ion-interaction model to predict chemical conditions at high ionic strength. On the basis of this evaluation, the NRC staff finds that the in-drift precipitates/salts model is appropriate for approximating the relationship between relative humidity and ionic strengths exceeding 1 molal.

Abstraction for Fluoride Concentration

The applicant's fluoride abstraction provides maximum fluoride values for discrete ionic strength intervals for each waste form cell. Although high-level waste glass may contain some fluoride, the major source of fluoride in breached waste packages is from liquid influx (i.e., incoming water). Under vapor influx conditions (where water vapor, simulated as pure water, is assumed to condense inside the waste package), there is no significant source of fluoride in the waste form cells and maximum fluoride concentration is set to zero. Therefore, the fluoride abstraction is only applicable under liquid influx conditions. At high ionic strengths, maximum fluoride values were selected on the basis of relationships between fluoride concentration and ionic strength observed in material degradation rate sensitivity analyses at various incoming water compositions. At low ionic strength, maximum fluoride concentration was set to the maximum concentration observed in pore waters from the Topopah Spring welded tuff: 4.8 mg/L [0.00025 molal] (SNL, 2007ak).

NRC Staff's Review

The NRC staff evaluated the modeling approach and information the applicant used to generate and support the in-package chemistry abstraction for fluoride. On the basis of its review, the NRC staff finds that the major source of fluoride in breached waste packages comes from liquid influx. At high ionic strengths, fluoride often concentrates as incoming water is consumed by degradation reactions, resulting in fluoride levels that tend to correlate with ionic strength. Therefore, the NRC staff finds that the approach used to select maximum fluoride levels at high ionic strength is appropriate because it is based on relationships between fluoride concentration and ionic strength observed in material degradation rate sensitivity analyses at varying incoming water compositions. In addition, on the basis of its review and evaluation of information provided in BSC [2005ad, Section 6.10.3(a)], the NRC staff finds that the applicant's maximum abstracted fluoride levels are set conservatively high when compared to fluoride levels observed in model output from the degradation rate sensitivity analyses at varying incoming water compositions. At low ionic strengths, the fluoride concentration either remains in the vicinity of the concentration of the incoming liquid or decreases due to mineral precipitation. Therefore, the NRC staff finds that setting maximum fluoride concentrations to those observed in pore waters from the Topopah Spring welded tuff {i.e., 4.8 mg/L [0.00025 molal]} is appropriate at low ionic strengths because this is the maximum possible fluoride concentration that can occur in the waste package without water loss due to degradation reactions. On the basis of this evaluation, the NRC staff finds that the approach the applicant used to set maximum fluoride levels is acceptable and will not result in an underestimation of risk.

Summary of NRC Staff's Review of In-Package Chemical and Physical Environment

The NRC staff finds that, in modeling the in-package chemical and physical environment, the applicant appropriately incorporated design features of commercial SNF and codisposal waste packages. The applicant used appropriate conceptual and mathematical models and assumptions to simulate geochemical interactions between fluids, gases, and internal components of the waste package and generate abstractions for pH, ionic strength, and fluoride concentration. The applicant used sufficient and technically defensible data to establish initial and boundary conditions for model simulations. These data included the thermodynamic properties of solids, gases, and aqueous species; incoming water chemistries; and the compositions, surface areas, and degradation rates of waste forms and material components of the waste package. The NRC staff finds that model simulations were appropriately applied over the full range of environmental conditions that might be expected inside breached waste packages. The NRC staff finds that the applicant appropriately approximated model uncertainty for propagation into the TSPA code by performing sensitivity analyses to assess the effects of uncertain thermal-hydrologic-chemical parameters on model outputs. Parameters with significant effects on model outputs (pCO_2 for pH and liquid influx rate and relative humidity for ionic strength) were incorporated as independent variables in the model abstractions. Parameters with smaller effects (material degradation rates) were used to quantify model uncertainty for propagation to the TSPA model. The applicant provided support for the in-package chemistry pH abstractions by comparing predicted secondary mineral phase formation and estimated pH ranges to observations from natural soils and groundwater, natural analogs, and laboratory experiments. Support for the in-package chemistry ionic strength abstractions was provided by comparing predicted high ionic strengths to observations from natural groundwater and laboratory experiments. Therefore, on the basis of its review, the NRC staff finds acceptable the applicant's abstraction and TSPA model implementation of the in-package chemical and physical environment.

2.2.1.3.4.3.2 Waste Form Degradation

This section describes the NRC staff's review of the applicant's abstraction and TSPA model implementation of radionuclide mobilization from waste form degradation. This radionuclide mobilization determines the quantity of radionuclides that may be transported by water from the solid waste form and eventually to the accessible environment. The waste form types include commercial SNF, high-level waste glass, and DOE SNF, as described in SAR Section 1.5.1.

Commercial SNF is composed of irradiated fuel rods from PWRs and boiling water reactors. High-level waste glass is made by melting high-level radioactive materials with silica and/or other glass-forming chemicals and then solidifying them. DOE SNF (including naval SNF) comes from a range of high-level waste generators, from noncommercial reactors, and from the use of radioactive material that encompasses a variety of fuel types. On the basis of the significance to risk, the NRC staff's review focused on the inventory of radionuclides and radionuclide distribution in the following areas: commercial SNF; degradation of commercial SNF; degradation of high-level waste glass; degradation of DOE SNF, naval SNF, and cladding; and associated model and data uncertainties, including waste form degradation under disruptive scenarios and microbial effects. Each waste form has a specific radionuclide inventory.
In the nominal scenario, the waste form degrades as it dissolves after the cladding, if any, corrodes and fails in the aqueous environment. In seismic or igneous scenarios, mechanically or thermally assisted degradation could also occur. For the applicant's waste form degradation abstractions in the TSPA code, the input information includes the design description of the waste package, the waste form, the waste package internals, and in-package water chemistry and temperature. The output from this section includes waste form mobilization rates to assess engineered barrier system radionuclide transport.

SAR Sections 2.3.7.1–2.3.7.4, 2.3.7.6–2.3.7.9, 2.4, and associated references summarized the applicant's model abstractions and related FEPs related to the degradation of commercial SNF and cladding, high-level waste glass, and DOE SNF (including naval SNF).

Inventory of Radionuclides and Radionuclide Distribution in Commercial SNF

The applicant provided inventory data in SAR Section 1.5.1, in terms of weight, volume, and package design, for each waste form. The NRC staff's review of the inventory data is discussed in SER Sections 2.1.1.2.3.4.1, 2.1.1.2.3.4.2, and 2.1.1.2.3.5.1. The applicant states that more than 100 radionuclides may be collectively present in the waste package at the time of repository closure. Among them, a total of 32 isotopes of 18 elements were selected by the applicant as important radionuclides to potential dose for scenario classes involving groundwater transport (SAR Table 2.3.7-2).

NRC Staff's Review

The NRC staff's evaluation of the radionuclide inventory incorporated the applicant's design features of the waste forms in the waste package (SAR Section 1.5.1). The design features include thermal loading, structural characteristics, radionuclide inventory, chemical composition, and microstructural characteristics. The NRC staff independently evaluated (i) the total mass of the waste form, and (ii) the long half-lived radionuclides that need to be considered in the repository, as summarized in NRC/Center for Nuclear Waste Regulatory Analyses (CNWRA®) reports (Leslie, et al., 2007aa; Jain, et al., 2004aa; Manaktala, 1993aa). In these independent evaluations, the NRC staff reviewed open literature information. The NRC staff finds acceptable the total mass inventory and 32 isotopes of 18 elements of the long-lived radionuclides the applicant used because this information is adequately supported by data and models, and underestimation of dose exposure will not occur.

The NRC staff verified the applicant's radionuclide inventory calculations by comparing the results with other published results and with independent calculated inventory histories for times from 1 year to 1 million years (Leslie, et al., 2007aa). The trends in the inventories were compared with those published elsewhere (e.g., Roxburgh, 1987aa). Most radionuclide inventories decrease with increasing times, some remain relatively constant over long periods of time (those with long half-lives), and others increase with time (daughters in a decay chain). The NRC staff selected 43 radionuclides from hundreds of radionuclides present in commercial SNF. Screening criteria included half-lives, solubilities, and radiotoxicities of the radionuclides. Similar evaluations were performed for high-level waste glass and DOE (and naval) SNF.

Most radionuclides, and essentially all of the rare earth and actinide radionuclides (e.g., plutonium isotopes), are retained in the UO_2 matrix. Transition metals and fission products (e.g., technetium) are partly partitioned into metallic phases embedded in the matrix, SNF grain boundaries, and gap region (i.e., the interface between the pellets and the cladding). The NRC staff independently evaluated the distribution of radionuclides in the matrix and the accumulated radionuclides in the gap and grain boundaries. The NRC staff relied on open literature information and calculated radionuclide concentrations as a function of time, as earlier summarized in NRC/CNWRA reports (Leslie, et al., 2007aa; Jain, et al., 2004aa; Manaktala, 1993aa). The measured gap and grain boundaries in open literature include work by Bremier, et al. (2000aa); Gray (1999aa); Johnson and Tait (1997aa); and Lassmann, et al. (1995aa); and the changes in the radionuclide inventories were calculated using ORIGEN-ARP 2.00 (Bowman and Leal, 2000aa).

The NRC staff finds acceptable the applicant's evaluation of the radionuclide distribution in the SNF matrix because the inventory was based on experimental data and standard analytical techniques the industry uses to establish the distribution of actinides, transition metals, and fission products, and is consistent with values documented in the open literature.

Degradation of Commercial SNF

The applicant reported that if the waste package and cladding are breached, oxidation and dissolution of the commercial SNF matrix may occur. If the temperature exceeds approximately 100 °C [212 ºF], solid-state oxidation or hydration will occur, depending on the relative humidity. Commercial SNF dissolves by oxidative reaction of the UO_2 matrix in humid air or in solution at temperatures less than approximately 100 °C [212 ºF]. Oxidation and hydration occur faster than dissolution. Oxidation, hydration, and dissolution can be preferentially enhanced along grain boundaries. In the applicant's commercial SNF degradation evaluation (BSC, 2004ah), the high end of the dissolution rate range was obtained from tests in fast-flowing carbonate solutions. The low end of the dissolution rate range was obtained from commercial SNF rod segment tests under dripping groundwater conditions with precipitates deposited and failed cladding present. Un-irradiated UO_2 was also tested because there is no significant difference between the dissolution rates of un-irradiated UO_2 and commercial SNF under air-saturated groundwater conditions. Data from long-term immersion and dripping water tests up to 8.7 years in duration were included in the evaluation.

On the basis of these data analyses, the applicant presented quantitative models and model parameter values for (i) the instantaneous release of radionuclides from the gap and grain boundaries and (ii) matrix dissolution inducing slow long-term radionuclide releases (SAR Sections 2.3.7.7, 2.4.2.2, and 2.4.2.3). Mean fractional matrix dissolution rates were 5×10^{-4} to 6×10^{-3} year^{-1} at pH of 5.5–8.0 and a temperature range of 25–90 °C [77–194 °F] under wet conditions, according to DOE (2009an, Enclosure 5). Fission products and activation products were released with the matrix dissolution. Actinide releases may be controlled by solubility limits of dissolved radionuclides and also may be affected by colloids. The applicant addressed uncertainties of its models and parameter values.

In the applicant's TSPA model, high-solubility fission products and activation products (e.g., I-129 and Tc-99) are released from the waste form at rates controlled by (i) the decay of radionuclide inventory of each waste package and (ii) waste package failure rate (e.g., SAR Section 2.4.2.2.3.2.1). The waste package failure rate is related in series to waste form dissolution rate and radionuclide diffusion rate. The slowest rate among the three controls the rate of release. The applicant supported its model by stating that the dissolution rates of waste forms, including commercial SNF, are sufficiently faster (e.g., hundreds to a few thousand years) than the time intervals of each waste package failure. Low-solubility radionuclides (e.g., plutonium isotopes) are released at rates controlled mainly by the concentration limits of dissolved or colloidal species.

NRC Staff's Review of the Initial Condition of SNF at Receipt

The NRC staff finds that the model abstraction of the degradation of commercial SNF properly incorporates the applicant's waste form design features (which include thermal loading, structural characteristics, radionuclide inventory, chemical composition, and microstructural characteristics; SAR Section 1.5.1), in that the commercial SNF conditions at receipt will not be altered during transportation and interim storage, as outlined in DOE (2009ax, Enclosure 3). The NRC staff finds acceptable the applicant's assumption that the pressure of the residual water vapor inside the transportation, aging, and disposal canister would not be sufficiently high to degrade the commercial SNF matrix by grain boundary hydration. The NRC staff's review and conclusion are based on the applicant's use of standard vacuum drying procedures in packaging waste. The vacuum drying lowers the residual water vapor, resulting in no matrix degradation by grain boundary hydration. In addition, the NRC staff finds from the NRC staff's literature review that potential matrix disintegration by high burnup in the range of ~60–65 MWD/kgU (e.g., Finch, et al., 1999aa; NRC, 2008aa) is not likely (Spino, et al., 2003aa). Also, in the TSPA model, release out of the waste package to the invert from SNF dissolution is insensitive to the initial condition. The release is mainly controlled by either (i) the radionuclide inventory of each waste package and waste package failure rate or (ii) concentration limits. Therefore, the NRC staff finds acceptable the applicant's technical arguments that the pressure of residual water vapor inside the transportation, aging, and disposal canister would not be sufficiently high to disintegrate the commercial SNF matrix by grain boundary hydration, and the residual water vapor will not cause the release of radionuclides out of the waste package to the invert from SNF dissolution to be underestimated.

NRC Staff's Review of Releases from the Matrix and from the Gap and Grain Boundaries

The NRC staff evaluated the applicant's processes and modeling for matrix dissolution and radionuclide release from the gap and grain boundaries. The NRC staff reviewed open literature information for this evaluation, including results from the NRC staff's independent modeling (NRC, 2008aa; Leslie, et al., 2007aa; Jain, et al., 2004aa). The NRC staff finds acceptable the applicant's conclusion that the release of high-solubility radionuclides (e.g., I–129 and Tc–99) will be at the same rate of oxidative UO_2 matrix dissolution, whereas release of low-solubility radionuclides (e.g., plutonium isotopes) may be limited by solubility. These findings are consistent with laboratory test results (Wilson and Gray, 1990aa).

The UO_2 matrix would dissolve in the oxidizing environment expected at the proposed Yucca Mountain repository (Shoesmith, 2000aa). This is consistent with the alteration process of natural analog uraninite (BSC, 2004ah). The applicant's mathematical models are empirical. The applicant identified important environmental and commercial SNF parameters controlling the dissolution rate. Those parameters included oxygen partial pressure, carbonate concentration, temperature, pH, and the surface area of the matrix contacted by water. These parameter values were obtained from accelerated test results in oxidizing environments, with consideration of data uncertainties. The NRC staff finds these parameters and their values acceptable because they were derived from appropriate tests that are based on electrochemical, flow-through, and round-robin tests. In addition, the NRC staff performed independent evaluations (Leslie, et al., 2007aa) that confirmed the applicant's conclusions. The NRC staff also finds acceptable the applicant's assumption that radionuclide release from the gap and grain boundaries is rapid because these radionuclides are not atomically bound in the matrix. Therefore, the NRC staff finds that the applicant conservatively presented the conceptual and mathematical models for radionuclide releases that are faster than those expected to occur from matrix degradation and from the gap and grain boundaries.

NRC Staff's Review of Environmental Conditions inside a Waste Package

On the basis of the applicant's in-package chemistry models (BSC, 2005ad) and the NRC staff's independent analyses (Leslie, et al., 2007aa; NRC, 2008aa, 1996ab), the NRC staff finds that the applicant accounted for the range of environmental conditions expected inside the breached waste packages. The applicant based the commercial SNF degradation model on pure carbonate solutions in the concentration range of 2×10^{-4} to 2×10^{-2} mol/L. The NRC staff finds that this is conservative because carbonate solutions lead to faster matrix dissolution in a waste package (NRC, 2008aa). The NRC staff's independent review and analysis of open literature information (Leslie, et al., 2007aa; NRC, 2008aa, 1996ab) suggest that dissolved constituents, such as calcium and silica, may reduce the dissolution rate. Slow-dripping groundwater, partial protection by failed cladding, and iron corrosion products may also reduce the dissolution rate. As explained in DOE (2009ax, Enclosure 16), fast SNF dissolution in pure carbonate solutions may cause more actinide release associated with colloids. The NRC staff reviewed the applicant's derived dissolution processes of SNF in terms of the rates of dissolution and colloid formation of actinides, such as plutonium. On the basis of the applicant's conservative approach using carbonate solutions, the NRC staff's independent review of dissolution data in the open literature (NRC, 2008aa, 1996ab) and the TSPA's dose consequence, the magnitude of this increased actinide release is not significant because realistic matrix dissolution rates are lower. In the TSPA model, however, for high-solubility radionuclides, such as technetium, the release rate is controlled by the radionuclide inventory of each waste package and the waste package failure rate. In this rate-controlled TSPA model, the slow realistic dissolution rates may result in increased release rates because multiple waste package failures at low dissolution rates may have a greater contribution to the release. The NRC staff assessed the effects of multiple waste package failures and finds that the magnitude of the increase in release is minimal and insignificant. To reach this conclusion, the NRC staff compared the release rates at different dissolution rates per unit area, assuming spherical SNF particles from single- or multiple-remnant-failed waste packages at slow, realistic dissolution rates a factor of 10 lower (NRC, 2008aa). The release rate from aggregated multiple slower dissolution rates is higher than the release rate from a single faster dissolution rate because the release rate decreases as the particle size shrinks (Poinssot, et al., 2005aa).

NRC Staff's Review of Alternative Models for Matrix Dissolution

In BSC (2004ah, Section 6.4.2), the applicant presented an electrochemical model and a surface complexation model as alternative models for the dissolution of the commercial SNF. The electrochemical model describes the process of the matrix dissolution by electric current flow under oxidizing conditions, and the surface complexation model describes the dissolution process by carbonate complexation. On the basis of its review, the NRC staff finds that the applicant's model results for dissolution rate are consistent with the base case results within the uncertainty limits and are, therefore, acceptable. In addition, the NRC staff performed independent assessments of the processes involved in these two alternative models by reviewing open literature information (NRC, 2008aa, 1996ab). On the basis of its literature review, the NRC staff finds acceptable the applicant's basis and justification for the alternative models. The NRC staff finds that the applicant's results are consistent with the NRC staff's independent evaluations (NRC, 2008aa, 1996ab).

Degradation of High-Level Waste Glass

The applicant conceptually modeled high-level waste glass as being congruently dissolved for glass constituent elements and radionuclides at relative humidity greater than or equal to 44 percent (SAR Section 2.3.7.9). At lower relative humidity, the glass dissolution rate is set to zero. Dissolution kinetics were considered to be chemically controlled by dissolved orthosilicic acid (H_4SiO_4). As glass reacts with solution and reaches saturation with respect to mineral phases, precipitation occurs on the glass surface. The applicant's model was supported by dissolution studies conducted with a wide range of borosilicate glass compositions under various environmental conditions. On the basis of the data available from the applicant's tests, the applicant presented a quantitative model and model parameter values for the high-level waste glass dissolution process.

NRC Staff's Review

The NRC staff finds acceptable the model abstraction of the degradation of high-level waste glass because it incorporates the applicant's waste form design features, which include thermal loading, structural characteristics, radionuclide inventory, chemical composition, and microstructural characteristics (SAR Section 1.5.1). The NRC staff evaluated the processes and modeling that the applicant presented for the dissolution of high-level waste glass. The NRC staff's evaluation was based on open literature information originally compiled by Leslie, et al. (2007aa).

The NRC staff finds acceptable the applicant's assumption that the release of high-solubility radionuclides (e.g., I-129 and Tc-99) will be at the same rate of matrix dissolution, especially at repository relevant pH 5–8 in diluted in-package water chemistry at lower temperatures of 25–90 °C [77–194 °F] after waste package failure (BSC, 2005ad; Leslie, et al., 2007aa). The NRC staff notes that the basis for this assumption is that higher solubility radionuclides will not tend to form immobile precipitates. As addressed in SER Section 2.2.1.3.4.3.1, the applicant's assessment of this environmental condition (BSC, 2005ad) is acceptable. The applicant considered both sodium- and calcium-based pore waters and data on the matrix dissolution under immersion, dripping groundwater, and vapor conditions. The data from these conditions were consistent with each other, and the applicant also used some accelerated tests such as fast-flowing water tests to determine model parameter values. The NRC staff finds that the condition of mild aqueous chemistry used in the assessment would persist for the dissolution because the waste package would fail after several ten (for stress corrosion cracking) to hundred (for general corrosion) thousand years. The NRC staff also finds the applicant's assertion that release of low-solubility radionuclides (e.g., plutonium isotopes) may be limited by solubility to be appropriate. These two release modes and models are consistent with laboratory data presented by the applicant and in the literature for borosilicate high-level waste glass in simulated Yucca Mountain in-package water, as described in BSC (2005ad), under immersion, dripping groundwater, and vapor conditions.

The NRC staff concludes that the applicant appropriately included in mathematical models important environmental and high-level waste glass parameters controlling the dissolution rate. The NRC staff finds the mathematical models acceptable because they are consistent with an independent assessment of the dissolution rate based on open literature data (Leslie, et al., 2007aa). The applicant quantitatively modeled the release rate of radionuclides as a function of surface area of glass contacted by water, intrinsic glass dissolution rate (i.e., release rate of boron as an indicator), pH, activation energy for temperature dependence, and the extent of orthosilicic acid saturation in solution with the glass. The dissolution rates in acidic and alkaline regimes of pH were separately modeled. At 100–250 °C [212–482 °F], a fixed pH was used. The mean fractional dissolution rates are 2×10^{-5} to 4×10^{-3} year^{-1} at pH of 5.5–8.0 and temperature of 25–90 °C [77–194 °F] under wet conditions, according to DOE (2009an, Enclosure 5).

The NRC staff finds that the applicant obtained model parameters and data uncertainties from sufficient and suitable data, as documented in its application (SAR Section 2.3.7.9). The equation used by the applicant to calculate the area of glass surface contacted by water as glass dissolves accounts for (i) an increase in surface area from thermal and mechanical cracking, water access and reactivity with water and (ii) a loss in the surface area due to dissolution, as detailed in DOE (2009ax, Enclosure 2) and DOE (2009cz, Enclosure 2). The increased surface area leads to increased release of radionuclides out of the waste package to the invert from high-level waste glass dissolution. In the applicant's model, this increasing factor of surface area is expressed as “exposure factor.” The NRC staff concludes that model or data uncertainties for other processes were appropriately discussed with respect to the distributions of model parameters (e.g., the extent of orthosilicic acid saturation) and conservatisms. The applicant’s uncertainty assessment is acceptable to the NRC staff because the uncertainties of parameter values did not significantly affect the radionuclide release out of the waste package. Model support was also discussed with respect to long-term field tests and natural analog basalt glass. The NRC staff considers basalt glass to be an appropriate analog to high-level waste glass because of similarities in composition, reaction conditions, and transformation of the glass matrix during alteration into a range of minerals. Jantzen, et al. (2008aa) reviewed and analyzed existing field tests of high-level waste glass buried for approximately 24 years; this review further supports the license application model and data for high-level waste glass dissolution. Jantzen, et al. (2008aa) showed superior or equivalent performance of this burial glass in unsaturated and saturated sediments, compared to saturated accelerated laboratory tests that are the basis for the applicant’s model.

Literature data and discussions on cracking (or pitting) of various glasses in more aggressive solutions (Pulvirenti, et al., 2006aa; Morgenstein, et al., 1999aa) did not show significant increase in dissolution rates. The test solutions used by Pulvirenti, et al. (2006aa) and Morgenstein, et al. (1999aa) were aggressive, unlike those expected in the repository, and many glasses considered were not based on borosilicates intended for disposal in the proposed repository. The applicant also assessed the volume occupied by porosity in the altered layer during the glass hydration. The porosity may increase the glass volume in a confined canister, which may create stress that further fractures the glass. The calculated porosity volume was close to the elemental mass percentage of soluble elements in the glass. Therefore, the applicant considered that isovolumetric hydration would occur. For high-solubility radionuclides in the TSPA, the release rate is controlled by the radionuclide inventory of each waste package and waste package failure rate because the dissolution rate is faster, as discussed for the degradation of commercial SNF. The NRC staff finds that the increased dissolution (even with further cracking by any means) associated with data or model uncertainties would not be rate controlling in the release. The waste form dissolution rate is related in series to the waste package failure rate and the radionuclide diffusion rate. The slowest among the three rates, the waste package failure rate in this case, controls the release rate.

Therefore, the NRC staff finds acceptable the applicant’s conceptual and mathematical models for radionuclide releases from high-level waste glass degradation. The NRC staff also finds acceptable the applicant’s accounting of the range of environmental conditions expected inside breached waste packages because the applicant’s in-package water chemistry models (BSC, 2005ad) were developed using current analytic techniques and thermodynamic approaches, acceptable in standard technological practice. In addition, the NRC staff’s independent models (Leslie, et al., 2007aa) predicted similar conditions considering immersion, dripping groundwater, and vapor environments.

As an alternative model for high-level waste glass dissolution, the applicant presented a dissolution model without considering the extent of orthosilicic acid saturation (i.e., affinity) (BSC, 2004ai). On the basis of its review of open literature information (e.g., Leslie, et al., 2007aa; BSC, 2004ai), the NRC staff finds that the alternative model is conservative and consistent with available data/analyses and current scientific understanding.

Degradation of DOE and Naval SNF and Cladding

The applicant divided its SNF into 34 distinct waste forms. Except for naval SNF, these DOE SNF types were modeled as degrading instantaneously upon waste package breach. Commercial SNF waste packages were used to represent the naval SNF waste packages for all scenario classes because radionuclide release rates from the naval SNF waste packages were predicted to be considerably lower than from commercial SNF waste packages (BSC, 2004ao). Data uncertainties were discussed with respect to the conservatisms used.

NRC Staff's Review

The NRC staff finds acceptable the applicant's assumption that DOE's SNF will degrade instantaneously because it would not underestimate the radiological consequences. The applicant's approach to model the naval SNF as commercial SNF is also acceptable because this modeling assumption would not underestimate the radiological consequences. The naval SNF is more robust and would release less radionuclides (BSC, 2004ao). This is also supported by the TSPA model. For high-solubility radionuclides, the release rate is controlled mainly by the inventory of each waste package and waste package failure rate. For low-solubility radionuclides, the release rate is controlled mainly by the concentration limits of dissolved species or colloids. The TSPA abstraction provides results consistent with output from detailed process-level models and/or empirical observations of DOE SNF characteristics (DOE, 2003ad).

The assumption of instantaneous degradation for DOE's SNF may result in greater and faster colloidal release in a shorter period, a possibility addressed in DOE (2009ax, Enclosure 1) and DOE (2009cz, Enclosure 1). This is similar to the case of instantaneous degradation of the waste form under igneous intrusive scenarios as described in the following subsection (Other Model and Data Uncertainties – Waste Form Degradation under Disruptive Scenarios). More realistically, after the instantaneous degradation, the waste form would be altered into other solid forms, such as further oxidized UO_2, other oxide metal compounds, or hydrolysis or precipitation products (SNL, 2008ak, 2007ag), before being slowly dissolved, as described in DOE (2009cz, Enclosure 1). The applicant considered colloid formation during the instantaneous degradation and subsequent alteration of the waste form. The NRC staff finds acceptable the model support for the quantitative information on colloid concentrations during these processes. As outlined in DOE (2009ax, Enclosure 1), DOE (2009cz, Enclosure 1), and DOE (2009db, Enclosure 1), the applicant showed that faster dissolution does not necessarily result in greater plutonium colloid concentration in the bulk solution because the majority of the plutonium remains in a residue (altered) layer at the reaction front.

The applicant assumed that the zircaloy and stainless steel commercial SNF cladding failed upon emplacement. Therefore, degradation of commercial SNF cladding was not included in the TSPA analysis. The effect of the naval SNF structure on the release and transport of radionuclides was treated separately from other DOE SNF types in the assessment. DOE SNF was conservatively assumed to degrade instantaneously. The naval SNF degrades more slowly than the commercial SNF, and therefore the naval SNF can be represented by commercial SNF waste packages in the TSPA analysis. The applicant documented the technical basis for cladding behavior in the repository only at a high level because the applicant assumed that the zircaloy and stainless steel commercial SNF cladding failed at time of emplacement. The NRC staff finds this assumption conservative because fuel cladding may persist after emplacement.

Other Model and Data Uncertainties

Waste Form Degradation under Disruptive Scenarios

The applicant stated that under the seismic scenario class, stress corrosion cracking of the waste package would occur earlier than waste package failure in the nominal case. Seismic response (motion and rockfall) could damage the drip shield and waste package, resulting in earlier stress-corrosion cracking. The waste form dissolves in the diffused-in water vapor through the stress-corrosion cracks, as in the nominal case. The igneous scenario class includes eruptive and intrusive events. In the volcanic eruption case, the impacted waste form was transported to the surface; this case does not involve groundwater transport and is not discussed in this SER Section. In the igneous intrusive case, the waste form was assumed to be rapidly altered at expected elevated temperatures and made available to groundwater.

NRC Staff's Review

The NRC staff reviewed the degradation models implemented in the TSPA code abstraction to confirm that they provide consistent results with the output from the detailed process-level models and/or empirical observations on the characteristics of commercial SNF and high-level waste glass, as described in this SER section. The applicant presented a bounding assumption regarding radionuclide release from all waste forms under igneous intrusive conditions in SAR Section 2.3.11.3.2.4. The applicant assumed that all waste forms instantaneously degrade to be mobilized for release. The NRC staff finds that these instantaneous degradation models are conservative and bounding in terms of soluble radionuclide release. As discussed earlier, in the TSPA model the release from the failed waste package is insensitive to the fast rate of waste form degradation. The release rate is controlled mainly by the inventory of each waste package and waste package failure rate for high-solubility radionuclides and concentration limits of dissolved species or colloids for low-solubility radionuclides. The assumption of instantaneous degradation under igneous intrusive conditions may result in more and faster colloidal release in a shorter period, as addressed in DOE (2009ax, Enclosure 1) and DOE (2009cz, Enclosure 1). For the nominal scenario case, the applicant presented concentrations of 1.2×10^{-8} to 6.2×10^{-4} g/L [1.2×10^{-5} to 6.2×10^{-1} ppm (part per million)] for irreversible colloids from commercial SNF, as detailed in DOE (2009ax, Enclosure 16), and 2.7×10^{-6} to 1.4×10^{-5} g/L [2.7×10^{-3} to 1.4×10^{-2} ppm] for high-level waste glass colloids (SAR Section 2.3.7.11.3). The NRC staff finds the applicant's instantaneous degradation assumption conservative because the waste form can be altered into other solid forms such as oxidized UO_2, other oxide metal compounds, or hydrolysis or precipitation products (SNL, 2008ak, 2007ag) before being slowly dissolved, as detailed in DOE (2009cz, Enclosure 1).

The applicant considered colloid formation during the instantaneous degradation and subsequent alteration of the waste form. The NRC staff finds acceptable the model support for the quantitative information on colloid concentrations during these processes. In DOE (2009ax, Enclosure 1), DOE (2009cz, Enclosure 1), and DOE (2009db, Enclosure 1), the applicant showed that faster dissolution does not necessarily result in greater plutonium colloid concentration in the bulk solution, because the majority of the plutonium remains in a residue (altered) layer at the reaction front.

Microbial Effects

The applicant did not specifically address the microbial effects potentially affecting the dissolution of SNF and high-level waste glass. However, the applicant's models indirectly incorporate these effects to the extent that the models are consistent with natural analog and field test results.

NRC Staff's Review

The NRC staff reviewed the waste form degradation that was partially tested in solutions under projected repository conditions, along with field tests and analog studies. The NRC staff finds that the analog data or field test results could have been affected by the potential presence of microbial effects, compared with the laboratory test results. Aqueous environments, especially field test or analog environments, may have contained organic byproducts or microbes. The applicant's models also include the characteristics of natural analogs of the waste form or field test results. No indication of microbe effects (e.g., lowering pH) were reported from these literature data (BSC, 2004ah, 2004ai). Therefore, the NRC staff finds the applicant's approach to the effects of organic byproducts or microbes on the waste form degradation acceptable. Screening arguments are also found in excluded FEP 2.1.02.10.0A, Organic/Cellulosic Materials in Waste (SNL, 2008ab); although excluded FEPs are not explicitly discussed in this SER Section. The NRC staff's evaluation of the technical basis for the applicant's exclusion of FEPs is in SER Section 2.2.1.2.1.3.2.

Integration in the Engineered Barrier System Radionuclide Transport Abstraction

The applicant's waste form mobilization abstraction provides radionuclide inventory, mobilized radionuclides, and waste form colloids to the engineered barrier system radionuclide transport model. The NRC staff finds that the applicant has appropriately deployed the waste form mobilization using the GoldSim® computer code (2006aa) in the waste form domain. The NRC staff confirmed that the GoldSim results are consistent with the applicant's description and quantitative assessments of waste form lifetimes in the waste form domain.

Summary of NRC Staff's Review of Waste Form Degradation

The NRC staff finds that the applicant appropriately incorporated design features of the waste form. The design features include thermal loading, structural characteristics, radionuclide inventory, chemical composition, and microstructural characteristics. The NRC staff reviewed the applicant's proposed dissolution processes of waste forms and finds the following applicant conclusions acceptable: (i) the dissolution process of commercial SNF is based on oxidative dissolution of the UO_2 matrix and (ii) the dissolution process of high-level waste glass is based on orthosilicic acid release. The NRC staff also finds that the models for DOE SNF dissolution are bounded by the assumption of instantaneous dissolution and the models for naval SNF are conservatively assumed to be models for commercial SNF. In the models, the applicant reasonably includes important environmental and waste form parameters controlling the dissolution rate (e.g., pH, temperature, solution chemistry, cracking, and hydration). The NRC staff finds that the applicant's models are appropriately and objectively supported by laboratory and/or field test results. The applicant obtained sufficient data to vary environmental and waste form parameters. The model uncertainties were appropriately assessed with alternative models; the data uncertainties were appropriately assessed with sufficient data obtained under various environmental and waste form conditions. The TSPA model considers uncertainties associated with (i) conservative, fast dissolution of commercial SNF under nominal conditions and (ii) the instantaneous dissolution of all waste forms under igneous conditions and DOE SNF under nominal conditions. The NRC staff finds that the applicant properly implemented its TSPA code with the model abstraction of radionuclide mobilization from the waste form.

2.2.1.3.4.3.3 Concentration Limits

This section discusses the NRC staff's review of the applicant's abstraction and TSPA model implementation of dissolved radioactive element (radioelement) concentration limits, as described in SAR Section 2.3.7.10. In the performance assessment model, concentration limits exert strong controls on the concentrations in water of dose-important radioelements—particularly neptunium and plutonium—and, thus, on the release rates of those elements' isotopes from the engineered barrier system. These limits are based on chemical equilibrium relationships between the dissolved element and solid substances containing the element. The abstraction calculates, on the basis of water chemistry, maximum concentrations that limit how much of the total mass of a radioelement may remain dissolved in solution in the waste form domain, the corrosion products domain, and the invert (the drift floor, consisting of crushed rock). In the waste form domain, the radionuclide concentration calculated from the waste form degradation abstraction is reduced if it, along with the concentrations of other isotopes of the same element, exceeds the calculated concentration limit. The concentration limit comparison is also implemented for the radionuclide transport abstraction in the corrosion products domain and the invert. In each case, the remaining radionuclide mass is retained in the domain as a precipitated mass that is available for re-dissolution whenever the concentration is below the concentration limit. The inputs to the concentration limits abstraction are geochemical characteristics in the domain of the water from the in-package chemistry abstraction (waste form domain; SER Section 2.2.1.3.4.3.1) or the engineered barrier system radionuclide transport abstraction (corrosion products domain and invert; SER Section 2.2.1.3.4.3.5) and gas from the engineered barrier system chemical environment abstraction (SER Section 2.2.1.3.3). The outputs from the TSPA model abstraction are the concentration limits used in the engineered barrier system radionuclide transport abstraction (SER Section 2.2.1.3.4.3.5). The actual application of the concentration limits and retention of the precipitated mass is calculated in the GoldSim computer code, as outlined in GoldSim Technology Group (2006aa, pp. 253–255). Radionuclide Solubility, Solubility Limits, and Speciation in the Waste Form and Engineered Barrier System is an included FEP encompassing the abstraction evaluated in this section. The related excluded FEPs are addressed in SER Section 2.2.1.2.1.3.2.

To evaluate the applicant's abstractions of radioelement concentration limits, the NRC staff reviewed SAR Section 2.3.7, the analysis model report on concentration limits (SNL, 2007ah), and the applicant's responses to the NRC staff's requests for additional information (DOE, 2010aj; 2009ax,ay,cz,da,db,dc). The NRC staff also relied on the technical literature on solubility limits and the application of solubility limits in performance assessments, the NRC staff's independent solubility limit evaluations (e.g., Murphy and Codell, 1999aa; Mohanty, et al., 2003aa), and the NRC staff's independent laboratory studies (e.g., Prikryl, 2008aa).

Overall Abstraction Approach

The applicant's abstraction for concentration limits calculates concentration limits for plutonium, neptunium, uranium, thorium, americium, tin, and protactinium using lookup tables (SNL, 2007ah) that define values (in mg/L, a unit that is approximately equivalent to parts per million, or ppm) as functions of pH and fCO_2 (i.e., CO_2 fugacity). For radium, the value is specified as a constant that depends on the range in which the pH value falls. For technetium, carbon, iodine, cesium, strontium, selenium, and chlorine, no concentration limit is applied; this abstraction, therefore, does not affect their release rates from the engineered barrier system.

For plutonium, neptunium, uranium, thorium, americium, tin, and protactinium, the applicant determined the concentration limit value from the lookup table for each timestep in a realization. Uncertainty is incorporated into the abstraction by sampling, for each realization, two uncertainty terms that are then added to (or subtracted from) the value derived from the lookup table. No further uncertainty is applied to the determined radium value. Concentration limits are treated the same for nominal and disruptive events, with the exception of the uranium abstraction in the igneous intrusive case (see uranium discussion in the Concentration Limits Parameters section).

NRC Staff's Review

The NRC staff finds acceptable the applicant's overall approach to impose concentration limits because it is consistent with standard thermodynamic geochemical principles and uses consistent and appropriate assumptions. The use of pure-phase solubilities to constrain radioelement concentrations at the source is an accepted approach in performance assessments for radioactive waste disposal (e.g., Nuclear Energy Agency, 1997aa; Leslie, et al., 2007aa). When faced with uncertainty regarding the appropriate solid phase to model a solubility limit, the applicant appropriately chose the solid phase that would result in higher dissolved concentrations (e.g., hydrated PuO_2 instead of anhydrous PuO_2).

Chemical Environment for Concentration Limits

For the waste form domain, the applicant's in-package chemistry abstraction provides pH, ionic strength, and fluoride concentration to the concentration limits abstraction at each timestep, and CO_2 fugacity (fCO_2) is obtained from the engineered barrier system chemical environment abstraction. The same ionic strength and fCO_2 are used in the corrosion products domain, but pH is calculated from a formula that involves fCO_2 and the aqueous uranium concentration; this pH abstraction is based on the competitive surface complexation model discussed in SER Section 2.2.1.3.4.3.5. In the corrosion products domain, according to DOE (2009ay, Enclosure 3) and DOE (2009da, Enclosure 2), the abundant mass of products of stainless steel corrosion controls pH to a relatively narrow range of 7.0 to 8.4. For the invert, when there is no flow from the waste package, pH, ionic strength, and fCO_2 are obtained from the engineered barrier system chemical environment abstraction; when there is advective flow out of the waste package, according to DOE (2009ax, Enclosure 7), pH and ionic strength in the invert are the same as in the corrosion products domain.

NRC Staff's Review

The NRC staff finds that the applicant used appropriate tools to model concentration limits, including the important geochemical parameter inputs (e.g., pH and fCO_2) affecting the solubility model outputs. Concentration limits in the waste form domain are functions of chemical parameters developed by the in-package chemistry model (SER Section 2.2.1.3.4.3.1). Chemical conditions for concentration limits in the corrosion products domain were properly modeled using the surface complexation model, and the results of that model were supported by comparison with the applicant's independent modeling efforts in DOE (2009da, Enclosure 2).

An abundant secondary phase, such as steel corrosion products, can have an important influence on pH buffering in an environment with such high solid-to-water ratios. In DOE (2009db, Enclosure 2), the applicant provided information that (i) supported its selection of the uncertainty range for the stainless corrosion rate on the basis of laboratory data and (ii) showed that plutonium isotope release rates, which are sensitive to the pH-dependent plutonium solubility limit, are insensitive to the stainless steel corrosion rate. The NRC staff evaluated the applicant's analysis by reviewing corrosion rate data in the literature (Beavers and Durr, 1990aa; BSC, 2004ae; Glass, et al., 1984aa; McCright, et al., 1987aa). In addition, the applicant showed in DOE (2009db, Enclosure 3) that conservatisms in (i) the treatment of the timing of radionuclide release after waste package breach, (ii) assumptions regarding flow within the waste package, and (iii) the lower pH range meant that the applicant did not overestimate the effectiveness of stainless steel corrosion products in controlling pH. The NRC staff finds that this conclusion is appropriate, in that the abstraction would not result in underestimation of radionuclide release rates, and is consistent with the NRC staff's understanding of the abstraction.

On the basis of this review, the NRC staff finds acceptable the applicant's TSPA integration of the concentration limit abstraction and its accounting for the range of geochemical environments expected in the waste package and invert.

Concentration Limits Parameters

The applicant calculated concentration limits for plutonium, neptunium, uranium, thorium, americium, tin, protactinium, and radium (using barium as an analog) assuming pure-phase solubility at equilibrium with solution using the geochemical modeling code EQ3NR. The solubility models were conducted at a range of pH, fCO_2 (CO_2 fugacity), and fluoride values. For plutonium, neptunium, uranium, thorium, americium, tin, and protactinium, the pH and fCO_2 dependencies were incorporated into the lookup tables, and the fluoride sensitivity was applied through an uncertainty term. SAR Section 2.3.7.10 described the technical bases for the limiting minerals selected for the solubility models. For most of the modeled elements, equilibrium with atmospheric oxygen was assumed; this assumption was modified for plutonium in the waste package. As mentioned previously, the applicant did not apply a concentration limit for technetium, carbon, iodine, cesium, strontium, selenium, and chlorine.

NRC Staff's Review

The NRC staff finds acceptable the applicant's use of no concentration limits for technetium, carbon, iodine, cesium, strontium, selenium, and chlorine. This approach is consistent with the lack of a strong technical basis for concentration limits and is conservative because it does not underestimate the radiological consequences of release of isotopes of these seven elements. Application of a concentration limit can only reduce the dissolved concentration of an element; therefore, ignoring this process can only increase radionuclide release rates or leave them unchanged.

The NRC staff finds acceptable the applicant's use, in its chemical speciation and solubility models, of appropriate thermodynamic data that had been evaluated for this purpose and were based on extensive, scholarly, international reviews. More detailed concentration limit evaluations for each element are in the following paragraphs.

Plutonium

On the basis of the applicant's TSPA code dose modeling results, plutonium is a risk-significant radioelement. The applicant's plutonium concentration limits abstraction is based on equilibrium geochemical modeling but differs from other abstractions in that equilibrium with atmospheric oxygen was not assumed. Atmospheric oxygen would impose higher redox potentials that tend to lead to higher calculated dissolved plutonium concentrations. The applicant used an adjusted-Eh model, which assumes lower Eh than would be imposed by atmospheric oxygen (Eh is a measure of a solution's oxidation potential, which may be described as the tendency of the solution to convert dissolved elements to higher oxidation states). The adopted Eh-pH relationship for plutonium solubility models is more oxidizing than the line bounding a compilation of data from waters in contact with the atmosphere, as outlined in SNL (2007ah), DOE (2009ax, Enclosure 9), and DOE (2009cz, Enclosure 7). For the typical pH and CO_2 conditions in the waste package (SAR Figures 2.3.7-19 to 2.3.7-21), the sampled plutonium solubility limit in the TSPA model ranges from approximately 0.006 to 0.5 mg/L [0.006 to 0.5 ppm] (SAR Figure 2.3.7-29); uncertainty terms extend this range as much as ± two orders of magnitude (e.g., SAR Figure 2.3.7-38).

NRC Staff's Review

The NRC staff's review focused on the applicant's adjusted-Eh model, which assumes lower Eh than would be imposed by atmospheric oxygen. The NRC staff finds, on the basis of its review of information the applicant provided and on an independent NRC staff review of the scientific literature (e.g., Rai, et al., 1999aa; Neck, et al., 2007aa), that the plutonium solubility abstraction was properly developed using Eh–pH conditions that bound the in-package environment. As illustrated in SNL (2007ah), DOE (2009ax, Enclosures 8 and 9), DOE (2009cz, Enclosures 5 and 6), and DOE (2010aj), nearly all compared plutonium experimental data lie within two standard deviations of the uncertainty in the pH-dependent solubility relationship. The NRC staff finds that the applicant appropriately supported the plutonium model results by demonstrating that most calculated concentrations will be higher than the results from SNF dissolution tests.

Neptunium

The applicant identified neptunium as an important risk contributor. In general, the applicant modeled neptunium solubility limits assuming equilibrium with atmospheric oxygen but used two different solid phases depending on conditions. The controlling solid phase for the invert is the oxidized phase Np_2O_5. The choice of the neptunium-controlling mineral in the waste package (waste form and corrosion products domains)—NpO_2 or Np_2O_5—depends on the corrosion status of the steel components. The applicant stated that local reducing conditions during steel corrosion would promote precipitation of reduced NpO_2 over oxidized Np_2O_5. The applicant noted that the literature suggests that, in the presence of reducing materials, NpO_2 is an appropriate solubility-limiting solid phase under most modeled conditions. The applicant chose Np_2O_5 to limit neptunium concentration in the absence of reductants. A sodium neptunium carbonate is modeled at the high pH margin of the water chemistry range. For the typical pH and CO_2 conditions in the waste package (SAR Figures 2.3.7-19 to 2.3.7-21), the sampled neptunium solubility limit in the TSPA model analysis ranges from approximately 0.02 to 11 mg/L [0.02 to 11 ppm] for NpO_2 and from 0.3 to 180 mg/L [0.3 to 180 ppm] for Np_2O_5 (SAR Figure 2.3.7-30); uncertainty terms extend this range to more than an order of magnitude above and below those ranges (e.g., SAR Figure 2.3.7-39).

NRC Staff's Review

The NRC staff finds that the applicant appropriately chose NpO_2 and Np_2O_5 as solubility limiting solids, depending on oxidation–reduction conditions. In support of its neptunium model, the applicant appropriately demonstrated that the range of calculated neptunium concentrations, as a function of pH, exceeds the majority of concentrations from SNF dissolution tests. Furthermore, the mean-value curves for concentration limit versus pH are higher than all the SNF dissolution test concentration values (SAR Figure 2.3.7-39). In DOE (2009ax, Enclosure 12), the applicant compared the model to project and literature data that, in general, suggested higher neptunium solubility limits but are within the 2σ model uncertainty incorporated in its analysis. The applicant showed that the experiments that yielded especially high neptunium concentrations measured solubilities of metastable hydrous neptunium oxides, rather than the lower solubility anhydrous neptunium oxides expected on longer repository time scales. Therefore, the applicant concluded that the highest measured neptunium concentrations could be excluded from development of neptunium concentration limits for performance assessment. In addition, the applicant considered and evaluated an alternative conceptual model involving neptunium incorporation in secondary uranyl minerals resulting from SNF dissolution. The applicant excluded the alternative conceptual model because of inadequate technical bases for its inclusion in performance assessment, an exclusion the NRC staff finds acceptable for the same reasons (Pickett, 2005aa). Consequently, the NRC staff finds the applicant's neptunium model acceptable.

Uranium

The applicant produced two different lookup tables for uranium, depending on the particular chemical environment. In most cases for commercial SNF packages, the hydrated uranyl oxide schoepite is the solubility-limiting solid. For the typical pH and CO_2 conditions in a commercial SNF waste package (SAR Figure 2.3.7-19), the sampled uranium solubility limit in the TSPA code ranges approximately from 1 to 100 mg/L [1 to 100 ppm], as described in SNL (2007ah, Figure 6.7-1). The associated uncertainty terms extend this range as much as ± an order of magnitude, as detailed in SNL (2007ah, Figure 7-6). For codisposal packages in all scenarios and all packages in the igneous intrusion scenario, and for the invert in all scenarios, the uranyl silicate sodium-boltwoodite and $Na_4UO_2(CO_3)_3$ are included in the solubility models. For the typical pH and CO_2 conditions in a codisposal waste package (SAR Figures 2.3.7-20 and 2.3.7-21), the sampled uranium solubility limit in the TSPA code ranges approximately from 1 to more than 10,000 mg/L [1 to more than 10,000 ppm] (SAR Figure 2.3.7-31). The associated uncertainty terms extend this range as much as ± an order of magnitude, as illustrated in SNL (2007ah, Figure 7-6).

NRC Staff's Review

The NRC staff finds that the applicant chose appropriate solubility-limiting solid phases for the uranium solubility limit model. Secondary uranyl minerals such as uranophane (Prikryl, 2008aa) were conservatively excluded in favor of schoepite and sodium-boltwoodite. These excluded minerals could control uranium to even lower concentrations under certain chemical conditions. Laboratory and natural analog studies the applicant cited (SNL, 2007ah) support these choices. A relatively soluble sodium uranyl carbonate was modeled at highest pH and fCO_2. The NRC staff finds that the applicant appropriately compared model results with uranium concentrations measured in SNF dissolution tests; this comparison shows that most measured uranium concentrations plot near or below the model solubility limit curves, as shown in

SNL (2007ah, Figure 7-6). The NRC staff, therefore, finds the applicant's uranium model acceptable.

Thorium, Americium, Protactinium, Radium, and Tin

In developing lookup tables for the other modeled actinides—thorium, americium, and protactinium—the applicant chose solubility-limiting phases on the basis of available data from the literature and the Yucca Mountain program (SNL, 2007ah). For thorium, the limiting solid was ThO_2(am), the solubility model for which produced values more consistent with experimental measurements than models for the lower solubility, crystalline solid ThO_2. [In mineral formulas, "(am)" indicates that this is an amorphous, rather than orderly crystalline, solid. Amorphous solids tend to have higher solubilities than the corresponding crystalline forms.] The americium model used $AmOHCO_3$, which was identified as the controlling solid in Yucca Mountain program studies conducted under appropriate conditions. Protactinium was treated by analogy with neptunium and thorium; that is, the Np_2O_5 model was adopted for protactinium, with wide uncertainty terms accounting for the possibly lower concentrations if behavior was similar to thorium. The model for tin [addressed in SNL (2007ah) but not in the SAR] was also developed using available literature data and considerations of uncertainties; the selected controlling solid in the tin model was SnO_2(am). The applicant used barium as a radium analog for solubility calculations due to their similar chemical properties. The applicant constructed a pH-dependent, stepwise radium solubility limit on the basis of the model results.

NRC Staff's Review

The NRC staff finds that the applicant's concentration limits abstractions for the actinides thorium, americium, and protactinium are acceptable because they were based on solubility models constructed using appropriate solubility-limiting phases. The thorium model was appropriately supported, by comparison, with experimental solubility data that were independent of those used to develop the model. The applicant appropriately supported the americium model by showing that the uncertainty range of predicted concentrations exceeded all americium concentrations measured in SNF dissolution experiments. Although the independent solubility data used to support the protactinium model were sparse, the applicant additionally supported the model by noting that studies show protactinium solubility to be consistently lower than neptunium, which the applicant used for modeling protactinium. The NRC staff finds the tin concentration limit abstraction acceptable because the applicant based it on appropriate solubility models and corroborated the results by comparison to an independent modeling study. In addition, the choice of the tin solubility-limiting solid SnO_2(am) is acceptable because it is more soluble than the other considered tin mineral cassiterite (SnO_2).

The NRC staff finds the applicant's radium concentration limit abstraction to be acceptable. The chemical analogy to barium is supported by literature observations (cited by the applicant) showing their similar geochemical behaviors. The applicant appropriately compared the adopted values to literature data on radium solubility, showing that the abstracted limits are similar to or higher than published model and experimental values. In addition, radium is expected to be coprecipitated in sulfates of other alkaline earth elements, particularly barium (e.g., Zhu, 2004aa), such that dissolved radium would be constrained to very low concentrations. The NRC staff finds that the pure-phase radium solubility model is, therefore, appropriate and conservative.

Uncertainty

In the concentration limits abstraction, the applicant addressed uncertainty in (i) thermodynamic data supporting the solubility models and (ii) the effects of fluoride ion concentration. These uncertainties are applied as additional sampled terms added or subtracted to the lookup table values, with a pH-dependent coefficient applied to the sampled fluoride uncertainty term. These thermodynamic and fluoride uncertainty terms—with normal and triangular distributions, respectively—are sampled once per realization for each element.

NRC Staff's Review

The NRC staff finds that the applicant appropriately accounted for uncertainty in thermodynamic constants by using sampled uncertainty terms that were based on thermodynamic studies in literature the applicant cited. Regarding model uncertainty, the applicant explicitly accommodated the potential effects of fluoride on solubilities in uncertainty terms for the actinide abstractions. Uncertainty in chemical effects on solubility limits is applied implicitly by the in-package chemistry abstraction (SER Section 2.2.1.3.4.3.1).

Integration in the Engineered Barrier System Radionuclide Transport Abstraction

The applicant's dissolved concentration limits abstraction provides maximum concentration values for each element to the engineered barrier system radionuclide transport model, with different values provided for the waste form domain, corrosion products domain, and the invert.

NRC Staff's Review

The NRC staff finds that the applicant appropriately deployed the concentration limits using the GoldSim code in each domain because the implementation accounts for the different chemical conditions. The NRC staff confirmed expected behavior by inspecting plots of modeled dissolved concentrations in the engineered barrier release domains in the TSPA analysis (e.g., DOE, 2009dc). On the basis of the NRC staff's review of the plots, solubility limits in the TSPA model were within the ranges appropriate for the given domain, as would be predicted by the applicant's lookup tables and modeled chemical conditions. In addition, the radioelement concentration ranges either coincided with or were below the corresponding concentration limits.

Summary of NRC Staff's Review of Concentration Limits

The NRC staff finds that the applicant's abstraction of radioelement concentration limits is properly integrated into the engineered barrier system radionuclide release and transport abstraction. The solubility-limit models properly account for the engineered barrier system design, the response of system components to changing in-package conditions, and the effects of those responses on the chemical environment. The applicant properly used equilibrium geochemical models, with appropriately chosen solubility-limiting solid phases, to construct concentration limits lookup tables. Data supporting the various element abstractions are acceptable. Model support is acceptable; the applicant compared model results to appropriate laboratory data on solubility limits and concentrations during waste form dissolution studies. The applicant properly propagated model and data uncertainty through the abstractions.

2.2.1.3.4.3.4 Availability and Effectiveness of Colloids

This section describes the NRC staff's review of the applicant's abstraction and TSPA model implementation of the type, stability, and mass concentration of colloid suspensions in the engineered barrier system, as described in SAR Section 2.3.7.11 and references cited therein. Colloid suspensions inside the engineered barrier system are important to repository performance because if they are stable and exist at sufficiently high concentrations, they could enhance transport of radionuclides that are reversibly or irreversibly associated with them. In this SER Section , the term "irreversible colloids" refers to colloids with radionuclides irreversibly, or permanently, attached to them. The term "reversible colloids" refers to colloids to which radionuclides may attach and detach reversibly.

On the basis of the NRC staff's evaluation of the TSPA code results, the NRC staff concludes that the igneous intrusion and seismic ground motion modeling cases are potentially the most significant to risk with respect to long-term repository performance (SER Section 2.2.1.4.1). The NRC staff verified that irreversible colloids are modeled in the TSPA code as independent species, separate from reversible colloids and dissolved ionic species. The NRC staff verified that radionuclides that are reversibly bound onto colloids can move back into solution and, hence, become part of dissolved radionuclide mass even if reversible colloids are unstable and settle out. Therefore, the NRC staff's review focused specifically on processes and features that limit the stability and mass concentrations of irreversible colloid suspensions [including plutonium-rich zirconium oxide (commercial SNF) colloids, glass-waste-form colloids, and iron oxide colloids] in the engineered barrier system under igneous intrusion and seismic ground motion modeling cases.

The applicant's engineered barrier system colloid model calculates the mass concentrations of reversible and irreversible colloid suspensions in the waste package, corrosion products, and invert domains of the engineered barrier system on the basis of temporal variations in aqueous chemical characteristics (pH and ionic strength), flow rates, and failure status of the engineered barrier system components under nominal and disruptive modeling cases.

Inputs to the engineered barrier system colloid mass concentration abstraction, described in SNL (2008ak, Section 1.1), include in-package ionic strength and pH from the in-package chemistry abstraction, and in-drift ionic strength and pH from the engineered barrier system physical and chemical environment abstraction. Mass concentrations of colloidal suspensions are used to calculate colloid-assisted radionuclide transport in the engineered barrier system radionuclide transport abstraction. The NRC staff's evaluation of these three abstractions can be found in SER Sections 2.2.1.3.4.3.1, 2.2.1.3.3, and 2.2.1.3.4.3.5, respectively.

Colloid Types and Radionuclides Associated With Colloids in the Engineered Barrier System

Colloids are 1- to 2-μm- [4 to 8 × 10^{-5}-in]-sized particles, have the potential to facilitate transport of highly sorbing, low-solubility radionuclides, and may allow radionuclide concentrations in water above their solubility limit. In the TSPA code, colloids in the engineered barrier system are formed by degradation of waste package internals and waste forms and also exist as groundwater colloids in seepage water entering breached waste packages. The applicant used the engineered barrier system colloid abstraction in the TSPA code to determine the stability and mass concentrations of reversible and irreversible colloid suspensions in the waste form, corrosion products, and invert domains of the engineered barrier system (SAR Section 2.3.7.11).

The applicant's engineered barrier system colloid model abstraction focuses on the following five colloid suspension types: (i) glass-waste-form colloids, (ii) plutonium-rich zirconium oxide commercial SNF colloids, (iii) oxidized uranium colloids derived from the SNF, (iv) iron oxide colloids, and (v) groundwater colloids. The applicant considers true or intrinsic colloids to have negligible effects relative to these five colloid types (SNL, 2008ak). The conceptual model identifies two types of radionuclide attachment to colloids: (i) reversible (glass-waste-form colloids, oxidized uranium colloids, iron oxide colloids, and groundwater colloids), in which the radionuclides are reversibly (temporarily) sorbed onto colloid surfaces, and (ii) irreversible (glass-waste-form colloids, commercial SNF colloids, and iron oxide colloids), in which the radionuclides are permanently attached to or embedded in the colloid structure, as detailed in SNL (2008ak, Section 6.3.1) and SNL (2007aj, Section 6.3.4.4). Groundwater colloids and the three types of waste form colloids (glass-waste-form colloids, plutonium-rich zirconium oxide commercial SNF colloids, and oxidized uranium colloids derived from SNF) are considered in the waste form domain, as discussed in SNL (2007aj, Section 6.5.2.5). However, iron oxide colloids are excluded from the waste form domain on the basis of conservatisms and other rationales (e.g., solubility constraints and transport conditions), as outlined in DOE (2009ay, Enclosure 2). All colloid types are considered in the corrosion products domain and the invert. As shown by the data in SNL (2008ag, Tables 6.3.7-62, 6.3.7-63, 6.3.7-64, and 6.3.7-66), the sampled stable glass waste, commercial SNF, oxidized uranium, and groundwater colloid concentration ranges are 0.0004 to 2 mg/L [0.004 to 2 ppm], 0.000015 to 0.6 mg/L [0.000015 to 0.6 ppm], 0.001 to 200 mg/L [0.001 to 200 ppm], and 0.001 to 200 mg/L [0.001 to 200 ppm], respectively, in all the engineered barrier system domains. As shown in SNL (2008ag, Table 6.3.7-65), the sampled stable iron oxide colloid concentration range is 0.001 to 30 mg/L [0.001 to 30 ppm] in the corrosion products and invert domains.

In the applicant's engineered barrier system colloid model abstraction, two radioelements (plutonium and americium) are modeled to permanently attach onto iron oxide colloids or be irreversibly embedded in glass waste form and commercial SNF colloids. As described in SNL (2008ak, Section 6.5.1), seven radioelements (uranium, neptunium, thorium, protactinium, radium, tin, and cesium) are modeled to reversibly sorb onto glass-waste-form colloids, eight radioelements (plutonium, americium, thorium, protactinium, cesium, tin, neptunium, and radium) reversibly sorb onto oxidized uranium colloids, three radioelements (thorium, neptunium, and uranium) reversibly sorb onto iron oxide colloids, and nine radioelements (plutonium, uranium, neptunium, americium, thorium, protactinium, radium, cesium, and tin) reversibly sorb onto groundwater colloids. The irreversible colloids (and their associated masses of plutonium and americium) are modeled as independent species, separate from the dissolved plutonium and americium masses.

NRC Staff's Review

The NRC staff's review verified that the applicant relied on laboratory and field-scale experimental findings published in technical articles found in peer-reviewed journals to determine representative waste form and groundwater colloid types, and types of radionuclides reversibly and irreversibly associated with these colloids in the engineered barrier system colloid model abstraction, as detailed in SNL (2008ak, Section 6.3). The applicant considered both reversible and irreversible colloids in the abstraction and noted the uncertainties associated with the mass concentrations, the stability of these colloid types, and their upscaling and applicability to the Yucca Mountain site. The applicant constructed an empirical ionic strength threshold versus pH curve on the basis of existing experimental data in the open literature to account for uncertainties in the colloid stability. If the computed ionic strength for the in-package (or in-drift) environment is below the ionic strength threshold, then the colloids are stable in the corresponding environment. The applicant constructed empirical cumulative (uncertainty) distributions for the mass concentration of colloids by bounding the distributions using experimental data. The applicant addressed the uncertainty in the mass concentration of colloids by randomly sampling the mass concentration of stable colloids from the corresponding uncertainty distributions. The NRC staff finds that the applicant provided sufficient experimental evidence from the literature and provided adequate technical justifications for the choice of colloid types and the uncertainty ranges for their mass concentrations in the engineered barrier system model abstraction.

Excluded Colloid Processes

In the applicant's engineered barrier system colloid model abstraction, colloidal filtration, thin-film straining (retardation of colloid transport when colloid dimensions exceed the water film thickness), gravitational settling of colloids, and sorption of colloids on stationary surfaces and onto an air–water interface were excluded due to associated uncertainties, and exclusion of these processes was considered to be conservative. The abstraction did not include biocolloids because of low microbial activity and negligible mass concentrations of such colloids in comparison to groundwater colloids.

NRC Staff's Review

The NRC staff finds acceptable the applicant's exclusion of colloidal filtration, straining, gravitational settling of colloids, and sorption of colloids on stationary surfaces and onto an air–water interface, as detailed in SNL (2008ak, Section 5.8) and SNL (2007aj, Section 5.7). Exclusion of these processes results in higher modeled radionuclide releases, and hence exclusion of these processes is conservative and acceptable. Moreover, the NRC staff finds that the applicant's exclusion of biocolloidal (e.g., viruses, bacteria, spores, or other microorganisms) transport is acceptable because the low humic substance concentrations at the Yucca Mountain site will not support high biocolloid concentrations.

Importance of Colloids to Risk under Disruptive Events

The NRC staff's review of the importance of colloids to risk focused on two disruptive scenario modeling cases: igneous intrusion and seismic ground motion. The applicant showed, through its analyses, that these two modeling cases contribute the most to the total mean annual dose for 10,000 and 1 million years after repository closure (SAR Figure 2.4-18). These modeling cases are, therefore, useful for evaluating the potential significance to risk of colloids. Pu-242 is the most important contributor to the overall total mean dose after 200,000 years (SAR Figure 2.4-20). Under the nominal scenario, the maximum Pu-242 activity due to irreversible colloids is about 30 percent of the total Pu-242 activity leaving the engineered barrier system; under the seismic scenario, it is only about 18 percent of the total Pu-242 activity leaving the engineered barrier system. The maximum Pu-242 release rate leaving the engineered barrier system due to irreversible colloids is 2.5 percent of the total Pu-242 release rate under the igneous intrusion modeling case, as described in SNL (2008ag, Section P18.3) and DOE (2009an, Enclosure 5). The applicant concluded that, for plutonium in the engineered barrier system, colloid-facilitated radionuclide transport is less effective than dissolved phase radionuclide transport, as shown in SAR Figures 2.1-20 and 2.1-23 and SNL (2008ad, Table A–2, p. A–130).

In the applicant's igneous intrusion modeling case, the drip shield and waste packages entirely fail; hence, there is no distinction between seep and no-seep cases, and water chemistries and colloid stability remain nearly constant when the temperature drops below the boiling point of water and water flow to the waste form is established. For the igneous intrusion modeling case, smectite colloids (derived from high-level waste glass or from the tuff host rock) and oxidized uranium colloids (derived from SNF degradation) are stable in the engineered barrier system, but commercial SNF colloids are completely unstable in the corrosion products domain, according to DOE (2009ay, Enclosure 3).

For the igneous intrusion case, unstable and settled commercial SNF colloids in the corrosion products domain can be as important as the stationary corrosion products for the retention of Pu-242, as described in DOE (2009dc, Figure 1.1-26), DOE (2009dd, Enclosure 1), and DOE (2009ay, Enclosure 3, Figure 1), due to a narrow range of pH, 7 to 8.4. The narrow pH range is illustrated in DOE (2009ay, Enclosure 3) and supported in DOE (2009da, Enclosure 2). For a representative realization shown in DOE (2009dc, Figure 1.1-26), the contribution of suspended iron oxide colloids to Pu-242 mass in the corrosion products domain is about eight orders of magnitude smaller than the Pu-242 mass removed from inventory by the settled unstable commercial SNF colloids at 200,000 years for the igneous intrusion modeling case. Therefore, the applicant concluded that iron oxide colloids are insignificant for plutonium mobility when commercial SNF colloids sequester Pu-242. In contrast, DOE (2009da, Enclosure 1) shows that, in 62 percent of realizations in the igneous intrusion case, commercial SNF colloids are unstable in the waste form domain, but iron oxide colloids are stable in the corrosion products domain. Therefore, the applicant concluded that commercial SNF colloids have a negligible effect on Pu-242 waste package mobility in these cases. For a representative realization for these conditions, Pu-242 mass irreversibly associated with iron oxide colloids is three orders of magnitude lower than the Pu-242 mass in the dissolved phase and about seven orders of magnitude lower than the sorbed Pu-242 mass on stationary corrosion products, as shown in DOE (2009da, Enclosure 1, Figure 7). The applicant stated that this observation indicates insignificant effects of iron oxide colloids on Pu-242 waste package releases.

For the seismic ground motion case, the applicant assessed that damage on waste packages is mainly due to patch failures by general corrosion. Unlike in the igneous intrusion modeling case, colloid concentrations are sensitive to seep versus no-seep environments after corrosion patches develop on the waste packages, and the ionic strengths of water and pH vary in time. For the seismic ground motion case, the ionic strength of waters in the engineered barrier system depends on the relative humidity when the water flux is less than 0.1 L/yr [0.026 gal/yr] under the condition of complete filling of the drift with rubble; otherwise, ionic strength depends on the chemistry of the advective flux through corrosion patches. The ionic strengths of seep water also correlate with the rubble-filling status, as shown in DOE (2009ay, Enclosure 3, Figures 17, 18, 23, and 24).

For the seismic ground motion modeling case, the applicant noted that diffusive transport (under no-seep conditions) through the engineered barrier system and water chemistry (pH and ionic strength) could largely limit colloid-facilitated transport, as shown in SAR Section 2.4.2.2.3.2.2 and DOE (2009ay, Enclosure 3).

The applicant's TSPA model results indicate that for the seismic ground motion case, initially high ionic strength leads to unstable colloid suspensions. After the waste packages are breached in seep and no-seep conditions, the ionic strength (which depends on the relative humidity during this stage) drops to a level where smectite and oxidized uranium colloids become stable in the codisposal packages and largely stable (in more than 95 percent of realizations) in the commercial SNF waste packages. These colloids are stable for the remainder of the simulation. As was the case for the igneous intrusion modeling case, commercial SNF colloids are unstable in the corrosion products and invert domains for the seismic ground motion modeling case, as shown in DOE (2009ay, Enclosure 3, Figures 6, 29, and 30). Stability of colloid suspensions, except for the iron oxide colloids, is similar for both the seep and no-seep cases. Iron oxide colloids are stable in the corrosion products domain only when seep water enters the waste package through corrosion patches at later times, whereas they are largely unstable (in more than 95 percent of realizations) under the no-seep conditions, as shown in DOE (2009ay, Enclosure 3, Figures 15 and 16).

NRC Staff's Review

The NRC staff finds acceptable the applicant's conclusion that iron oxide colloids are insignificant contributors to Pu-242 releases from the corrosion products domain because Pu-242 is (i) largely sorbed onto stationary corrosion products, (ii) associated with settled and unstable commercial SNF colloids, or (iii) in the dissolved phase.

The NRC staff reviewed the TSPA model results for the disruptive modeling cases (those highly significant to risk) to evaluate processes and features that could limit availability and transport of colloid suspensions in the waste form, corrosion products, and invert domains of the engineered barrier system, as detailed in DOE (2009ay, Enclosure 3). The NRC staff finds acceptable the applicant's identification, description, and quantification of the distinct processes and features that control stable colloid concentrations in the engineered barrier system (e.g., considering the barrier capability of engineered barrier system components or the impact of seep and no-seep environments) for both the igneous intrusion and the seismic ground motion modeling cases. In addition to the simpler, long-term constant geochemical conditions considered in the igneous modeling case, the seismic ground motion modeling case demonstrated the effects of temporal variability in the stability and mass concentrations of colloids under both seep and no-seep conditions, as a function of patch failure developments on waste packages and resulting changes in ionic strength, pH, and relative humidity in the engineered barrier system.

The NRC staff conducted independent, simplified, confirmatory calculations on the effectiveness of iron oxide colloids in facilitating Pu-242 releases in the igneous intrusion modeling case. The igneous intrusion modeling case was chosen for independent calculations because (i) chemical conditions and colloid stability remain unchanged throughout the entire simulation after a relatively short cooling period (less than 1,000 years), (ii) this modeling case dominates the long-term total mean annual dose, and (iii) that dose is dominated by Pu-242 after 200,000 years. Using information provided by the applicant and used for Pu-242 in the TSPA model, the NRC staff's confirmatory calculations concluded that the ratio of (i) the plutonium attachment rate to iron oxide colloids to (ii) the plutonium attachment rate to stationary corrosion products is 2×10^{-7} (Pickett, 2010aa). These calculations showed that plutonium attachment to stationary corrosion products is much faster than attachment to iron oxide colloids.

As discussed in SER Section 2.2.1.3.4.3.5, the applicant showed in DOE (2009ay, Enclosure 8) that reversible, kinetic plutonium sorption onto stationary corrosion products can be approximated as an equilibrium process. Therefore, the rate of plutonium desorption from the stationary corrosion products is approximately equal to the rate of sorption. On the basis of this observation and the NRC staff's confirmatory calculation summarized in the previous paragraph, the NRC staff concludes that the rate of irreversible plutonium sorption to iron oxide colloids is many orders of magnitude slower than the rate of plutonium desorption from stationary corrosion products to solution. Therefore, any transfer of dissolved plutonium to iron oxide colloids would be compensated by desorption from the stationary corrosion products—which contain the majority of plutonium mass in the corrosion products domain—to maintain the quasi-equilibrium relationship. On the basis of its calculation, the NRC staff concludes that irreversible sorption of plutonium to iron oxide colloids cannot substantially deplete dissolved plutonium. Sorption to stationary corrosion products is more important to plutonium release from the corrosion products domain than are iron oxide colloids. This result is consistent with the applicant's conclusion that iron oxide colloids are not significant for Pu-242 releases from the engineered barrier system, as shown for representative realizations in DOE (2009dc, Figures 1.1-24 and 1.1-26) and DOE (2009da, Enclosure 1, Figures 5 and 7).

For codisposal packages in disruptive scenarios, the NRC staff compared the plutonium release effectiveness of high-level waste glass colloids against dissolved plutonium. Results of a representative igneous intrusion modeling case realization in DOE (2009dc, Enclosure 1, Figures 1.1-29 and 1.1-30) showed dissolved Pu-242 being released from the engineered barrier system more than ten times faster than Pu-242 associated with high-level waste glass colloids. In addition, the mean plutonium solubility limit in the corrosion products domain in the TSPA model, which the NRC staff estimated as 10^{-5} g/L [0.01 ppm] from applicant data presented in SNL (2007ah, Table 6.5-1), is about 10 times higher than the mean concentration of plutonium associated with high-level waste glass colloids, which the NRC staff calculated as 1.2×10^{-6} g/l [0.0012 ppm] from data presented in SNL (2008ak, Table 6-4). The NRC staff concludes, on the basis of these confirmatory calculations, that high-level waste glass colloids are, in general, less effective than the dissolved phase in effecting plutonium release from codisposal packages.

The NRC staff considered (i) the insignificance of iron oxide colloids for Pu-242 releases from the corrosion products domain; (ii) the relatively shorter residence times for colloids and radionuclides in the waste form domain than in the corrosion products domain; and (iii) the presence of nearly 60–70 percent less steel corrosion product mass in the waste form domain than in the corrosion products domain, as described in DOE (2009ay, Enclosure 2), and finds that excluding iron oxide colloids from the waste form domain will not be significant to risk.

The NRC staff finds that reduction in colloid mass concentration under no-seep environments (which leads to diffusion-dominant colloid migration) is acceptable because the size of the colloids is a few orders of magnitude larger than the ions of aqueous radioelements, and the colloids, therefore, will encounter higher diffusional resistance. Colloids diffuse much more slowly than dissolved ionic species, such that colloid-associated release under no-seep conditions is negligible. Therefore, the NRC staff finds acceptable the applicant's analysis of colloids under disruptive igneous and seismic events.

Data Support and Uncertainty Propagation for Colloid Transport Abstraction for the Engineered Barrier System

Mass Concentration of Colloids

The applicant used experimental and scientific literature data to support its colloid mass concentrations model. Uncertainties associated with the mass concentrations and stability of glass waste form colloids in the TSPA model rely on results from drip and immersion tests for degradation of alkali borosilicate glasses, as detailed in SNL (2008ak, Section 6.3.2.2). Experimental data were used to bound plutonium mass concentrations associated with zirconium oxide colloids formed from commercial SNF, as described in SNL

(2008ak, Section 6.3.2.4). Uranophane colloids are used as representative colloids for oxidized uranium colloids formed from defense and commercial SNF.

SNL (2008ak, Section 6.3.2.6) described an empirical cumulative distribution for the mass concentrations of groundwater colloids that was adopted for uranophane colloid suspensions because both colloid suspensions display a similar stability profile. For iron oxide colloids, the applicant relied on bench-scale experiments using a carbon-steel miniature waste package in bathtub and flow-through configurations. These experiments used water chemically similar to well water near Yucca Mountain to induce corrosion and subsequently to determine the geometric mean concentration of iron oxide colloids. Empirical cumulative distributions of colloid mass concentrations were constructed, on the basis of laboratory-scale experimental data to address uncertainties in colloid mass concentrations, as detailed in SNL (2008ak, Section 6.3). The applicant adopted colloid concentrations in groundwater at the Yucca Mountain site for colloid concentrations in seepage water entering breached waste packages. The applicant collected colloid data from nine different sources and fitted them to a cumulative mass distribution to address uncertainties.

NRC Staff's Review

With regard to the applicant's colloidal mass concentration model, the NRC staff verified that the applicant relied on data from laboratory experiments published in peer-reviewed technical journals to determine the range for mass concentrations of reversible and irreversible colloids. The NRC staff reviewed (i) how the applicant addressed uncertainties associated with how closely geochemical and hydrogeological conditions were represented in these experiments and (ii) how the applicant upscaled laboratory findings to the field scale at the Yucca Mountain site. The applicant acknowledged uncertainties associated with the mass concentrations of colloid suspensions due to, among other things, measurements, experimental factors, and upscaling of experimental data and observations to repository scale and conditions (SAR Section 2.3.7.11.2). The applicant set the upper bound for mass concentrations of iron oxide colloids to be larger than natural iron oxide colloid concentrations in groundwater (SAR Section 2.3.7.11.2). The NRC staff finds that, in the absence of field data, the use of laboratory data is adequate because the applicant incorporated associated uncertainties (through sampling from uncertainty distributions) in determining the mass concentration of waste form and iron oxide colloids. For mass concentrations of groundwater colloids, the applicant appropriately employed existing field data. The applicant kept the pH and ionic strength range, which was based on experimental and/or literature data, wide enough in colloid stability analyses to cover expected stability conditions at the Yucca Mountain site.

The NRC staff's review verified that the applicant addressed uncertainties in mass concentrations by sampling them from empirically constructed cumulative mass distributions obtained from experimental data. The NRC staff finds that the applicant adequately propagated data uncertainty by (i) constructing empirical mass concentration distribution functions for each colloid type using relevant experimental data and sampling the mass concentration for a particular colloid suspension type and (ii) adequately addressing model integration and information exchange between the engineered barrier system abstraction and other abstractions.

In-Package and In-Drift Stability of Colloids

In TSPA code calculations, in-package and in-drift stability of colloid suspensions is determined by the ionic strength and pH. The applicant constructed an empirical ionic strength threshold versus pH curve using experimental data specific to each colloid suspension type and using the Derjaguin-Landau-Verwey-Overbeek (Derjaguin and Landau, 1941aa; Verwey and Overbeek, 1948aa) theory. In-package and in-drift pH and ionic strengths and dissolved radionuclide concentrations are computed outside the engineered barrier system abstraction and then fed into the empirical ionic strength threshold versus pH curve in the engineered barrier system abstraction, as outlined in SNL (2008ak, Section 6.5). For stability calculations, the applicant modeled glass waste form colloids and groundwater colloids as smectite, plutonium-rich zirconium oxide colloids as zirconium oxide, oxidized uranium oxide colloids as uranophane, and iron oxide colloids as hematite, as described in SNL (2008ak, Section 6.3.1). If the colloid suspensions are stable, their mass concentrations are sampled from empirical distribution functions specific to the colloid type (constructed from experimental data). If a colloid type is unstable, the mass concentration is set to a low nonzero value, selected such that the colloid mass is too low to have any impact on radionuclide release and transport. In the case of groundwater colloids in the waste package, the initial concentration is set to 0 mg/L [0 ppm] until flow begins in the waste package (SNL, 2008ak).

NRC Staff's Review

The NRC staff's review of the in-package and in-drift colloidal stability abstraction verified that the applicant constructed empirical relations (on the basis of experimental data in the literature) for each colloid suspension type that related the ionic strength threshold to pH to determine stability of the colloidal suspension in the waste package and in the drift. These empirical relations were constructed on the basis of the Derjaguin-Landau-Verwey-Overbeek model (Derjaguin and Landau, 1941aa; Verwey and Overbeek, 1948aa). The applicant stated that these empirical relations were purely mathematical and driven by experimental data (SNL 2008ak, Section 6.3.2.4). The engineered barrier system model abstraction computes the ionic strength of the in-package fluid (or in-drift fluid) and compares it against the ionic strength threshold versus pH curve. If the in-drift (or in-package) fluid ionic strength exceeds the ionic strength threshold, then the corresponding colloidal suspensions become unstable in the abstraction, as detailed in SNL (2008ak, Section 6.3.2). The NRC staff finds that the stability calculations for colloid suspensions are adequate because (i) the Derjaguin-Landau-Verwey-Overbeek theory (Derjaguin and Landau, 1941aa; Verwey and Overbeek, 1948aa) has been commonly used in the literature to determine the stability of relatively dilute colloid suspensions, (ii) the empirical relations for ionic strength versus pH were constructed on the basis of experimental data, and (iii) the applicant addressed temporal variations in in-package and in-drift water chemistry in colloid stability calculations.

Radionuclide Mass Sorption on Colloid Suspension

The applicant referred to previously published data to determine surface areas for reversible glass waste and groundwater colloids, uranophane colloids, and iron oxide colloids in SNL (2008ak, Sections 6.3.2.3.1, 6.3.2.7, and 6.3.12.2, respectively). This information is used in calculating sorbed radionuclide mass on colloid suspensions.

NRC Staff's Review

The NRC staff's review verified that the applicant determined the range of specific surface area for reversible colloid suspensions based on open literature experimental data and sampled from this range to address data uncertainties. On the basis of the NRC staff's confirmatory review of the literature data cited in SNL (2008ak) and the applicant's use of sampled specific surface area to account for uncertainties, the NRC staff finds that the uncertainty distributions and literature data the applicant provided for specific surface area are acceptable.

Kinetic Attachment Rates for Plutonium and Americium onto Iron Oxide Colloids

In the applicant's engineered barrier system abstraction, plutonium and americium are modeled to be irreversibly attached onto iron oxide colloids. As described in SNL (2008ak, Section 6.3.12.2), the applicant constructed an uncertainty distribution function for the attachment rate constant for iron oxide colloids on the basis of data from sorption experiments. The applicant noted that the attachment rate is sampled from an experimentally supported lognormal uncertainty distribution under a no-seep condition or a condition where colloid suspensions are unstable in the corrosion products domain. Otherwise, the maximum of the sampled rate constant from a lognormal uncertainty distribution and the computed rate constant using the target flux-out ratio is used, as detailed in SNL (2007aj, Section 6.5.2.4.6).

NRC Staff's Review

The NRC staff reviewed the basis of the applicant's description for the attachment rate calculations and finds that the applicant used attachment rates sampled from an experimentally supported uncertainty distribution for modeling irreversible attachments of plutonium and americium onto iron oxide colloids. The model favors attachment onto iron oxide colloids by implementing the target flux-out ratio if the computed attachment rate remains within the experimentally determined range for the attachment rate; otherwise, the sampled attachment rate is used and the target flux-out ratio is not implemented. The NRC staff finds this modeling feature is conservative because it would allow more radionuclides to be transported from the corrosion products domain to the invert by stable iron oxide colloids (SER Section 2.2.1.3.4.5). The NRC staff further finds that because iron oxide colloids do not play a significant role in dose calculations (as evaluated in the previous section), the method chosen for irreversible attachment rate calculations is not important to dose calculations.

Alternative Conceptual Model Consideration

The applicant considered two alternative conceptual models with respect to colloids: the first uses the waste degradation rate to calculate the generation rate of glass waste form colloids and the second focuses on air–water limitations on particle releases from weathered waste form surfaces under unsaturated conditions, as identified in SAR Section 2.3.7.11.3.2 and SNL (2008ak, Section 6.4). The applicant did not implement these alternative models in the TSPA code.

NRC Staff's Review

The NRC staff's review verified that the applicant used a conceptual model in the TSPA code for irreversible and reversible colloidal transport on the basis of a set of mass-balance equations. The NRC staff finds that a mathematical framework for this conceptual model is consistent with colloid transport models in the literature (e.g., Corapcioglu and Jiang, 1993aa; van de Weerd, et al., 1998aa).

Without high water flow rates, mobile colloidal generation in the alternative conceptual models would be negligible because of strong and irreversible attachment of hydrophobic colloids to air bubbles on the surface of waste package components. In addition, the NRC staff finds that available data are insufficient to support the quantities of mobile colloids predicted by these two alternative conceptual models.

The applicant also considered the bathtub flow model as an alternative conceptual model to the flow-through model, which is implemented in the TSPA code, in simulating water flow and radionuclide transport in a breached waste package. As discussed in SER Sections 2.2.1.3.3.3.3 and 2.2.1.3.4.3.5, the flow-through model is bounding to the bathtub model for flow and radionuclide transport simulations. The applicant did not discuss potential implications of the bathtub flow model on colloid transport in the engineered barrier system. The NRC staff concludes that this was acceptable because if the commercial SNF colloids are unstable in the corrosion products domain during pulse periods, large fractions of commercial SNF colloids irreversibly associated with Pu-242 would be removed from the inventory, and hence, the implementation of the bathtub model in the TSPA code would underestimate Pu-242 releases from the engineered barrier system.

Summary of NRC Staff’s Review of Availability and Effectiveness of Colloids

The NRC staff finds that the applicant adequately described the engineered barrier system colloid transport model abstraction and its integration with other abstractions used in the TSPA code. The mathematical framework and the underlying conceptualization for the reversible and irreversible colloids and colloid stability analyses are consistent with models in the scientific literature. The absence of other conceptualizations used in the TSPA code is acceptable given that sufficient data were not available to support alternative models. The abstraction adequately propagates uncertainties through laboratory-data- or field-data-based cumulative distributions for mass concentrations, stability, and transport parameters.

The NRC staff also finds that the applicant adequately used disruptive modeling cases that are most significant to risk (igneous intrusion and seismic ground motion modeling cases) to show the ineffectiveness of colloid-assisted radionuclide releases from the engineered barrier system in comparison to radionuclide releases in a dissolved phase. On the basis of the NRC staff’s evaluation of the TSPA code results and the NRC staff’s independent confirmatory analyses, the NRC staff concludes that the limitation on stable colloid masses is the main reason for the insignificance of colloid-assisted radionuclide transport in both seep and no-seep environments. Moreover, diffusive release by colloids is limited due to size effects, making colloid transport further ineffective under no-seep environments. In conclusion, the NRC staff finds that dissolved radionuclides will be more significant than colloid-associated radionuclides to radionuclide release and transport and, therefore, to dose.

2.2.1.3.4.3.5 Engineered Barrier System Radionuclide Transport

This section details the NRC staff’s review of the applicant’s abstraction and TSPA implementation for radionuclide transport in the engineered barrier system, as described in SAR Section 2.3.7.12 and references cited therein (particularly SNL, 2007aj). The abstraction estimates the rate of movement of radionuclides from degraded waste forms to the unsaturated zone and provides radionuclide fluxes (rates of mass transfer) versus time to the unsaturated zone transport abstraction (SER Section 2.2.1.3.7). Major inputs to the abstraction include the flow conditions inside the engineered barrier system (SER Section 2.2.1.3.3.3.3), the chemical conditions inside the engineered barrier system (SER Section 2.2.1.3.4.3.1), waste form degradation rates (SER Section 2.2.1.3.4.3.2), dissolved concentration limits (SER Section 2.2.1.3.4.3.3), and colloid parameters (SER Section 2.2.1.3.4.3.4).

The applicant’s abstraction for radionuclide transport in the engineered barrier system is highly significant to risk because large masses of plutonium and other dose-significant actinides are retained in the engineered barrier system in the applicant’s TSPA calculations. For example, in DOE (2009dc), the applicant provided results for a representative realization of the igneous intrusion modeling case showing that approximately 8,000 kg [17,600 lb] of Pu-242 is permanently immobilized in the engineered barrier system for one percolation subregion. In the same realization and subregion, approximately 30,000 kg [66,000 lb] of Np-237 is retained on the waste package corrosion products at 100,000 years; Np-237 is released from the engineered barrier system slowly enough that more than 1,000 kg [2,200 lb] remained at 1 million years.

On the basis of the importance to the abstraction, the NRC staff’s review focused on model framework and process conceptualization within the TSPA code implementation, representation of diffusion, sorption on stationary corrosion products, colloid-facilitated transport, and reasonableness and consistency of TSPA code results. The abstraction contains several included FEPs. Excluded FEPs are discussed in SER Section 2.2.1.2.1.3.2.

Model Framework and Process Conceptualization

Overall Conceptualization

The applicant based the abstraction and TSPA implementation on one-dimensional mass transport through three computational domains: (i) waste form domain, (ii) corrosion products domain, and (iii) invert domain. The waste form domain contains a single computational cell representing a porous rind of degraded waste form for the commercial SNF packages. The waste form domain for codisposal packages comprises a computational cell representing high-level waste glass upstream of a cell representing DOE SNF. Corrosion products formed from the degradation of steel waste packages and package internals are represented in the corrosion products domain. The invert domain is assumed to be in close contact to the waste package and composed of crushed tuff material. A fourth domain, the invert-unsaturated zone interface, facilitates transfer of the radionuclide mass from the engineered barrier system transport model to the unsaturated zone transport model.

The applicant conceptualized the transport pathway as a flow-through model in which water flows vertically through a degraded waste package. The applicant considered an alternative conceptual model in which the outlet for water is not on the underside of the waste package. In this bathtub model, water would fill the partially intact waste package until it reaches a spill point corresponding to a breach on the side of the waste package. In a variant of the bathtub model, the stored water and dissolved radionuclides would be suddenly released when a second breach develops on the underside of the waste package.

NRC Staff’s Review

The NRC staff finds acceptable that the overall structure of the transport abstraction is based on a one-dimensional spatial discretization because it appropriately represents the transport pathway in the engineered barrier system with sufficient flexibility to represent the range of expected conditions and processes (Painter, 2010aa).

The applicant has adequately demonstrated that the flow-through model provides an upper bound on radionuclide transport in the absence of a sudden release based on considerations of water residence times, as discussed in SNL (2007aj, Section 6.6.1.2.3). A sudden release of water stored in the waste package in the bathtub scenario could create a short-duration, high-intensity pulse in radionuclide release from the engineered barrier system. However, several mechanisms mitigate the effects of such a pulse by dispersing it in time. The pulse from a single package would be dispersed by physical processes such as sorption and dispersion in the engineered barrier system and the lower geological barrier. Radionuclide concentrations in the pulsed water would, therefore, become lower as the water moves from the engineered barrier system and through the natural barriers. Moreover, the pulses from individual realizations would be distributed in time, and the effect of any one pulse on the mean dose would be greatly reduced because the combined, averaged effects of time-distributed pulses would be similar to the effect of continuous flow-through releases. On the basis of its review of this information, the NRC staff finds that the applicant's choice of the flow-through conceptualization over the bathtub model is acceptable and does not underestimate release.

Transport Model Framework

In the applicant's abstraction, dissolved radionuclides and radionuclides sorbed onto the five types of mobile colloids (SAR Section 2.3.7.11 and SER Section 2.2.1.3.4.3.4) are transported by diffusion and, if water is flowing in the engineered barrier system, advection. Advective velocities for colloids are identical to the water velocity. Advective transport of selected dissolved radionuclides is slowed by sorption onto stationary corrosion products. Solubility limits on the dissolved radionuclide concentrations are also imposed.

NRC Staff's Review

On the basis of prelicensing interactions (e.g., MacKinnon, 2008aa) and the NRC staff's experience with modeling similar systems, the NRC staff considers the represented diffusive and advective processes to be the dominant processes for transport. The NRC staff thus finds the processes and couplings represented in the applicant's abstraction acceptable, such that release of radionuclides is not underestimated.

Transport Under Disruptive Events

The applicant's TSPA implementation of engineered barrier system transport is similar for the disruptive and nominal modeling cases, although conditions within the engineered barrier system (and, thus, inputs for the abstraction) may be different following disruptive events. Most importantly, the applicant assumed instant degradation of the waste forms and advective conditions within the engineered barrier system following an igneous intrusion event.

NRC Staff's Review

The NRC staff finds acceptable the applicant's modeling assumption that under an igneous intrusion event, the waste forms are instantaneously degraded and are transported under advective conditions. These assumptions for igneous intrusion events provide an upper bound on engineered barrier system transport, and ensure that the release of radionuclides is not underestimated.

Model Abstraction and TSPA Model Results

In response to the NRC staff's request for additional information regarding consistency between the process model abstraction and the TSPA code calculated results, the applicant provided additional information on the mass retained in and flux out of each computational domain for key representative radionuclides using single representative realizations of the igneous intrusion and nominal modeling cases (DOE, 2009da,dc). For I-129 in commercial SNF packages, 99.96 percent of the initial inventory is transported out of the engineered barrier system in the first TSPA timestep following the intrusion event. Release of Np-237 is significantly delayed but not eliminated by precipitation and sorption onto stationary corrosion products. For example, in one realization, about 14 percent of the initial Np-237 inventory is released from the engineered barrier system in the first 40,500 years following the igneous intrusion event; after 1 million years the released fraction is 88 percent of the initial inventory (including ingrowth).

For Pu-242, the applicant provided information for two realizations: one with stable waste form colloids in the waste form domain, described in DOE (2009dc), and one with unstable waste form colloids in the waste form domain, described in DOE (2009da, Enclosure 1). For the realization with stable waste form colloids in the waste form domain, which according to DOE (2009dd, Enclosure 1) is representative of about 38 percent of realizations, about 11 percent of the initial inventory is released from the engineered barrier system in the first 204,000 years. Nearly all of the remaining Pu-242 mass is retained in the corrosion products domain irreversibly associated with permanently immobilized (settled) waste form colloids in this realization. In the applicant's realization results with unstable waste form colloids in the waste form domain, which is representative of 62 percent of realizations, precipitation of plutonium-bearing minerals and sorption onto stationary corrosion products delay release but do not permanently sequester Pu-242—similar to the case described for Np-237 in the previous paragraph.

NRC Staff's Review

The NRC staff finds that rapid transport of I-129 following the igneous intrusion event is consistent with the conceptual process model for highly soluble, nonsorbing species, for which no significant engineered barrier system retention processes are represented. The NRC staff finds that the analyzed TSPA abstraction results for Np-237 release are consistent with the conceptual process models because dissolved neptunium concentrations are limited by precipitation of neptunium-bearing minerals and by strong sorption onto stationary corrosion products in this realization. The NRC staff finds the mechanism by which Pu-242 is retained in the corrosion products domain to be consistent with the process conceptualization when waste form colloids are calculated to be stable in the waste form domain and unstable in the corrosion products domain. The NRC staff finds the Pu-242 behavior when waste form colloids are unstable in the waste form domain to also be consistent with the process conceptualization.

Summary of Diffusion Models

The various analytical models used to simulate diffusive transport in the TSPA computer code are summarized next.

The applicant calculated diffusion coefficients for dissolved radionuclides as the product of tortuosity (the effect of flow path shape in a porous medium) and species-dependent free-water diffusion coefficients. The diffusion coefficients were adjusted for temperature. The tortuosity was empirically related to porosity and liquid saturation using standard models. The applicant based diffusion coefficients for colloids on the Stokes-Einstein equation, which accounts for temperature and particle size. The NRC staff finds that the mathematical representation of diffusion and the approach for relating diffusion coefficients to porosity, water content, and temperature are acceptable, in that they are based on established models in the peer-reviewed scientific literature and are applied to conditions within their valid range.

For the no-dripping situation, the applicant calculated liquid water content from relative humidity using empirical adsorption isotherms. This information is needed to establish diffusion coefficients, which are dependent on water content. The applicant developed a diffusion model for these conditions and conducted a literature review on data relevant to predicting water content on the basis of relative humidity, as described in SNL (2007aj, Section 6.3.4.3). The applicant compared the output of the abstraction for adsorbed water content versus relative humidity with literature data for adsorption on goethite, hematite, Cr_2O_3, and NiO. The applicant showed that, within uncertainty bounds, the output of the abstraction is consistent with the experimental data for relative humidity of about 0.42 and greater, which is the range of relative humidity in which diffusion may be significant, as outlined in SNL (2007aj, p. 7-22). The applicant also compared the model output to the results of independent modeling studies. The NRC staff reviewed the applicant's data, model development, and model corroboration and finds that the water adsorption isotherm model for diffusion under no-dripping conditions is acceptable because it was derived using applicable literature data and was appropriately corroborated.

The mass of steel corrosion products is needed to establish the liquid water content, which the applicant calculated as a function of time from the degradation of steel internals of the waste package. The NRC staff compared the corrosion rates for stainless and carbon steels used in this abstraction with literature values and additional information the applicant provided in DOE (2009db, Enclosure 2) and concluded that the uncertainty distributions are appropriate. The NRC staff concludes that stainless steel corrosion rates are appropriate for representing diffusion because a possible overestimation of the corrosion rate at the lower end of the range would be conservative for simulating diffusion. The upper end of the corrosion rate is consistent with the data and would not result in an underestimate of liquid water content (and, therefore, the diffusion coefficient). The NRC staff also reviewed the applicant's use of design information in establishing the corrosion product mass and finds that the appropriate information on waste package design was used.

For the dripping situation, the applicant assumed the porous materials were saturated with liquid water. The NRC staff finds that this assumption is acceptable because it provides an upper bound with respect to diffusive transport; in general, contaminants will diffuse faster in a fully saturated medium than in a partially saturated medium.

The applicant used data on diffusion in crushed tuff material to develop uncertainty distributions for invert diffusion coefficients. Uncertainty in the diffusion coefficients and invert porosity are explicitly represented. The NRC staff finds that the supporting experiments are representative of the expected conditions in the invert and that the applicant's uncertainty distributions are, thus, appropriately developed.

The applicant compared the output of the invert diffusion coefficient model with two sets of experimental results. The applicant's abstraction predicts larger diffusion coefficients than the experimentally determined values. The NRC staff finds the abstraction is acceptable because it produces an upper bound of the diffusion coefficient and the diffusive transport of radionuclides in the invert.

The applicant considered two alternative conceptual models related to diffusion in the invert. One of these alternative conceptual models is based on a dual-continuum representation of diffusion. The NRC staff finds that the applicant's selection of the single-continuum model over the dual-continuum model is appropriate because little intergranular moisture is expected in the invert, except in the dripping situation, in which case diffusion is expected to be minor compared with advection. (Little moisture implies negligible intergranular diffusion, and the adopted single-continuum model provides an upper bound on diffusive transport.) The applicant's other alternative conceptual model considers alternative relationships between diffusion coefficients and moisture content at low-moisture content. The NRC staff finds that the applicant's selection of the single-continuum model is appropriate because it is bounding. The second alternative, like the dual-continuum model, does not result in higher diffusive transport.

In the applicant's TSPA model, radionuclide mass enters the unsaturated zone after leaving the invert. Because the mass flux out of the invert is partly the result of diffusion, radionuclide concentrations in the unsaturated zone are needed to obtain estimates of the mass flux. The applicant modeled a portion of the unsaturated zone in the engineered barrier system/unsaturated zone interface to calculate the diffusive fluxes into the unsaturated zone transport model. The engineered barrier system/unsaturated zone interface is a network of computational cells representing a local region of the unsaturated zone just below a drift. Hydrological conditions in the engineered barrier system/unsaturated zone interface are established similarly to the unsaturated zone transport model (SER Section 2.2.1.3.7). A zero-concentration boundary condition is used at the lower boundary of the interface zone. Because of this assumption, the applicant's use of an engineered barrier system/unsaturated zone interface zone to couple the engineered barrier system and unsaturated zone abstractions produces an upper bound on diffusive transport into the unsaturated zone, and the NRC staff, therefore, finds use of the interface zone acceptable.

Sorption on Corrosion Products

In the applicant's abstraction, radionuclides enter the corrosion products domain from the waste form domain both in solution and associated with colloids. The transport of selected radionuclides in the abstraction is retarded by sorption onto stationary corrosion products (SAR Section 2.3.7.12). The applicant treated the corrosion products domain as a single mixing cell containing a homogenous porous medium with no preferential flow paths, on the basis of a conceptualization of degraded waste form disseminated within a corrosion product mass. The NRC staff finds that the applicant's conceptualization and TSPA implementation as a mixing cell provide appropriate representations of radionuclide release because a disseminated radionuclide source supports the underlying assumption of uniform radionuclide concentrations within the represented volume.

In the applicant's abstraction, corrosion product surface area is used to calculate the volume of adsorbed water and the mass of radionuclides sorbed onto corrosion products. The applicant assumed a mixture of hydrous ferric oxide and goethite for calculating corrosion product surface area without considering the aging of hydrous ferric oxide to more crystalline iron oxides with lower surface area. The applicant summarized additional sensitivity information in DOE (2009ay, Enclosure 6) demonstrating that the result of the abstraction is not significantly affected by the assumed relative abundance of hydrous ferric oxide. NRC staff reviewed the sensitivity analysis and finds that the abstraction adequately represents uncertainty in the corrosion product surface area.

The abstraction calculates the corrosion product surface area as a function of time using an uncertainty distribution of stainless steel corrosion rates based on literature data. On the basis of additional information the applicant provided in BSC (2004ae) and DOE (2009db, Enclosure 2) and an independent literature review, the NRC staff finds that the range of stainless steel corrosion rates that were used to simulate growth of the corrosion products that provide the sorption substrate is appropriate because it is consistent with literature corrosion rate data (Beavers and Durr, 1990aa; Glass, et al., 1984aa; McCright, et al., 1987aa).

The applicant modeled sorption on corrosion products for uranium, neptunium, thorium, americium, and plutonium using a surface complexation model to develop effective distribution coefficients (K_ds) taking into account the chemical conditions. A surface complexation model simulates equilibrium attachment of dissolved ions onto solid surfaces and incorporates the chemical complexity of the system. Other radioelements that are tracked in the unsaturated zone transport abstraction (SAR Section 2.3.8)—such as cesium, protactinium, radium, selenium, strontium, tin, technetium, iodine, chlorine, and carbon—are assumed in the applicant's engineered barrier system transport abstraction to not sorb onto stationary corrosion products. Nickel is included in the surface complexation model to represent competition for sorption sites, but the sorbed mass of nickel is not explicitly tracked for transport purposes.

The surface complexation model is not directly incorporated in the applicant's TSPA model abstraction, but the distribution coefficients developed from the surface complexation model are directly applicable to the transport of uranium, neptunium, and thorium, which undergo rapid and reversible sorption (SNL, 2007aj). Kinetic reversible sorption is modeled for americium and plutonium sorption on stationary corrosion products. The forward sorption rate constant in this case is sampled from an uncertainty distribution. The desorption rate constants are then calculated from this forward rate and the K_ds calculated using the surface complexation model.

To develop the K_d distributions used in the TSPA model abstraction, the applicant carried out the surface complexation models using the PHREEQC geochemical software the U.S. Geological Survey developed. To represent parameter uncertainty in the surface complexation model, the applicant conducted about 5,000 PHREEQC simulations, each with a unique combination of surface properties and aqueous chemistry parameters as inputs. The applicant then analyzed the PHREEQC simulation results using multiple regressions to produce functions that calculated actinide sorption as a function of key geochemical properties. These functions provide the K_d values used in the TSPA model abstraction of sorption to stationary corrosion products. The NRC staff reviewed the applicant's approach and determined that it is consistent with the NRC staff's independent evaluations of radionuclide sorption (e.g., Leslie, et al., 2007aa, and references therein). The NRC staff finds that the applicant considered uncertainty for appropriate key geochemical properties, such as pH and pCO_2, and surface properties related to sorption site concentration such as site density, surface area, and solid mass.

The applicant compared the calculated surface complexation model results from about 5,000 simulations to U.S. Environmental Protection Agency (EPA) compilations of soil K_ds obtained in the laboratory (SAR Section 2.3.7.12.3.4; SNL, 2007aj). The applicant compared values over a pH range from 6 to 9 (SAR Section 2.3.7.12.3.4), and although the ranges are broad, there was general agreement between the applicant's estimated values and the EPA compilation over the pH range considered. The applicant provided reasons for potential discrepancies at pH <7, but the NRC staff observed that there was a consistent bias of calculated K_d values to the high end of the EPA ranges at pH >8. For example, for moderately sorbing radioelements such as neptunium and uranium, the calculated K_d values for $8 < pH < 9$ were about one to three orders of magnitude above the lower limit for the EPA compilation at similar pH. In DOE (2009ay, Enclosure 9), the applicant provided additional experimental data that show actinides sorb more strongly onto hematite (an iron oxide) than onto clays and silicate minerals that tend to be more common in the soils assessed in the EPA compilation. In addition, analyses in DOE (2009da, Enclosure 3) show a narrower range with a lower maximum

K_d value generated in the TSPA model for all five radionuclides simulated using the surface complexation model. In effect, this TSPA approach reduces the maximum mass of actinides retained by sorption to stationary corrosion products from what might be expected from the surface complexation model results alone, increasing radionuclide concentration in the water and, all else being equal, increasing dose. The NRC staff finds that, although there are differences, the applicant's sorption model results are of the same order of magnitude with respect to actinide-iron oxyhydroxide sorption coefficients reported in the scientific literature. The NRC staff finds that this consistency supports the acceptability of the K_d values used in the applicant's TSPA model abstraction.

The applicant used several different approaches in developing its final TSPA abstraction for sorption to stationary corrosion products and considered and evaluated several different surface complexation model approaches before selecting the diffuse-layer model, as outlined in DOE (2009da, Enclosure 3). This modeling approach is widely used in the technical community, and the applicant discussed its advantages and disadvantages relative to other surface complexation models. The NRC staff finds that the surface complexation model, as the applicant implemented it, is consistent with established geochemical modeling principles; uses available experimental data to constrain chemical parameters; and is thus acceptable.

In SNL (2007aj, Section 6.5.2.4.2), the applicant stated that aqueous thermodynamic data were propagated through the surface complexation model, but did not explicitly consider uncertainty in equilibrium constants for surface complexation constants used in the surface complexation model. The NRC staff finds the exclusion of this source of uncertainty to be acceptable because the effects of these uncertainties on calculated radionuclide sorption would be small given the large excess number of available sorption sites as compared to aqueous radionuclide concentrations in all realizations. The large number of sorption sites in the surface complexation model for the corrosion products domain leads to the calculation of very high actinide K_d values, such that any additional uncertainty from surface complex thermodynamic data would have an insignificant effect on the transport model results. The NRC staff finds that, as described previously, the applicant provided an acceptable evaluation of the uncertainties associated with parameters that control the number of sorption sites, such as surface area, mass of corrosion products, and site density.

The applicant's abstraction assumes a single-rate, first-order kinetic model for plutonium and americium sorption. The forward-rate constants for sorption of americium and plutonium onto corrosion products were estimated from plutonium sorption experiments in SNL (2008ak). Although experiments on plutonium sorption onto iron oxide colloids have been shown to be inconsistent with a single-rate model (SNL, 2007aj; Painter, et al., 2002aa), the applicant provided additional information in DOE (2009ay, Enclosure 8) showing that sorption in the corrosion products domain is approximately an equilibrium process because water residence times are long compared with characteristic times for sorption. The NRC staff finds the single-rate kinetic sorption model acceptable because the near-equilibrium condition for sorption means that details of the kinetic model have negligible effects on transport model results.

Colloid-Facilitated Transport

In the applicant's abstraction, the transport of selected radionuclides may be enhanced by the five colloid types (SAR Section 2.3.7.12, SER Section 2.2.1.3.4.3.4). The applicant's mathematical models for colloid-assisted transport with reversible and irreversible sorption, with and without kinetic limitations, are consistent with approaches established in the scientific literature (e.g., Corapcioglu and Jiang, 1993aa) and the NRC staff's independent analyses

(Cvetkovic, et al., 2004aa; Painter and Cvetkovic, 2006aa). The NRC staff, therefore, finds the applicant's mathematical models for colloid-assisted transport in the engineered barrier system acceptable. Because of limited data about colloid retardation or physical straining in the engineered barrier system, the NRC staff finds that the applicant appropriately excluded these retention processes when considering colloid migration in the engineered barrier system.

The applicant's abstraction considers plutonium and americium to be irreversibly associated with plutonium and zirconium colloids that originate from commercial SNF and montmorillonite colloids originating from defense high-level waste glass. Uranium is irreversibly associated with the uranophane colloids that originate in SNF. These three colloid types and the associated radionuclides originate in the waste form. Selected radionuclides (isotopes of americium, uranium, neptunium, cesium, protactinium, plutonium, radium, tin, and thorium) are allowed to sorb reversibly and without kinetic limitations onto uranophane colloids and groundwater or waste-derived montmorillonite colloids. The empirical sorption model represents equilibrium, site-limited sorption with competition for sites. Although the assumed equilibrium distribution coefficients for plutonium are higher than the supporting data, the NRC staff finds the abstraction acceptable because kinetically limited desorption is required to enhance transport for the range of colloid concentrations considered (Cvetkovic, et al., 2004aa; Painter and Cvetkovic, 2006aa), and kinetic limitation is not modeled or expected for these colloid types. The NRC staff finds that the assumed equilibrium sorption model is consistent with NRC staff understanding of sorption on silicate minerals and is thus appropriate for the application.

In the applicant's abstraction, uranium, neptunium, and thorium sorb reversibly and without kinetic limitations to corrosion product colloids. The applicant calculated equilibrium distribution coefficients (K_ds) for these elements using the same surface complexation modeling approach as for sorption on stationary corrosion products. As described in the previous subsection, the surface complexation model simulates equilibrium attachment of dissolved ions onto solid surfaces (in this case, the surfaces of mobile colloids), incorporating the chemical complexity of the system. The applicant's abstraction does not include corrosion product colloids in the waste form domain. Kinetic irreversible sorption onto corrosion product colloids is modeled for americium and plutonium in the corrosion products domain. The NRC staff finds that the applicant's mathematical model for colloid-associated radionuclide transport with kinetic transfers is acceptable because it is consistent with the applicant's process conceptualization, models established in the literature (e.g., Corapcioglu and Jiang, 1993aa), and the NRC staff's independent analyses (Cvetkovic, et al., 2004aa; Painter and Cvetkovic, 2006aa).

In the applicant's abstraction, the forward-rate constants for americium and plutonium are not sampled directly. Instead, the fraction of total mobile radionuclide mass exiting the corrosion products domain that is associated with colloids (the target flux-out ratio) is sampled from a uniform distribution with a range of 0.9 to 0.99. An analytical inverse solution is then used to calculate the forward rate corresponding to the sampled ratio. In the model abstraction, the computed forward rate constant and other parameters computed from the inverse solution are further compared against physically allowed ranges. If any computed value is outside its allowed range, the corresponding maximum or minimum value is used in the forward model for colloid-assisted transport in place of the sampled ratio. The NRC staff finds the approach based on a target flux-out ratio to be acceptable because, for conditions under which colloids are stable and could potentially affect transport, this approach tends to enhance radionuclide release over that expected from a mechanistic process model.

The applicant stated that colloids will not significantly enhance transport of radionuclides in the engineered barrier system because commercial SNF-derived colloids are expected to be unstable in the corrosion products domain and because radionuclide concentrations associated with iron oxide and high-level waste glass form colloids are expected to be smaller than dissolved concentrations. The NRC staff finds the applicant's conclusions acceptable because (i) the applicant's analyses showed that commercial SNF-derived colloids are not expected to be stable in the pH ranges expected in the corrosion products domain (DOE, 2009ay), (ii) the NRC staff's confirmatory calculations (SER Section 2.2.1.3.4.3.4) show that iron oxide colloids exert negligible control on dissolved plutonium concentrations, and (iii) the mean sampled plutonium concentration associated with high-level waste glass colloids is an order of magnitude lower than the mean plutonium solubility limit (SER Section 2.2.1.3.4.3.4).

TSPA Results for Engineered Barrier System Radionuclide Releases

The NRC staff performed simplified confirmatory analyses to assess whether the applicant's TSPA results for the engineered barrier system radionuclide releases are consistent with the applicant's abstractions. As discussed in the following paragraphs, the NRC staff used confirmatory calculations (Painter, 2010aa; Pickett, 2010aa) to estimate peak expected total-repository engineered barrier system release rates for Tc-99, Pu-242, and Np-237, using applicant-provided information, and then compared these releases with applicant-provided values. The NRC staff's confirmatory calculations used parameters and probabilities the applicant provided in the SAR (e.g., SAR Section 2.1.2.2.6), data from supporting reports (e.g., SNL, 2007ah, 2008ag), and associated responses to requests for additional information (e.g., DOE, 2009 da,dc).

The NRC staff's confirmatory calculations focused on the igneous intrusion and seismic ground motion modeling cases because these cases result in the largest release rates from the engineered barrier system. These estimates are based on advection and diffusion of dissolved radionuclides and neglect transport of radionuclides associated with colloids. As discussed in SER Section 2.2.1.3.4.3.4, transport in the engineered barrier system is not significantly enhanced by colloids because of low colloid concentrations in the corrosion products domain. The NRC staff's simplified calculation estimates are presented only for selected cases for which representative applicant results were available for comparison. Results are presented for waste package release rates (i) of Pu-242 and Np-237 in the igneous intrusion modeling case, on the basis of control of the dissolved concentration limited by solubility (SER Section 2.2.1.3.4.3.3) or corrosion product sorption (SER Section 2.2.1.3.4.3.5) and (ii) of Tc-99 in the seismic ground motion modeling case, on the basis of release rate control by waste package failure.

The applicant provided information showing that engineered barrier system releases of low-solubility, sorbing radionuclides (e.g., plutonium and neptunium) are mainly controlled by processes within the corrosion products domain because waste form dissolution and invert transport processes are fast, relative to transport within the corrosion products domain. In the TSPA analyses, the important dose contributions from plutonium and neptunium isotopes result from the igneous intrusion modeling case (SAR Section 2.4.2.2.1.1.3), in which all waste packages fail and releases of these radionuclides are controlled by advection modified by sorption and precipitation of radionuclide-bearing minerals. The NRC staff performed simplified estimates to confirm the applicant's release calculations for Pu-242 and Np-237 for the igneous intrusion modeling case in the engineered barrier system. The NRC staff assumed that these release rates for the corrosion products domain are controlled by advection and either (i) precipitation of solubility-limiting minerals or (ii) sorption onto corrosion products. The solubility limit (SER Section 2.2.1.3.4.3.3) is a chemically-based maximum value for the dissolved concentration of an element, in the absence of sorption. If, however, there is capacity for sorbing the dissolved element onto solid surfaces, the dissolved concentration may not reach the solubility limit and could be controlled to a lower value by sorption (SER Section 2.2.1.3.4.3.5). Because both solubility and sorption are viable processes in the corrosion products domain, the better estimate of release rate is obtained by calculating release based on the process that limits the dissolved concentration to the lowest value. The NRC staff, therefore, used simplified calculations to estimate release rates controlled by both potential limits on dissolved concentration—solubility and sorption—and selected the lower value of the two for comparison to the applicant's results.

Using information provided by the applicant, the NRC staff estimated a peak mean, repositorywide, sorption-limited Pu-242 engineered barrier system release rate of 7.9 g/yr [0.017 lb/yr] (Painter, 2010aa; Pickett, 2010aa). The NRC staff calculated a solubility-limited Pu-242 release rate of 1.2 g/yr [0.0026 lb/yr] (Painter, 2010aa; Pickett, 2010aa), which is lower than the sorption-limited rate and is, therefore, taken as the estimated value.

The applicant provided representative commercial SNF package release results for percolation subarea 3 (which includes approximately 40 percent of the waste packages) and a single realization from the igneous intrusion case in DOE (2009da, Enclosure 1, Figure 2). For Pu-242, the peak corrosion products domain release from subarea 3 is approximately 30 g/yr [0.07 lb/yr] at 100,000 years, conditional on an igneous event having occurred at 10,000 years. Scaling that value to the full repository and multiplying by the probability of having a single igneous intrusion event in 1 million years (1.7 percent) gives a value of 1.3 g/yr [0.0029 lb/yr]. The NRC staff's confirmatory estimate of 1.2 g/yr [0.0026 lb/yr] is consistent with this applicant result, which is, therefore, acceptable.

The NRC staff calculated an estimated peak mean, repositorywide, sorption-limited release rate for Np-237 of 3.0 g/yr [0.0066 lb/yr] for the igneous intrusion modeling case (Painter, 2010aa; Pickett, 2010aa). The applicant provided representative results for Np-237 release from the corrosion products domain in percolation subarea 3 and a single realization of the igneous intrusion modeling case in DOE (2009dc, Figure 1.1-12). That realization assumes that an igneous intrusion event occurs. Thus, the applicant's calculated peak release rate of approximately 200 g/yr [0.4 lb/yr] must be weighted by the probability of an igneous event in 1 million years (1.7 percent) and scaled to the full repository to compare with the NRC staff's peak mean estimate. The result of that calculation is 8.5 g/yr [0.019 lb/yr] for Np-237 release for the igneous intrusion modeling case. The NRC staff's simplified calculation estimate of 3.0 g/yr [0.0066 lb/yr] is in reasonable agreement with this applicant result, which is, therefore, acceptable.

The NRC staff calculated an NpO_2 solubility-limited neptunium-engineered barrier system release rate of 12 g/yr [0.026 lb/yr] (Painter, 2010aa, Pickett, 2010aa). This result is higher than the sorption-limited rate and was, therefore, not used for comparison; as discussed previously in this section, the lower calculated rate is more appropriate for comparisons. A rate calculated using the Np_2O_5 solubility model (applicable after all in-package steel has corroded) would be even higher and was, therefore, not used for comparison.

For high-solubility radionuclides that are weakly sorbing or nonsorbing (e.g., technetium), radionuclide-specific engineered barrier system releases in the seismic ground motion and nominal modeling cases are controlled primarily by the waste package failure rate because dissolution rates of the waste form and transport of these radionuclides through the engineered barrier system are sufficiently fast compared with typical intervals between package failures (SER Section 2.2.1.4.1.3.3.1.2). Using the information provided by the applicant, the NRC staff calculated a peak mean, repositorywide release rate for Tc-99 of 22 g/yr [0.049 lb/yr] for the seismic ground motion modeling case (Painter, 2010aa; Pickett, 2010aa). The applicant provided plots of mean cumulative Tc-99 engineered barrier system release in SAR Figure 2.1-24. From the slopes of these plots, the NRC staff estimated that the release rate peaks at approximately 12 g/yr [0.026 lb/yr] for the period between 10,000 and 100,000 years. This peak mean rate estimated from the applicant's Tc-99 release information is in reasonable agreement with the NRC staff's simplified calculated estimate of 22 g/yr [0.049 lb/yr] (Painter, 2010aa; Pickett, 2010aa), and is, therefore, acceptable.

Summary of NRC Staff's Review of Engineered Barrier System Radionuclide Transport

The NRC staff finds that, in modeling the transport of radionuclides in the engineered barrier system, the applicant adequately described the system and models used; applied appropriate conceptual models; and considered alternative conceptual models. The applicant used appropriate mathematical models to represent transport in the engineered barrier system. Transfer of information between the radionuclide transport abstraction and other TSPA code abstractions was consistently and appropriately implemented. Relevant design information for the waste package was appropriately incorporated. The applicant used appropriate data to establish model parameters and to represent uncertainty. Intermediate results of the abstraction were appropriately compared to independent information.

TSPA release rates for radionuclide transport in the engineered barrier system vary significantly by radionuclide and modeling case. The engineered barrier system does not significantly delay transport of soluble, nonsorbing radionuclides, such as Tc-99 and I-129, and the waste package failure rates control the engineered barrier system release rates for those radionuclides. Transport of low-solubility, sorbing radionuclides, such as Np-237 and Pu-242, is significantly slower and is generally controlled by sorption onto stationary corrosion products and precipitation of radionuclide-bearing minerals in the corrosion products domain. Colloid-assisted transport is not significant compared with transport of dissolved radionuclides because of limited colloid concentrations in the engineered barrier system. The NRC staff finds that the TSPA code results for the engineered barrier system release rates are consistent with the NRC staff's simplified confirmatory calculations, confirming the appropriateness of the TSPA results.

2.2.1.3.4.4 Evaluation Findings

The NRC staff has reviewed the applicant's SAR and other information submitted in support of the license application, which includes information required by 10 CFR 63.21(c)(3),(9),(15), and finds, with reasonable expectation, that the relevant requirements of 10 CFR 63.114 and 10 CFR 63.342 are satisfied regarding the abstraction of radionuclide release rates and solubility limits. In particular, the NRC staff finds that the applicant has adequately

- Included appropriate data related to the geology, hydrology, and geochemistry of the surface and subsurface from the site and the region surrounding Yucca Mountain and provided adequate information on the design of the engineered barrier system to define parameters and conceptual models used in the performance assessment calculation, in compliance with 10 CFR 63.114(a)(1)

- Accounted for uncertainty and variability in the parameter values used to model radionuclide release rates and solubility limits, in compliance with 10 CFR 63.114(a)(2)

- Considered and evaluated alternative conceptual models that are consistent with currently available data and scientific understanding, in compliance with 10 CFR 63.114(a)(3)

- Provided a technical basis for the inclusion of FEPs affecting radionuclide release rates and solubility limits, consistent with the limits on performance assessment in 10 CFR 63.342, and evaluated in sufficient detail those processes that would significantly affect repository performance, in compliance with 10 CFR 63.114(a)(4-6)

- Provided technical bases for the models of radionuclide release rates and solubility limits used in the performance assessment to represent the 10,000 years after disposal, in compliance with 10 CFR 63.114(a)(7)

- Used performance assessment methods for the period of geologic stability consistent with the methods used to demonstrate compliance for the initial 10,000 years following permanent closure, in compliance with 10 CFR 63.114(b)

- Included in the performance assessment for the period of geologic stability those FEPs used for the initial 10,000-year period, consistent with specific limits, in compliance with 10 CFR 63.342(c)

2.2.1.3.4.5 References

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CHAPTER 8

2.2.1.3.5 Climate and Infiltration

2.2.1.3.5.1 Introduction

Safety Evaluation Report (SER) Section 2.2.1.3.5 provides the U.S. Nuclear Regulatory Commission (NRC) staff's evaluation of the U.S. Department of Energy's ("DOE" or "applicant") representation of climate and infiltration, as presented in the applicant's Safety Analysis Report (SAR) Section 2.3.1 (DOE, 2008ab), relevant references, and DOE responses to staff's requests for additional information (RAIs). DOE considers the reduction of water flux from precipitation to net infiltration to be a barrier capability for the Upper Natural Barrier. Because of the generally vertical movement of percolating water through the unsaturated zone in the DOE representation of the natural system, water entering the unsaturated zone at the ground surface (infiltration) is the only source for deep percolation water in the unsaturated zone at and below the proposed repository.

DOE used the term "net infiltration" to define the volumetric flux of water passing below the active plant root zone, but often refers to net infiltration simply as "infiltration." DOE also used the term net infiltration to refer both to the output of the net infiltration model (SAR Section 2.3.1) and to the top boundary condition of the unsaturated zone model (SAR Section 2.3.2). This distinction is important because the average values from the net infiltration model differ from those used as net infiltration at the top boundary of the site-scale unsaturated zone model. NRC staff evaluates the former in the present section and the latter in SER Section 2.2.1.3.6.3.2.

Climate and infiltration are treated differently in DOE's performance assessment for the initial 10,000 years of the repository and the period from 10,000 to 1 million years. For the initial 10,000 years, DOE used paleoclimate records for the region to predict future climatic conditions and uses these predictions as input for estimating future net infiltration. In addition, DOE used the climate predictions to scale groundwater fluxes in the saturated zone portion of the performance assessment for this period. DOE described its approach for scaling the groundwater flux in SAR Section 2.3.9; the NRC staff's review of groundwater flux in the saturated zone is in SER Section 2.2.1.3.8. For the period from 10,000 years to 1 million years after disposal, 10 CFR 63.342(c)(2) allows the applicant to consider long-term-average deep percolation flux at the proposed repository horizon instead of explicitly predicting climate and infiltration. DOE chose to use the prescribed deep percolation flux in its performance assessment for the post-10,000-year period. In SER Section 2.2.1.3.6.3.2, NRC staff evaluates the DOE implementation of 10 CFR 63.342 with respect to average deep percolation flux at the proposed repository horizon for post-10,000-year performance assessment calculations.

DOE used several descriptions of the initial 10,000-year repository period in the SAR. The NRC staff determines that the difference between the phrase *the next [or initial] 10,000 years* and the phrases *10,000 years after (or following) disposal, emplacement, closure, or permanent closure* does not significantly affect estimates of climate and net infiltration, because the difference in the time period is small relative to the period of consideration.

This SER Section provides the NRC staff's evaluation of DOE's consideration of climate and infiltration in the initial 10,000 years after disposal in DOE's Total System Performance Assessment calculations. NRC staff reviewed the DOE technical bases, input data, models, and net infiltration results against the applicable regulations. The NRC staff used its understanding of relative risk within the repository system to inform its review, by focusing on those aspects that are most significant for repository performance. In determining the significant aspects of DOE's net infiltration model, NRC staff considered how the flux of water through the unsaturated zone affects (i) seepage (flux of water dripping into drifts), (ii) release of radionuclides from the engineered barrier system, and (iii) radionuclide transport through the natural system (evaluated in SER Sections 2.2.1.3.6.3.4, 2.2.1.3.4.3.5, and 2.2.1.3.7, respectively). On the basis of the downstream uses of climate and infiltration calculations, NRC staff focused its review on DOE's estimates of the magnitude, spatial distribution, and uncertainty of net infiltration over the initial 10,000-year period.

2.2.1.3.5.2 Regulatory Requirements

The applicant is required to provide information specified in 10 CFR 63.21(c)(1), (9), (10), and (15) that is related to the abstraction of climate and infiltration. The requirements in 10 CFR 63.114 (Requirements for Performance Assessments) and 10 CFR 63.342 (Limits on Performance Assessments) include postclosure performance assessment to demonstrate compliance with 10 CFR 63.113 (Performance Objectives for the Geologic Repository after Permanent Closure). Compliance with 10 CFR 63.113 is reviewed in SER Section 2.2.1.4.1.

The regulations in 10 CFR 63.114 require, in part, that a performance assessment

- Include data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the surface and subsurface from the site and the region surrounding Yucca Mountain [10 CFR 63.114(a)(1)]

- Account for uncertainty and variability in the parameter values [10 CFR 63.114(a)(2)]

- Consider and evaluate alternative conceptual models [10 CFR 63.114(a)(3)]

- Provide technical bases for either the inclusion or exclusion of features, events, and processes (FEPs), including effects of degradation, deterioration, or alteration processes of engineered barriers that would adversely affect performance of the natural barriers, consistent with the limits on performance assessment in 10 CFR 63.342, and evaluate in sufficient detail those processes that would significantly affect repository performance [10 CFR 63.114(a)(4–6)]

- Provide technical basis for the models used in the performance assessment to represent the 10,000 years after disposal [10 CFR 63.114(a)(7)]

The NRC staff's evaluation of inclusion or exclusion of FEPs is presented in SER Section 2.2.1.2.1.

The requirements in 10 CFR 63.342(c)(2) pertain to the use of specified constant-in-time deep percolation rates to account for the effects of climate change on performance for the period from 10,000 to 1 million years after disposal. The NRC staff's evaluation of the applicant's use of these deep percolation rates is given in SER Section 2.2.1.3.6.3.2. In addition, the requirements in 10 CFR 63.342(c)(1) pertain to the effects of seismic and igneous activity on repository performance, subject to the probability limits in 10 CFR 63.342(a) and 10 CFR 63.342(b). Specific constraints on the analysis required for seismic and igneous activity are given in 10 CFR 63.342(c)(1)(i) and 10 CFR 63.342(c)(1)(ii).

The following requirements for characteristics of the reference biosphere to be used in this abstraction for climate and infiltration are specified in 10 CFR 63.305:

- FEPs that describe the reference biosphere must be consistent with present knowledge of the conditions in the region surrounding the Yucca Mountain site. [10 CFR 63.305(a)]

- DOE should not project changes in society, the biosphere (other than climate), or human biology or increases or decreases of human knowledge and technology; in all analyses done to demonstrate compliance with this part, DOE must assume that all of those factors are constant as they are at the time of submission of the license application. [10 CFR 63.305(b)]

- DOE must vary factors related to the geology, hydrology, and climate based upon cautious but reasonable assumptions of the changes in these factors that could affect the Yucca Mountain disposal system during the period of geologic stability [10 CFR 63.305(c)], which for the climate and infiltration abstraction is limited to the initial 10,000 years after disposal

- Biosphere pathways must be consistent with arid or semi-arid conditions. [10 CFR 63.305(d)]

The requirements of 10 CFR 63.305 apply to the abstraction reviewed in this SER Section to the extent that the characteristics of the reference biosphere affect climate and infiltration.

The NRC staff's review of the SAR and supporting information follows the guidance in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa; Section 2.2.1.3.5, Climate and Infiltration). The YMRP acceptance criteria for model abstractions that provide guidance for the NRC staff's review of the applicant's abstraction of climate and infiltration are

1. System description and model integration are adequate
2. Data are sufficient for model justification
3. Data uncertainty is characterized and propagated through the abstraction
4. Model uncertainty is characterized and propagated through the abstraction
5. Model abstraction output is supported by objective comparisons

In its review of the SAR and supporting information, the NRC staff used a risk-informed approach and guidance in the YMRP for aspects of climate and infiltration that are important to repository performance. The NRC staff considered all five criteria provided in the YMRP in its review of information provided by DOE. In the context of these criteria, only those aspects of the model abstraction that substantively affect the performance assessment results, as determined by the NRC staff, are discussed in detail in this SER section. The NRC staff's determination is based both on risk information provided by DOE and on the NRC staff's independent analyses and knowledge gained through experience.

2.2.1.3.5.3 Technical Review

The review of the technical information DOE provided for climate and infiltration for the initial 10,000-year period is divided into three subsections within this SER section. The first subsection reviews the applicant's identification and description of features and processes for climate and infiltration. The second subsection focuses on the climate data, future-climate model, and climate predictions. The third subsection addresses the applicant's description of net infiltration processes, models, and estimates of net infiltration at Yucca Mountain using the future climate predictions.

2.2.1.3.5.3.1 Identification of Features and Processes

In this section, NRC staff reviews the DOE identification and description of processes important for estimating climate and net infiltration. This section addresses YMRP acceptance criteria related to system description and model integration. DOE's overall screening of FEPs is reviewed in SER Section 2.2.1.2.1.

DOE used regional and site characteristics to develop conceptual models for climate and net infiltration at Yucca Mountain. The natural features of topography and surficial soils of the Upper Natural Barrier were identified as important to waste isolation in SAR Section 2.1.2.1. On the basis of field observations, synthesis of data, and modeling over more than two decades, DOE indicated (SAR Section 2.3.1.1) that the features and processes important to the capability of the Upper Natural Barrier are (i) climate change, (ii) climate modification increases recharge, (iii) precipitation, (iv) topography and morphology, (v) rock properties of host rock and other units, (vi) surface runoff and evapotranspiration, (vii) infiltration and recharge, (viii) fractures, and (ix) fracture flow in the unsaturated zone.

The following summary is based on the information in SAR Section 2.3.1.1 and illustrates how DOE integrated these features and processes in its conceptual models of climate and infiltration. DOE described the present climate at Yucca Mountain as semi-arid, with low annual precipitation. DOE expects the climate to change over the initial 10,000 years, remaining semi-arid but with changes in precipitation patterns and rates. DOE recognized that surface temperature and vegetation will also vary with changes in climate. Evapotranspiration (the combination of evaporation and plant transpiration) removes a large portion of the annual precipitation that infiltrates into the soil. In this environment, evapotranspiration is strongly influenced by temperature and low atmospheric relative humidity. In DOE's conceptual model, net infiltration events occur in pulses during and for a short period following some of the larger or longer duration precipitation events. Evapotranspiration continually dries the soil between precipitation events. DOE considered snow as providing a source of delayed infiltration during snowmelt events. In DOE's conceptual model, runoff and the soil's water-holding capacity limit the magnitude of net infiltration pulses, but runoff from one area may subsequently infiltrate downstream.

DOE developed the conceptualization that soil, fractures, and bulk rock hydraulic properties affect the rate of water movement below the root zone, with a competition between downward flow and upward movement of water by evapotranspiration. In the DOE conceptual model, water flows quickly into the rock below shallow soil in areas where the bedrock is fractured or has a highly permeable matrix. Such rapid flow limits the effect of evapotranspiration. Surface water runoff, influenced by topography and surface morphology, spatially redistributes the flux of water. This process may reduce net infiltration in some areas (e.g., high on hillslopes) and increase net infiltration in others (e.g., washes and channels). DOE recognized that lateral movement of water below the ground surface, termed interflow, is known to spatially redistribute water in the soils. However, the DOE conceptual model does not include interflow, because DOE determined that it was not significant at Yucca Mountain (SNL, 2007az, Section 5.1). For the semi-arid climate of Yucca Mountain, the overall water balance in DOE's model is dominated by precipitation and evapotranspiration, with infiltration and runoff representing relatively small portions of the balance.

DOE implemented its conceptualization in (i) a climate model that predicts future climatic states, (ii) climatic input data for each climate state, and (iii) a net infiltration model linked to a surface-water-routing algorithm for runoff. The infiltration and routing algorithms are integrated into the Mass Accounting System for Soil Infiltration and Flow (MASSIF) model. DOE described

- A climate model for predicting climate over the initial 10,000 years that uses Earth-orbital parameters and paleoclimatic data from the southwestern United States covering the past approximately 800,000 years

- Climatic input data for each climate state that uses recorded meteorological data from local, regional, and western U.S. stations

- Submodels of the net infiltration model that consider precipitation, evapotranspiration, snowmelt, runoff and run on, and infiltration

DOE used site characterization data, as available, to develop inputs for the MASSIF model (SAR Section 2.3.1.3). Where sparse or no site observations are available, other information from scientific literature or other sites was used to develop inputs.

NRC staff concludes that DOE adequately identified and included features and processes in its climate models that are important for estimating future climatic conditions at Yucca Mountain. This conclusion is supported by a comparison of the information provided by DOE with staff's knowledge of past climates in the southwest United States, including the Yucca Mountain region, obtained from literature reviews and independent analysis (NRC, 2005aa; Stothoff and Musgrove, 2006aa; Stothoff and Walter, 2013aa). NRC staff compared the applicant's description of infiltration, and the incorporation of features and processes, with NRC staff's understanding of near-surface features and processes at Yucca Mountain obtained from literature reviews, field observations, and independent analysis (NRC, 2005aa; Stothoff, 2013aa,ab, 2008aa,ab, 2009aa). Because DOE's description of infiltration and incorporation of features and processes are consistent with those from other sites and with staff's independent analyses (see references in previous sentence), NRC staff concludes that DOE adequately identified and included in its overall conceptual model features and processes important for net infiltration at Yucca Mountain.

2.2.1.3.5.3.2 Climate

This section contains NRC staff's review of the DOE approach and results for predicting climate states for the initial 10,000-year period, and for predicting climatic conditions within each of the climate states. NRC staff evaluated the performance assessment calculations representing the initial 10,000 years of repository performance to assess DOE's net infiltration estimates. In its performance-based review, NRC staff focused on identifying whether the data, models, and results adequately represent climate and the uncertainty of predicting future climate conditions. Because DOE chose to use the range and distribution of average deep percolation specified in 10 CFR 63.342(c) for the period from 10,000 to 1 million years after closure, the applicant did not have to provide information on climate or meteorology during this period. Therefore, consistent with 10 CFR 63.342(c)(2), there is no NRC staff evaluation for climate or meteorology for the post-10,000-year period in its SER. The NRC staff evaluates the DOE approach to representing deep percolation during the period from 10,000 to 1 million years after closure in SER Section 2.2.1.3.6.3.2.

The NRC staff evaluates DOE's approach to modeling climate during the initial 10,000 years by separately considering the applicant's approach to estimating long-term-average climate (SER Section 2.2.1.3.5.3.2.1) and the applicant's approach to estimating daily weather parameters given a long-term-average climate state (SER Section 2.2.1.3.5.3.2.2). These two sections separate climatic considerations into long and short time scales, respectively.

2.2.1.3.5.3.2.1 Climate Change for the Initial 10,000 Years

This section addresses YMRP acceptance criteria related to data for model justification, and characterization and propagation of data uncertainty, for the applicant's climate prediction for the initial 10,000 years.

DOE predicted climate states covering the initial 10,000 years using paleoclimate proxies from regional records and the understanding of orbital variations as the principal drivers of Earth climate over the past several million years. DOE's main paleoclimate proxies are layered mineral precipitates from Devils Hole and fossils preserved in continuously layered lake sediments from Owens Lake. Owens Lake, a present-day playa, and Devils Hole, a water-filled cave, are both within 140 km [87 mi] of Yucca Mountain. Cores from both sites record past cyclic changes in regional climate, between glacial and interglacial phases, and are generally consistent with other global climate proxy records.

From these records, the applicant derived three representative states for future climate. DOE predicted these three climates and the timing of step changes by (i) identifying the past point in time in the Devils Hole record that is equivalent to the present moment within the glacial cycle, (ii) identifying the same equivalent point in the Owens Lake sediment sequence, (iii) identifying the sediment sequence corresponding to the 10,000 years following the equivalent point, and finally (iv) attributing climate states to the sediment sequence (SAR Section 2.3.1.2.3.1.1).

The climate-sequence timing DOE described is based on two Earth-orbital parameters, which are recognized as climate forcing functions operating over geologic time scales: orbital eccentricity, with a period of approximately 100,000 years, and precession of the equinoxes, with a period of approximately 23,000 years (SAR Section 2.3.1.2.1.2.3). Values for these orbital parameters can be calculated to high precision from astronomical relations. DOE used oxygen isotope ratio ($\delta^{18}O$) records in Devils Hole vein calcite, dated using uranium-series methods, to relate the orbital parameters to past glacial stages. In the SAR, DOE explained that obliquity, another recognized orbital forcing with a period of approximately 41,000 years, was not used in its model, because no consistent relationship was shown between obliquity and the Devils Hole $\delta^{18}O$ record. The SAR asserted that groups of four eccentricity cycles, totaling approximately 400,000 years, provide analogous repetitions of glacial cycles. DOE built confidence in the selection of the first 10,000 years of the cycle as an analog for the initial 10,000 years at Yucca Mountain by demonstrating that the last 400,000-year cycle was similar to the previous 400,000-year cycle (400,000 to 800,000 years ago).

DOE constructed past climates for particular glacial stages (SAR Section 2.3.1.2.1.2.4) using the Forester, et al. (1999aa) analysis of ostracode occurrences in lake sediment obtained from composite core OL-92, drilled in 1992 at Owens Lake, California, together with observed flows in Owens River for wet years. Ostracodes are microfossils. Different species of ostracodes have different environmental preferences for salinity and temperature. In DOE's analysis, the ostracode-based salinity of the paleo-Owens Lake serves as a proxy for annual precipitation. DOE used the abundance of five different species to infer compatible climatic parameters of temperature and seasonality and then developed future-climate parameters from meteorological stations with long records in locations where those species currently exist. DOE built confidence in the climate estimators using diatom records from the same core samples for periods when surface waters were not saline and diatoms were better preserved.

DOE's procedure yielded three representative climates for the initial 10,000 years after closure: (i) modern (present-day) climate for the first 600 years, (ii) monsoonal climate for 1,400 years, and (iii) glacial transitional climate for the remaining 8,000 years. Relatively speaking, these three climates can be described as hot and dry, hot and wet, and cool and wet, although all are classified as arid or semi-arid climates. DOE calculated sample-average mean annual precipitation values for the monsoon and glacial transition climates that were 1.59 and 1.63 times the sample-average mean annual precipitation for the present-day climate (SAR Tables 2.3.1-2 through 2.3.1-4). Also, DOE calculated nominal mean annual temperatures for the monsoon and glacial transition climates that were 0.9 °C [1.6 °F] warmer and 5.5 °C [9.9 °F] cooler than the nominal mean annual temperature for the present-day climate using values from SNL [2007az, Tables F-22 through F-24 and Eq. F-47(a)]. DOE's representation of a monsoonal climate also exhibited a shift in seasonality, with summer convection storms making up a larger fraction of its annual total precipitation than for either the present-day or glacial-transition climates.

The NRC staff determined that DOE provided a transparent and traceable description of its approach of using orbital cycles covering the past 800,000 years integrated with available paleoclimatic data to estimate the timing and duration of climates over the initial 10,000 years. The NRC staff compared the DOE description with staff's understanding (e.g., Stothoff and Walter, 2013aa) of paleoclimatic data and approaches for projecting future climates based on paleoclimatic information. On the basis of this comparison, the NRC staff concludes that DOE adequately incorporated features and processes important for using paleoclimate reconstructions in projecting future climates. The NRC staff concludes that the paleoclimatic information DOE used to develop the timing and duration of climates over the initial 10,000 years is adequately representative of Yucca Mountain because DOE obtained the paleoclimatic data from the region near Yucca Mountain.

DOE considered the two primary uncertainties in the data sets used to forecast future climates at Yucca Mountain to be the standard deviation associated with the Devils Hole ages and the uncertainty of the timing of climate change implied by the Devils Hole record (SAR Section 2.3.1.2.2.1.4). DOE also considered the uncertainties in model forecasts of future climates to include (i) uncertainty in the location of the past–present equivalency point in the Owens Lake record, (ii) uncertainty arising from the chaotic nature of the climate system, and (iii) uncertainty in selecting a particular past climate sequence to forecast the future (SAR Section 2.3.1.2.3.2).

From the DOE-identified primary uncertainties and the downstream uses of the DOE future-climate model, NRC staff identifies three specific aspects of the DOE future-climate model where uncertainties may have a potential effect on repository performance for the initial 10,000 years: (i) uncertainty in timing and duration of climate states used for performance assessment calculations, (ii) uncertainty in climatic conditions during the post-thermal-pulse period when temperatures near drifts drop below boiling (dominated by the glacial-transition climate state in the DOE future-climate model), and (iii) uncertainty in climatic conditions from anthropogenic activities.

Uncertainty in Timing and Duration of Climate States

DOE used three climate states (present-day, monsoon, and glacial-transition), each with a constant climate, to represent millennial-scale temporal variations found in the paleorecord. Uncertainty in the timing of transitions between the projected climate states could potentially impact estimates of future infiltration, which in turn may affect unsaturated-zone flow, seepage rates, thermal conditions near the repository, and radionuclide transport. The NRC staff concludes that the timing for the transition between the monsoon and glacial-transition climate states has low consequence on performance assessment results for the following reasons:

- The mean annual precipitation values for these two climate states are similar. DOE calculated sample-average mean annual precipitation values for the monsoon and glacial-transition climate states that were 1.59 and 1.63 times the sample-average mean annual precipitation for the present-day climate state (SAR Tables 2.3.1-2 through 2.3.1-4).

- The mean annual infiltration values for these two climate states are similar. DOE calculated weighted-average mean annual infiltration values over the repository footprint for the monsoon and glacial-transition climate states of 15.88 and 21.25 mm/yr [0.625 in/yr and 0.837 in/yr], respectively, as shown in DOE (2010ai, Enclosure 1, Table 1). The weighted-average glacial-transition mean annual infiltration was 1.34 times larger than the weighted-average monsoon mean annual infiltration.

- The consequence of a much larger difference in mean annual infiltration during any portion of the initial 10,000 years has little consequence for performance. DOE demonstrated that concurrently increasing areal-average mean annual infiltration by a factor of 2.39 for the monsoon state and 1.81 for the glacial-transition state has little effect on performance assessment results, as described in DOE (2009bo, Enclosure 5).

The NRC staff concludes that the timing of the transition between the present-day and monsoon climate states, as set by DOE at 600 years following repository closure, has low consequence in performance assessment calculations for the following reasons:

- The present-day climate state predominantly corresponds to above-boiling conditions within emplacement drifts, and SAR Section 2.3.3.1 asserted that seepage into drifts is not expected to occur where rock above the repository exhibits above-boiling temperatures or dryout conditions. DOE estimated that above-boiling conditions within emplacement drifts may persist for several hundred to more than 1,000 years, depending on emplacement drift location (SAR Section 2.3.3.3.3.1). NRC staff's evaluation in Section 2.2.1.3.6.3.4 concludes that dripping water will not likely reach the waste packages during above-boiling conditions.

- An early onset of a high-infiltration (monsoon or glacial transition) climate may reduce the duration of the thermal period. However, the duration of the post-thermal-pulse period affected by seepage cannot increase by more than 600 years out of the initial 10,000 years. The 600-year constraint comes from elimination of the present-day climate. NRC staff maintains that an increase of the duration with seepage from 9,400 years to 10,000 years would have a low consequence in performance assessment results.

- A delayed onset of a high-infiltration climate after the present-day climate state would result in smaller estimates of mean annual infiltration than are incorporated in the performance assessment.

The NRC staff concludes that the potential for an early transition to a full-glacial climate state has low probability of occurring during the initial 10,000 years and low consequence if the transition occurs after 10,000 years for the following reasons. DOE estimates a return to a full-glacial climate state 30,000 years (SAR Section 2.3.1.2.1.2.3) after permanent closure. On the basis of the paleoclimatic data from the region around Yucca Mountain, NRC staff concludes it is unlikely that the full-glacial climate would occur during the initial 10,000 years. Furthermore, a return of the full-glacial climate up to 20,000 years sooner than the DOE prediction would not change flux rates used in performance assessment calculations, because DOE used the prescribed deep percolation flux rates specified in 10 CFR 63.342 for the post-10,000-year period.

The glacial-transition climate represents 8,000 years of constant climate in the DOE model. Because millennial-scale fluctuations in climate are reflected in the paleo-records, NRC staff evaluated the acceptability of using a constant climate for 8,000 years instead of an alternative representation reflecting millennial-scale variations in climate. The NRC staff concludes that the constant climate that DOE used for the final 80 percent of the 10,000-year period, rather than the use of millennial-average net infiltration rates in performance assessment calculations, has low consequence for performance for the following reasons:

- The applicant included extreme events in developing the millennial-average net infiltration rates. DOE included expected variability in annual precipitation over a 1,000-year period in the net infiltration calculations (SAR Section 2.3.1.3.3) (e.g., including the calculated wettest year within each 1,000-year sequence).

- Using DOE information from its performance assessment sensitivity analyses, NRC staff concludes in SER Section 2.2.1.3.6.3.2 that DOE performance assessment results would not be significantly affected by including reasonable temporal variability of percolation that varies around the long-term-average percolation flux within a climate state. Because net infiltration and percolation are closely linked, DOE performance assessment calculations would not be strongly affected by fluctuations in climate that lead to infiltration flux varying around the value of the long-term-average infiltration flux.

In summary, the NRC staff concludes that the applicant adequately represents the timing and duration of climate states for performance assessment, and need not include uncertainty in the timing and duration, because the applicant demonstrated that changes in the representation for the timing and duration of the climate states have a low consequence for performance assessment calculations.

Uncertainty in Climatic Conditions During the Post-Thermal-Pulse Period

The applicant used the glacial-transition climate state to represent climate during the final 80 percent of the initial 10,000 years of repository performance. Of the three climate states that DOE used within the initial 10,000 years of performance, the NRC staff considers the glacial-transition climate state to have the largest potential for affecting repository performance because this climate state has the longest duration, and seepage into emplacement drifts is least affected by the thermal pulse during this climate state. During the thermal pulse, above-boiling conditions and evaporation reduce the flux of water reaching drifts

(SAR Section 2.3.3.1). In evaluating the DOE approach to representing uncertainty in the glacial-transition climate state, the NRC staff considers (i) methodology that DOE used to estimate future climatic conditions, (ii) parameters that DOE found contribute most heavily to uncertainty in downstream applications, (iii) available estimates for climate change from the last glacial maximum, and (iv) representation of intermediate climate fluctuations using a constant climate for a climate state.

DOE first considered the presence and absence of indicator species within Owens Lake to infer changes in climatic conditions relative to present-day climate, then estimated compatible climatic conditions from present-day locations where the same indicator species currently exist. DOE represented uncertainty regarding climatic conditions using upper and lower bounds for mean annual precipitation and mean annual temperature for each climate state. To account for the uncertainties in translating paleoclimatic indicators into meteorological records, DOE selected several meteorological stations to represent each of the bounding climatic conditions at an elevation of 1,524 m [5,000 ft] at Yucca Mountain (SAR Section 2.3.1.2.3.1.2). The criteria for selecting present-day meteorological stations, outlined in SAR Section 2.3.1.2.3.1.2, include (i) presence of Owens Lake indicator species, (ii) mean annual temperature, (iii) rain-shadow effects, (iv) position of the polar front, and (v) length of observational record. Further, DOE selected meteorological stations such that (i) the climate states have larger mean annual precipitation values for upper bounds than for lower bounds and (ii) the upper-bound mean annual precipitation values are larger than the present-day observations for both the monsoonal and glacial transition climate states.

DOE concluded that mean annual precipitation is one of the two parameters that control uncertainty in MASSIF model estimates of net infiltration for all climate states (SAR Section 2.3.1.3.3.2.2). Further, DOE based the selection of representative meteorological stations on the station record length and on observations that are sensitive to mean annual temperature and precipitation seasonality (i.e., ostracode species) without using criteria based on specific values of mean annual precipitation. Uncertainty in mean annual precipitation is a dominant source of uncertainty in the DOE net infiltration estimates, and the DOE procedure for selecting representative meteorological stations yields relatively large uncertainty in mean annual precipitation.

DOE selected meteorological stations for the glacial-transition climate with average observed annual precipitation between 207 and 241 mm/yr [8.1 and 9.5 in/yr] for the lower bound and between 419 and 455 mm/yr [16.5 and 17.9 in/yr] for the upper bound (SAR Table 2.3.1-6). For comparison, meteorological stations at Yucca Mountain have observed mean annual precipitation between 183 and 213 mm/yr [7.2 and 8.4 in/yr], averaging 199 mm/yr [7.8 in/yr]. Accordingly, DOE's upper- and lower-bound mean annual precipitation estimates for the glacial-transition climate state are approximately 2.2 and 1.2 times the average observed present-day precipitation of 199 mm/yr [7.8 in/yr] at Yucca Mountain.

DOE does not expect that a full glacial climatic state would occur within the next 30,000 years, and the SAR did not estimate climatic conditions for a full glacial climatic state. NRC staff has nonetheless reviewed published estimates for mean annual precipitation during the last glacial maximum in the region surrounding Yucca Mountain (Stothoff and Walter, 2013aa). Several of the published estimates listed in Stothoff and Walter (2013aa) quantitatively considered the effects of mean annual precipitation on a water balance. Such quantitative estimates inferred changes in mean annual precipitation and mean annual temperature by considering elevation changes for plant species that have known environmental preferences, hydrologic balances for paleolakes, extent of glacial advances, and regional groundwater balances. Among these estimates, the largest estimated value for mean annual precipitation at the last glacial maximum suggests that mean annual precipitation was 1.9 times larger than present-day mean annual precipitation at Yucca Mountain.

The NRC staff concludes that the applicant adequately represents uncertainty in the magnitude of mean annual precipitation change during the glacial-transition climate state. The basis for this finding is that the upper bound of mean annual precipitation values that DOE used to represent the upper bound of mean annual precipitation during the glacial-transition climate state is substantially larger than published quantitative estimates for mean annual precipitation during the last glacial maximum in the region surrounding Yucca Mountain. The NRC staff further concludes that the applicant adequately represents mean annual temperature and precipitation seasonality during the glacial-transition climate state because DOE based the values on indicators in the region surrounding Yucca Mountain that are sensitive to these factors.

Uncertainty in Climatic Conditions From Anthropogenic Activities

DOE stated that the predicted modern climate is based on “climate records that implicitly include effects of modern society over the duration of historical record” (SNL, 2008ab, Section 6.2, FEPs 1.4.01.00.0A and 1.4.01.02.0A; DOE, 2009cr, Enclosure 8). Uncertainty in the incorporation of anthropogenic effects on climate predictions used as input for net infiltration estimates is twofold. First, monsoonal and glacial-transition climate analog sites are derived from interpretation of the paleoclimatic record (e.g., Owens Lake ostracode and diatom observations). However, current levels of greenhouse gases (i.e., dominantly CO_2 but including other gases) are elevated beyond any levels indicated in paleoclimate records covering the past 800,000 years. Second, the effect of the global climatic changes on Yucca Mountain climate is uncertain. To address these uncertainties, DOE described consequences to infiltration estimates caused by likely projections of climate change in the desert Southwest considering anthropogenic influences.

In DOE (2009cr, Enclosure 8), DOE considered projected climate changes in the desert Southwest, described by the International Panel on Climate Change (Christensen, et al., 2007aa) to assess potential consequences of anthropogenic climate change on repository performance. Projected regional climate change estimates indicate the desert Southwest is likely to see temperature increases that are higher than average global warming and annual precipitation that is likely to decrease in the next century. DOE (2009cr, Enclosure 8) described the projected regional climate changes as having potential consequences, including

- Improved repository performance under warmer and drier conditions, because warmer temperatures and decreased precipitation lead to decreased net infiltration

- Insignificant effects on repository performance under a warmer and wetter climate or early onset of monsoon conditions induced by anthropogenic climate change, because most of the repository would be above boiling during the initial 600 years [the NRC staff notes that the climate change projected by Christensen, et al. (2007aa) is similar but smaller than that used by DOE for a switch from present-day to monsoonal climate]

- Improved repository performance if anthropogenic climate change caused a delay in the onset of the glacial-transition climate because net infiltration under the cool and wet glacial-transition climate state is higher than would occur for earlier climate states

The NRC staff notes that recent advances in understanding past and predicting future global climate patterns identify two types of El Niño events (e.g., Kidwell, et al., 2014aa) and indicate that extreme El Niño events may occur with increased frequency relative to historical observations (Cai, et al., 2014aa). El Niño events are a climatic change that leads to wet winters in the desert Southwest and therefore may lead to infiltration events that progress to deep percolation. Because infiltration pulses at Yucca Mountain may occur in years identified as El Niño years, NRC staff compared projections of future climate considered by DOE (2009cr, Enclosure 8) with more recent estimates of potential changes in regional climate. Focusing on regional impacts of global warming, Cayan, et al. (2013aa) project little change in annual precipitation and less than 3 percent increase in winter precipitation for southern Nevada under the worse of two scenarios of greenhouse gas emissions. NRC staff concludes that the range of projected climatic conditions indicated by this more recent information remains in the range considered in DOE (2009cr, Enclosure 8).

The NRC staff concludes that DOE adequately bounded the effects of anthropogenic climate change in its performance assessment calculations because DOE demonstrated that (i) net infiltration is not consequential to repository performance during the initial 600-year period and (ii) net infiltration under credible projected climate changes would be overestimated by using the climate states already used for performance assessment calculations.

Summary of Conclusions Regarding Climate Change for the Initial 10,000 Years

In summary, the NRC staff concludes that the applicant adequately represents future climate uncertainty because (i) DOE provided a transparent and traceable description of the approach and data used to represent climate states, and furthermore, that those climate states and their uncertainties are adequately representative of Yucca Mountain during the initial 10,000 years; (ii) the DOE representation of the timing and duration of the climate states has a low consequence for performance assessment calculations; (iii) DOE used upper-bound values for mean annual precipitation during the glacial-transition climate state (post-thermal-pulse) that are more extreme than the largest available published estimate for the last glacial maximum; and (iv) DOE projected changes to climate stemming from anthropogenic activities that are either inconsequential to repository performance or are bounded by the climate states used for performance assessment calculations.

2.2.1.3.5.3.2.2 Local Spatial and Temporal Variation of Meteorological Conditions

In this section, NRC staff evaluates the DOE model for climatic and meteorological conditions during each climate state (i.e., climate conditions at short time scales). This section addresses YMRP acceptance criteria related to input data characterization and uncertainty.

DOE represented meteorological conditions for each climate state using sampled 1,000-year sequences of daily estimates for total precipitation and temperature extremes, representing conditions at a reference elevation of 1,524 m [5,000 ft]. The MASSIF model subsequently estimates precipitation and temperature variability over each day using the daily values. DOE represented spatial variation by projecting the daily precipitation and temperature values to the infiltration-model cells using elevation-dependent lapse rates. DOE considered precipitation rates with up to a 1,000-year recurrence period in generating the 1,000-year sequence. The applicant selected 10 representative 1-year sequences out of each 1,000-year sequence to estimate long-term-average net infiltration. DOE used a water year representation, initiating each 1-year sequence on October 1 to capture the cycle of winter precipitation and large summer potential evapotranspiration. Each simulation was initiated with conditions representing extended summer evapotranspiration. DOE stated that the wettest years were sampled to ensure the disproportionate influence of wet years was captured for net infiltration estimates.

NRC staff reviewed the adequacy of the applicant's representation of spatial and temporal variability in meteorological parameters for estimating net infiltration in the two following sections.

Spatial Variability in Meteorological Parameters

DOE considered the effect of elevation on meteorological parameters by adjusting the estimated daily values for precipitation and temperature extremes according to regional patterns in mean annual precipitation and mean annual temperature. The NRC staff notes that the dependency of meteorological parameters on elevation provides the only spatial variability of those parameters across the site in the DOE model. NRC staff concludes, however, that the applicant's approach of representing spatial variability in precipitation and temperature is acceptable for the model's purpose to provide long-term-average net infiltration estimates as a boundary condition to the site-scale unsaturated zone flow model, for the following reasons:

- DOE assumed all precipitation events were larger than the net infiltration model domain. NRC staff notes that the size of the east- and northeast-trending washes is small relative to the size of typical precipitation events. Precipitation patterns during individual precipitation events are likely to be relatively spatially uniform within a few kilometers [a few miles], especially during large frontal storms that are predominant in the cooler periods of the year when potential evapotranspiration is small. For example, maximum observed 24-hour precipitation ranged between 78.5 and 87.1 mm [3.1 and 3.4 in] (SAR Table 1.1-23) for the 10 meteorological stations within 5 km [3 mi] of the repository footprint in the largest 24-hour event ever recorded with onsite meteorological stations (September 21 to 22, 2007). The repository footprint has an area of 4.6 km^2 [1.8 mi^2] and includes more than a dozen washes. The NRC staff concludes on the basis of a comparison of the smaller size of individual washes relative to the larger size of frontal storms that precipitation patterns are likely to be relatively spatially uniform within individual washes above the proposed repository footprint, and therefore, DOE need not consider spatial variation of precipitation for mean annual precipitation.

- To estimate long-term-average net infiltration, DOE assumes the washes are hydrologically independent (i.e., there is no lateral flow between washes). On the basis of site topography and drainage system (SAR Figure 1.1-5; NRC, 2010aa, Section 1.5.3.2.2.2), the NRC staff concludes that the washes within the repository footprint can reasonably be considered hydrologically separate.

- To estimate long-term-average net infiltration, DOE assumed a given meteorological sequence is equally applicable to each wash in the model domain. The NRC staff concludes this is an acceptable assumption because each small wash is likely to exhibit similar frequencies of meteorological patterns over long periods of time due to close proximity.

- DOE cited regional studies in SNL (2007az, Sections 6.4.11 and 6.4.5.3) indicating that mean annual precipitation and mean annual temperature are correlated with elevation even though local topography can modify the relationship. DOE derived the precipitation lapse rate used for infiltration calculations from meteorological stations in the Yucca Mountain region, as outlined in SNL (2007az, Appendix F.2). The NRC staff confirmed that the DOE precipitation lapse rate is comparable to other regional relationships, within the bounds of uncertainty, over an elevation difference typical of the repository footprint (Stothoff, 2008aa). Because the applicant demonstrated that the net infiltration model results, other model results for areas in the desert Southwest and western United States, and observational evidence from other locations all exhibit a systematic trend of larger net infiltration as mean annual precipitation increases (SAR Section 2.3.1.3.4), the NRC staff concludes that the applicant adequately considers the systematic elevation-dependent variation in mean annual precipitation on net infiltration for calculating long-term average net infiltration.

- DOE derived the temperature lapse rate used for infiltration calculations from a textbook value for the dry adiabatic lapse rate, representing an upper bound representation, as detailed in SNL (2007az, Appendix C.1.4). Although this temperature lapse rate may overestimate the regional lapse rate, DOE demonstrated in SNL (2007az, Section 7.1.4) that the net infiltration model results are not sensitive to the temperature lapse rate. Therefore the NRC staff concludes that the temperature lapse rate is acceptable for calculating long-term-average net infiltration.

In summary, the NRC staff concludes that the DOE approach of spreading a single, generated meteorological time history throughout the model domain using elevation-dependent lapse rates is acceptable for calculating boundary condition fluxes for the site-scale unsaturated zone flow model. The NRC staff finds that this approach is unlikely to systematically bias the calculated areal-average long-term-average net infiltration. This conclusion is further based on the observations that (i) the individual washes within the repository footprint have relatively small areas compared to typical storms, (ii) the individual washes within the repository footprint are hydrologically independent with respect to lateral flow, and (iii) systematic trends in the meteorological parameters that most strongly affect net infiltration are adequately incorporated in the infiltration calculations.

Temporal Variability in Meteorological Parameters

DOE represented temporal variability of meteorological conditions using sampled daily values for precipitation, temperature minimum and maximum, and wind speed. On days with precipitation, the applicant subdivided the daily calculation into two parts, representing storm and nonstorm conditions, and used wet-day instead of dry-day temperature values. The applicant described the statistical parameters characterizing these meteorological components as varying sinusoidally over the year. DOE considered the representation of temporal variability adequate because measured regional and Yucca Mountain site data were used to develop precipitation and temperature sequences.

The NRC staff concludes that the applicant’s representation of cool-season (winter) precipitation is a risk-significant aspect of temporal variability of meteorological conditions, because warm-season precipitation has a disproportionately small effect on net infiltration. NRC staff’s conclusion is based on its literature surveys and independent confirmatory investigations in Stothoff and Walter (2013aa) and Stothoff (2013aa,ab). In particular, the NRC staff notes measurements and analyses indicate that approximately 10 percent of recharge at Mount Charleston (in the Spring Mountains, southeast Nevada) has the isotopic signature of summer precipitation, which represents approximately 30 percent of the annual precipitation (Winograd, et al., 1998aa).

Accordingly, the NRC staff compared the applicant's mathematical representation of precipitation in SNL (2007az, Appendix F) with summary observations from meteorological stations the applicant used to represent mean winter and summer precipitation for potential future climate states. In the analysis (Stothoff, 2010aa), the NRC staff concluded that several statistical properties from the observed precipitation data sets fell close to statistical properties of DOE's precipitation representation. The NRC staff recognizes that there is uncertainty in estimating mean annual precipitation from observations; for example, average precipitation totals from 1994–2006 for five Yucca Mountain Project meteorological stations, reported in SAR Tables 1.1-10 through 1.1-12, 1.1-15, and 1.1-18, differ on average by approximately 7 percent from values for 1993–2004 reported in SNL (2006aa, Table 6.1-4). The NRC staff concludes that DOE's representation of precipitation is acceptable for calculating daily precipitation for net infiltration because the statistical model has seasonal patterns for precipitation values comparable to observations.

The NRC staff compared DOE's representation for temperature as sinusoidally varying during the year to observations from meteorological stations in Nevada, Utah, California, and Arizona, as described in Stothoff (2008aa, Figures 5-8 and 5-09). The amplitude and seasonality DOE represented is comparable to these observations. The NRC staff considers net infiltration calculations to be relatively sensitive to temperature on days with precipitation because evaporation during precipitation affects the amount of water infiltrating during an infiltration event. Calculations of net infiltration, however, are relatively insensitive to temperature fluctuations on days without precipitation because it typically takes weeks for evapotranspiration rates to remove the soil moisture from a large storm. The NRC staff concludes that DOE adequately considered temperature in daily meteorological sequences because the applicant used a representation for temperature that has amplitude and seasonality comparable to observations, and because the applicant used separate parameterizations for wet and dry days.

2.2.1.3.5.3.3 Net Infiltration

In this section, NRC staff evaluates DOE's model for net infiltration during the 10,000 years following repository closure. The NRC staff evaluates the downstream uses of the net infiltration results and the effect of uncertainty in net infiltration on DOE performance assessment calculations. The focus of the NRC staff's review is on those aspects of DOE's net infiltration model that are most important for repository performance.

In evaluating repository performance with respect to unsaturated flow (see SER Sections 2.2.1.3.6.3.2 and 2.2.1.3.6.3.4), the NRC staff identified systematic changes in seepage as the dominant performance-affecting consequence of spatial and temporal variability in percolation fluxes within the unsaturated zone below the Paintbrush Tuff nonwelded (PTn) Formation. Also in those SER sections, the NRC staff concluded that the DOE model is relatively insensitive to other aspects such as spatial variability in deep percolation, local flow focusing, decadal-to-centennial climatic variability, episodic deep percolation pulses, and calibration of net infiltration uncertainty. Because seepage and percolation closely track net infiltration, the NRC staff considers areal-average net infiltration to be the dominant performance-affecting feature of the net infiltration model with respect to DOE's performance assessment calculations.

DOE also used the areal-average net infiltration results in performance assessment calculations related to the saturated zone, adjusting saturated zone groundwater fluxes for future, wetter climates using net infiltration (SAR Section 2.3.8). Net infiltration contributes a flux of water to deep percolation (and hence seepage) and to recharge to the saturated groundwater flow system. The NRC staff notes, however, that repository performance is more sensitive to changes in seepage than changes in saturated zone groundwater fluxes. Seepage directly affects radionuclide releases, so uncertainty in seepage directly affects the uncertainty in calculated dose. Uncertainty in groundwater flux rates has a much smaller effect on uncertainty in dose calculated by the DOE performance assessment because (i) nonsorbing radionuclides (e.g., Tc-99) are transported through the saturated zone in a small fraction of the performance period regardless of uncertainty in groundwater flux rates and (ii) uncertainties in transport rates for sorbing radionuclides are dominated by uncertainties in aspects other than groundwater flux, such as sorption characteristics. The following subsections contain the NRC staff's evaluation of the technical bases for DOE's models and inputs used to estimate areal-average net infiltration.

Submodels for Net Infiltration

This subsection addresses YMRP acceptance criteria related to model integration, justification, and uncertainty.

DOE used a water balance approach to integrate processes and features acting at and near the ground surface to a depth at the bottom of the root zone. A water balance approach requires that the supply of water (precipitation and run on) at any location is equal to the sum of other components (e.g., evapotranspiration, change in water storage, runoff, and net infiltration). In a water balance, uncertainty in precipitation and evaporation, the largest components, can dramatically affect estimation of the much smaller component of net infiltration. DOE described the development and integration of features and processes into conceptual and numerical models in SAR Sections 2.3.1.3.1 and 2.3.1.3.2 and SNL (2007az, Sections 6.3 and 6.4). DOE separates the water balance into the key MASSIF model elements of

- Climate and meteorology, using daily precipitation, temperature, and snowmelt
- Subsurface water movement and storage, using a one-dimensional vertical soil water balance
- Surface runoff and run on, using topography-based flow routing
- Evapotranspiration, using the FAO-96 approach (Allen, et al., 1998aa) modified for natural vegetation
- Reference evapotranspiration, using the FAO Penman-Monteith method (Allen, et al., 2005aa)

The MASSIF model is comprised of linked submodels for the identified processes and routines to manipulate geographically distributed input data into the formats required for the calculations. The geographically distributed input data, which are defined at each 30-m [98-ft] pixel across the Yucca Mountain area, include soil and rock hydrologic properties, topography, vegetation factors, and climate information (e.g., precipitation and temperature).

In the MASSIF model, net infiltration is defined as the water that moves below the active zone where evaporation and plant uptake are significant processes. DOE assumed that the active zone does not penetrate the bedrock, so that water passing into the bedrock becomes net infiltration. DOE considered this assumption to be conservative with respect to the magnitude of net infiltration, because plant roots, especially in areas with thin soil cover, develop in bedrock fractures and therefore would reduce net infiltration by taking up water from fractures for transpiration.

DOE described in SNL (2007az, Section 6.3.2) six criteria used for selecting components (elements, features, or submodels) to include in the MASSIF model: (i) the components should be consistent with the overall project purpose, (ii) the component complexity should be consistent with the available input data, (iii) the components should be consistent with other components in the MASSIF model, (iv) the model should be computationally efficient, (v) the model should be accessible and open, and (vi) the model and its components should demonstrate reasonable predictive capability. The applicant described cases in the literature where the algorithms and approaches in submodels have been utilized at other locations in semi-arid areas. DOE explained that these criteria were motivated by the large spatial and temporal scales being modeled, the limited objectives of the net infiltration model, and the need for numerous simulations to assess sensitivities and address multiple climate scenarios. DOE further explained that the net infiltration model is not intended to describe the detailed spatial and temporal character of water movement.

The NRC staff has experience in evaluating the features, processes, and models used for arid zone hydrology gained from two decades of prelicensing interactions with DOE and from documenting the NRC staff’s independent modeling (Stothoff, 2013aa,ab, 2008aa; Stothoff and Walter, 2013aa). On the basis of knowledge gained through this experience, the NRC staff concludes that the applicant adequately (i) identified and included features and processes in its net infiltration models that are important for estimating future net infiltration at Yucca Mountain, and (ii) described the technical basis for the infiltration conceptual model and the associated mathematical model in SAR Section 2.3.1.3 and supporting documents. Furthermore, NRC staff concludes that the MASSIF submodels are acceptable for their intended use because the algorithms and approaches (i) are widely used in the scientific community, (ii) are appropriate for the spatial and temporal scales described in DOE’s six criteria for the net infiltration model (see previous paragraph for DOE’s criteria), and (iii) consider downstream uses. The DOE submodels are consistent with staff’s experience regarding approaches for modeling net infiltration in arid environments (i.e., Stothoff and Musgrove, 2006aa; Stothoff, 2013aa,ab).

To build confidence in the water balance approach for one-dimensional water storage and movement, DOE compared results with both measured data and results from an alternative model solving the Richards equation for unsaturated flow (SAR Sections 2.3.1.3.4.1 and 2.3.1.3.4.2). The Richards equation includes capillary effects, unlike the water balance approach in the MASSIF model. The DOE comparisons focused on the MASSIF submodels for water storage, evapotranspiration, and one-dimensional vertical movement of water. The measured lysimeter data were from two locations: the Reynolds Creek Experimental Watershed in New Mexico and the Nevada Test Site. DOE also compared MASSIF model results with results from the Richards-equation-based models for one-dimensional, stylized problems with varying soil and plant root depths. The NRC staff reviewed the DOE comparisons of MASSIF and Richards-equation-based model results with measured data and the stylized one-dimensional problem. On the basis of this review, the NRC staff concludes that the water storage and evapotranspiration submodels in MASSIF adequately represent both the measured data and the Richards-equation-based model results. The NRC staff finds that the MASSIF and the Richards-equation-based models have similar responses, and that the two numerical models track the observations to a quantitatively similar degree of accuracy.

In another comparison with measured data, DOE contrasted MASSIF model results against measurements from streamflow gauges in three subwatersheds at Yucca Mountain for several storm events, as outlined in SAR Section 2.3.1.3.4.1 and SNL (2007az, Section 7.1.3). SAR Figure 2.3.1-46 illustrated that the timing and magnitude of measured and modeled runoff are reasonably well matched with a particular set of input properties that lie within the uncertainty range considered in the net infiltration model. The input properties used to match streamflow observations represent the nominal properties with soil hydraulic conductivity adjusted to increase upland runoff and enhance channel infiltration. DOE indicated that local variations within the watersheds may have also been a factor in the comparison. On the basis of the adjustments DOE made to match streamflow observations, the uncertain input parameter distributions used for performance assessment calculations may, based on NRC staff's knowledge gained through experience and independent analyses, create a bias toward over-estimating the fraction of MASSIF model-calculated total net infiltration that upland infiltration contributes relative to channel infiltration.

The NRC staff recognizes, however, that uncertainty in runoff predominantly affects the spatial distribution of infiltration in the DOE model and has minimal effect on the areal-average infiltration; it is this latter factor that is of greater importance to repository system performance. The NRC staff therefore concludes that the surface runoff submodel provides a reasonable basis for predicting runoff over the entire repository for the initial 10,000 years because (i) the surface runoff submodel algorithms are commonly used; (ii) DOE demonstrated that, with appropriate input parameters, the MASSIF surface runoff submodel is capable of providing a reasonable match to observed runoff during storm events within representative subwatersheds; (iii) the DOE results are reasonably comparable to an independent, alternative model for runoff and infiltration (Woolhiser, et al., 2000aa, 2006aa); (iv) the uncertainty in net infiltration stemming from uncertain runoff is small; and (v) the DOE performance assessment calculations are not sensitive to different representations of spatial variability in net infiltration, as described in DOE (2009cr, Enclosure 4). A more general evaluation of uncertainty in spatial variability is in a subsequent section of SER 2.2.1.3.5.3.3 entitled "Net Infiltration Results."

Input Parameters

This subsection addresses YMRP acceptance criteria related to characterization of data and propagation of data uncertainty.

NRC staff understands that most net infiltration within the proposed repository footprint occurs in shallow soil as a pulse over a few days to weeks following large precipitation events during periods of low potential evapotranspiration. This understanding is based in part on the DOE documentation of its net infiltration modeling. It is also based on the NRC staff's experience with infiltration processes and modeling at Yucca Mountain (Stothoff, 2013aa,ab, 2008aa, and references therein). During the short intervals with large net infiltration pulses, which dominate the long-term-average net infiltration, the flux of water passing into the bedrock in shallow soil can dominate evapotranspiration in the water balance. Accordingly, the NRC staff focused its review in this section on aspects of the site affecting rapid transmission of pulses to bedrock in shallow soil; in particular, factors related to soil water storage.

Available soil water storage during a precipitation event depends on soil depth, soil water-holding capacity, and the antecedent soil moisture content (i.e., how dry the soil column is prior to the event). Because the MASSIF model components for precipitation and evapotranspiration affect levels of soil moisture, the focus of the review here is on properties of the porous media. Soil and bedrock hydraulic properties affect the rate of soil water movement toward and into the bedrock. In DOE's representation, water drains into the bedrock once the water storage capacity of the overlying soil layers is exceeded, thereby avoiding evapotranspiration. The drainage rate into the bedrock is controlled by the layer—soil or bedrock—with the smaller value bulk permeability. Using analyses reported in SNL (2007az, Section 7.1.4), DOE stated that net infiltration estimates are not sensitive to uncertainty in bulk bedrock permeability, in part because of the limited spatial extent where bedrock controls the drainage rate. The NRC staff's assessment of the distribution of hydrological properties is consistent with the DOE assertion of the limited extent where bedrock controls the drainage rate (Stothoff, 2008aa). Therefore, NRC staff's review focuses on the soil porous media properties of soil depth, soil water-holding capacity, and changes to these porous media properties in future climates. Soil water-holding capacity depends on the soil porosity and unsaturated hydraulic properties.

Areas with thin soil cover are particularly important because DOE identified infiltration as most readily occurring in shallow soil. Areas with shallow, or thin, soils comprise 70 percent of the unsaturated model domain (Soil Depth Class 4, SAR Section 2.3.1.3.2.1.3) and appear to cover an even larger fraction of the repository footprint (SAR Figure 2.3.1-19). In DOE (2009cr, Enclosure 5), DOE identified soil depth in areas with shallow soil cover as the most important hydrologic property input for the net infiltration model, with model results approximately as sensitive to uncertainty effective soil depth as uncertainty in precipitation. The effective soil depth is defined as the single soil depth value that, if applied everywhere, yields the same areal-average net infiltration as the actual soil-depth distribution.

Soil Depth Class 4 represents soil depths between 0 and 0.5 m [0 and 1.6 ft] and corresponds to eolian deposits with various mixtures of entrained rock on hillslopes and ridgetops. DOE sampled a single effective soil depth value to characterize Soil Depth Class 4 for each realization and assigned the value to every grid cell representing that class for the corresponding simulation. DOE used two datasets to support its effective soil depth distribution for this class: (i) 35 site observations recorded as point measurements ranging from depths of 0 to 3 m [0 to 9.8 ft], with a recommended median of 0.25 m [0.82 ft], as described in SNL [2007az, Table 6.5.2.4-2(a)], and (ii) 8 site observations recorded as general site characteristics at locations such as drill pads. DOE described the measurements as approximately lognormally distributed and, on the basis of geometric and arithmetic means of the two sets of observations, derived bounds on effective soil depth ranging from 0.1 to 0.5 m [0.33 to 1.6 ft]. DOE also analyzed 56 NRC soil depth measurements (Fedors, 2007aa) obtained from site visits focusing on the thin soils of the east-trending upper washes over the southern half of the repository footprint. DOE described the available NRC staff's measurements as approximately following a lognormal distribution.

For performance assessment calculations, DOE described the effective soil depth in Soil Class 4 as equally likely for any value between the upper and lower bounds. In selecting the uniform statistical distribution, DOE considered the difficulty in measuring soil depth, uncertainty in the mean of the observations, and uncertainty in how soil depth affects net infiltration, as detailed in DOE (2009cr, Enclosure 5). In the same document, DOE stated that sensitivity results in SNL (2007az, Appendix H) indicate that calculated areal-average net infiltration is approximately linearly dependent on effective soil depth for Soil Class 4, and that shallow soil depths are not underrepresented in the effective soil depth distribution for Soil Depth Class 4. From this, DOE concluded that use of a uniform distribution for effective soil depth does not underestimate average net infiltration.

NRC staff concludes that DOE adequately characterized the statistical properties of the soil depth observations for use in the net infiltration modeling on the basis of the following two reasons. First, the NRC staff concludes that DOE provided acceptable bounds on the uncertainty in effective soil depth in Soil Depth Class 4. A shallower extreme would require that more than half of the area with shallow soil has a soil depth less than 0.1 m [0.33 ft], which, in contrast to observations, implies extensive exposures of bare rock in the net infiltration model domain; a deeper extreme would reduce net infiltration. Second, the NRC staff's conclusions are supported by its own confirmatory field investigations (Stothoff, 2008ab), which indicate that, consistent with the DOE description of an eolian source for the fine component of the soil, comparable topographic locations on hillslopes and ridgetops have similar soil depths across the repository footprint.

NRC staff considered the effect of uncertainty in the soil depth distribution for Soil Depth Class 4 on the estimate of net infiltration. As described in SNL [2007az, Table 6.5.2.4-2(a) and Section 7.2.4(a)] and DOE (2009cr, Enclosure 5), DOE used a uniform distribution, while noting that the measured data may better fit a lognormal distribution. To assess the DOE representation of soil depth uncertainty, the NRC staff used the applicant's sensitivity analyses for fixed aleatory uncertainty under present-day and glacial transition climate states, as outlined in SNL (2007az, Figures H–3, H–4, H–11, and H–12), to estimate the consequence of decreasing median soil depth. Using this alternative representation for how uncertainty might be distributed, the NRC staff's analysis suggested areal-average net infiltration would increase by 43 to 61 percent under the present-day climate state and by 29 to 38 percent under the glacial transition climate state if mean soil depth decreased from 0.3 m to 0.2 m [0.98 ft to 0.65 ft] for Soil Depth Class 4 (Stothoff, 2010aa). DOE (2009bo, Enclosure 5) demonstrated that increasing average net infiltration over the ambient unsaturated zone flow model domain (reviewed in Section 2.2.1.3.6.3) by percentages greater than 81 percent, due to uncertainty in the probability weights of the Generalized Likelihood Uncertainty Estimation methodology (reviewed in Section 2.2.1.3.6.3), did not strongly affect DOE's performance assessment calculations. Because the increase (29 to 61 percent) in net infiltration calculated by the NRC staff using an alternative representation of the uncertainty of soil depth distribution is significantly smaller than the increase DOE estimated for uncertainty in net infiltration due to the Generalized Likelihood Uncertainty Estimation weights, the NRC staff concludes that uncertainty in soil depth distribution of Soil Depth Class 4 is also not important to performance. The NRC staff concludes that DOE adequately represents effective soil depth in Soil Depth Class 4. This is because the bounds on the uncertainty distribution for net infiltration are sufficiently wide to cover the effects of uncertainty in soil depth.

Water-holding capacity is calculated from soil hydraulic properties (i.e., porosity and water retention characteristics). DOE utilized a pedotransfer function derived for Hanford soils to relate Yucca Mountain soil texture to hydraulic properties for each soil group in the infiltration model domain. In the MASSIF model, larger water-holding capacity values result in smaller values of net infiltration. Because the hydrologic property relationship to soil texture may be different for soils from Hanford, Washington, compared to that for soils at Yucca Mountain, DOE (2009cr, Enclosure 6) compared water-holding capacity used in the MASSIF model with two estimates made for local Yucca Mountain soils. The first set covers soils in Nye County, and the second set covers soils from the Yucca Mountain area but not used in the MASSIF model. DOE (2009cr, Enclosure 6) stated that the estimates of water-holding capacity used in the MASSIF model are smaller than those for the other two sets of soils. On the basis of its review of this information, NRC staff concludes that the development of average water-holding capacity values for soils from the Hanford-based pedotransfer function does not lead to underestimation of water-holding capacity for Yucca Mountain soils.

NRC staff considered the performance consequence of the DOE assumption that all geologic and geographic parameters in the net infiltration model remain the same over the transition from dryer to wetter climates during the initial 10,000 years. DOE (2009cr, Enclosure 2) described changes to soil depth, soil hydraulic properties, and bulk bedrock permeability under future climates that may include (i) greater chemical soil profile development and enhanced weathering of bedrock at the interface with soil, (ii) relatively larger soil depths and different soil depth distributions, and (iii) relatively smaller amounts of caliche filling bedrock fractures at the soil/bedrock interface. In DOE (2009cr, Enclosure 2), DOE described the potential consequences as either inconsequential or beneficial to repository performance:

- Projected changes to soil depth and soil hydraulic properties would tend to reduce the estimates of net infiltration. The NRC staff determines that the DOE (2009cr, Enclosure 2) description of projected changes is consistent with current understanding of geomorphic responses to climate change in the desert Southwest [e.g., Bull (1991aa, Section 2.5)].

- Where bedrock permeability values are greater than soil permeability, DOE considered the effect of a change in modeled bedrock properties as either inconsequential to net infiltration (if bedrock permeability increased) or reducing infiltration (if bedrock permeability became smaller than the soil permeability). For more than half the ambient site-scale unsaturated zone modeling domain, DOE (2009cr, Enclosure 2) stated that bedrock permeability is greater than soil permeability.

- In the remaining area, DOE considered net infiltration as having a potential to increase only in the realizations where sampled bedrock permeability is smaller than soil permeability in the present climate and only if bedrock permeability increases under a future climate. The NRC staff notes that this potential exists in less than half of the modeling domain for approximately half the realizations. The NRC staff calculated an upper-bound estimate for areal-average net infiltration (Stothoff, 2010aa) that is 1.5 times larger than the DOE-calculated value, conservatively assuming that (i) half the area in half the realizations has low bedrock permeability and zero infiltration, (ii) all DOE-calculated net infiltration occurs in the remaining area with high bedrock permeability, (iii) increased bedrock permeability results in an upper bound of the areal-average net infiltration from the high bedrock permeability, (iv) the entire area has high bedrock permeability in all realizations, and (v) projected changes to soil depth and soil hydraulic properties do not result in lower net infiltration.

NRC staff concludes that DOE’s approach to maintain constant, but uncertain, soil depth and hydraulic properties and bulk bedrock permeability over the initial 10,000 years of performance is not likely to lead to consequential increases in areal-average net infiltration, because most expected changes to these properties would tend to reduce net infiltration in the applicant’s model. For changes to properties that lead to increased estimates of areal-average net infiltration (third bullet in the paragraph above), the NRC staff concludes that the upper bound increase in areal-average net infiltration is less than the increase that DOE (2009bo, Enclosure 5) demonstrated is not significant to calculated repository performance.

Net Infiltration Results

This subsection addresses YMRP acceptance criteria related to propagation of uncertainty and support of model output for DOE’s estimates of net infiltration for each climate state. Effects of net infiltration model results on repository performance are considered in how the net infiltration model output is used in the unsaturated flow model, including seepage and unsaturated zone transport. NRC staff reviewed the net infiltration results in the context of average ratio of infiltration to precipitation, values of areal-average net infiltration, and spatial and temporal distribution of net infiltration.

DOE represents uncertain inputs to the MASSIF model with Monte Carlo sampling, using 40 realizations of selected hydraulic properties and climate characteristics to estimate net infiltration uncertainty for a climate state (SAR Section 2.3.1.3.3). For each realization, DOE calculated a process-level map of mean annual net infiltration by (i) creating a synthetic weather history representing 1,000 years, (ii) selecting 10 water years (a water year is October 1 through September 30 of the following year) out of the 1,000-year history to represent the range of dry to wet years, (iii) calculating total net infiltration for each water year using the MASSIF model, and (iv) averaging the 10 water-year net infiltration maps. The applicant selected 4 of the 40 equally likely process-level mean annual net infiltration maps to represent the uncertainty in infiltration by (i) calculating areal-average net infiltration for each map, (ii) ranking the average values from low to high, and (iii) selecting the 4th, 12th, 20th, and 36th ranked map to represent the 10th, 30th, 50th, and 90th percentile ranking. The 12 maps of net infiltration are output provided to the unsaturated zone model for use as top boundary flux for the initial 10,000 years. Because DOE used a standard approach for propagation of data uncertainty and conservatively assumed each stochastic realization was equally likely to occur, NRC staff concludes that DOE acceptably propagated uncertainty in climate and hydrologic parameter inputs in development of the net infiltration maps.

The applicant adjusted 4 of the 12 upper-boundary net infiltration maps developed for the initial 10,000 years to represent the probability distribution for deep percolation at the repository horizon for the post-10,000-year period, as specified by 10 CFR 63.342(c)(2). DOE selected the four upper-boundary net infiltration maps with the largest areal-average net infiltration within the repository footprint for the scaling procedure. The NRC staff reviews the procedure and technical basis for developing the post-10,000-year unsaturated zone model upper-boundary net infiltration maps in SER Section 2.2.1.3.6.3.2.

In NRC staff's analysis, three primary aspects of the response of net infiltration to climate may affect the performance of the proposed repository: (i) central tendency and uncertainty in areal-average time-averaged net infiltration, (ii) spatial variability in time-averaged net infiltration, and (iii) temporal changes in net infiltration. The NRC staff identified systematic changes in seepage as the dominant performance-affecting consequence of spatial and temporal variability in percolation fluxes below the PTn (SER Section 2.2.1.3.6.3.2). Through evaluation of DOE sensitivity analyses, the NRC staff concludes in SER Section 2.2.1.3.6.3.2 that DOE's performance assessment results are not strongly affected by any of the following factors: (i) systematic changes in seepage arising from different representations of local flow focusing, (ii) long-term climatic variability, (iii) episodic infiltration pulses, or (iv) calibration of infiltration uncertainty. Because seepage and deep percolation track net infiltration, the NRC conclusion for factors evaluated in SER Section 2.2.1.3.6.3.2 for deep percolation and seepage can be translated to the (similar) factors for net infiltration identified at the beginning of this paragraph. Therefore, the NRC staff concludes that spatial or temporal variability in net infiltration has little effect on performance assessment.

NRC staff compared the average reduction in precipitation that becomes net infiltration calculated for Yucca Mountain (SAR Tables 2.3.1-2 through 2.3.1-4) with reductions for other sites reported in the general literature. For Yucca Mountain, the ratio of calculated areal-average net infiltration to mean areal-average precipitation ranges from 0.022 to 0.154 for the present-day climate state, 0.023 to 0.191 for the monsoon climate state, and 0.047 to 0.166 for the glacial transition climate state. These areal-average estimates represent the entire MASSIF model domain. The NRC staff concludes that these ratios are consistent with infiltration estimates from other arid and semi-arid sites in Nevada and the western United States with comparable precipitation rates (e.g., Stothoff and Musgrove, 2006aa; SNL, 2007az, Section 7.2.1.2). Based on the comparison of DOE estimates for Yucca Mountain with other sites in Western United States, NRC staff concludes that the DOE estimates of areal-average net infiltration and its uncertainty are adequately supported by information from relevant analog sites.

Using NRC staff's independently developed net infiltration model and independent staff field and modeling confirmatory investigations related to infiltration at Yucca Mountain (e.g., Stothoff, 2013aa,ab, 2009aa, 2008ab, 1999aa, 1997aa, 1995aa; Stothoff, et al., 1999aa; Groeneveld, et al., 1999aa; Woolhiser, et al., 2000aa, 2006aa; Stothoff and Musgrove, 2006aa; Stothoff and Walter, 2013aa), the NRC staff further assessed DOE's calculated uncertainty regarding areal-average net infiltration estimates for the climate states used in the DOE performance assessment. The NRC staff's Infiltration Tabulator for Yucca Mountain (Stothoff, 2008aa) is an independent numerical model to estimate the uncertainty in net infiltration at Yucca Mountain. The Infiltration Tabulator for Yucca Mountain uses the same starting site characteristics of soil and bedrock maps and bedrock matrix hydraulic properties as the MASSIF model. The Infiltration Tabulator for Yucca Mountain differs from MASSIF model in its computational approaches; conceptual models for water redistribution in subsurface overland flow and evapotranspiration; and input parameters for soil depths, soil and bedrock hydraulic properties, and topography.

The NRC staff compared outputs from the DOE (SNL, 2007az) and NRC models (Stothoff and Walters, 2013aa) and concludes that the models produce comparable near-lognormal distributions for calculated areal-average net infiltration when using similar uncertainty in input parameters. Similarly, both models show that soil thickness and the soil retention characteristics are the two dominant factors (after precipitation) controlling net infiltration. The NRC staff concludes that DOE's model and the independent NRC model have generally comparable representations for uncertainty in areal-average net infiltration.

On the basis of these comparisons, the NRC staff concludes that (i) each set of four net infiltration maps used to represent a climate state in the initial 10,000 years falls within or above the range of estimates represented by other sites with comparable arid climatic conditions and (ii) DOE's uncertainty distribution adequately represents uncertainty in areal-average net infiltration.

In support of its consideration of spatial variability in DOE's net infiltration in the vicinity of the proposed repository footprint, DOE provided an analysis in DOE (2009cr, Enclosure 4) that considers the consequences on performance from a variant property set that favors focused (channel) infiltration instead of distributed infiltration. The variant property set was based on simulations of observations in Pagany Wash. DOE considered the consequences of both spatial and temporal aspects of the variant property set with the net infiltration model from two washes, stating that the consequences of a focused infiltration pattern are insignificant to repository performance calculations. The base case and variant simulations yield similar calculated values of areal-average net infiltration, with somewhat less net infiltration within the repository footprint for maps with larger total net infiltration, as described in SNL [2007az, Table 7.1.3.2-1(a)]. DOE explained that the similarity in areal-average net infiltration arises from conservation of mass, in that the water infiltrating into channels in the variant case would have otherwise infiltrated into hillslopes without the enhanced runoff. In other words, redistributing water through overland flow does not appreciably increase areal-average net infiltration in DOE's model. Besides runoff, interflow may also lead to lateral redistribution and change the spatial pattern of infiltration. However, NRC staff maintains that interflow is unlikely to occur at scales larger than the net infiltration model grid scale and hence would not change the spatial distribution of infiltration. NRC staff concludes that the uncertainty in areal-average net infiltration above the repository related to different spatial patterns of infiltration is small relative to other sources of uncertainty that the applicant considered. DOE showed that a reasonable alternative representation of spatial variability did not appreciably increase areal-average net infiltration.

In SER Section 2.2.1.3.5.3.2, NRC staff concludes that DOE's representations of temporal variability in climate and meteorology are both adequate for the intended use. The following discussion evaluates temporal variability in DOE's net infiltration estimates with respect to temporal resolution of net infiltration calculations and systematic effects on net infiltration.

The NRC staff considers temporal resolution to be an important factor for partitioning infiltration from runoff during precipitation events. This is because runoff calculations may be sensitive to the detailed representation of precipitation during storm events when the soil cannot accept all of the precipitation. By using a daily timestep in the net infiltration model simulations, DOE elected not to consider peak flow rates within a day, either as subsurface water movement or as runoff, as detailed in SNL (2007az, Section 6.2.3). As a consequence of not considering peak flow rates within a day, the spatial patterns of net infiltration may be affected by differences in runoff calculations. DOE supported the selected timestep by demonstrating in SNL (2007az, Appendix H) and DOE (2009cr, Enclosure 4) that calculated areal-average net infiltration has little sensitivity to uncertainty in the relationship between daily precipitation and precipitation duration. Because the treatment of temporal resolution of precipitation affects the partitioning between infiltration and runoff, which in turn affects the spatial patterns of infiltration, the NRC staff compared the uncertainty of different spatial patterns of infiltration with the other sources of uncertainty that DOE incorporated into the performance assessment. The NRC staff concludes that DOE acceptably represented temporal resolution in net infiltration calculations because (i) uncertainty in areal-average net infiltration above the repository related to different spatial patterns of infiltration is small relative to other sources of uncertainty that the applicant considered and (ii) NRC staff concluded in SER Section 2.2.1.3.6.3.2 that spatial variability itself does not significantly affect performance.

In summary, NRC staff concludes that DOE's representation of net infiltration for the initial 10,000 years is acceptable for use in performance assessment calculations. This is because the range and uncertainty of areal-average infiltration within the repository vicinity is consistent with (i) available information on infiltration in arid environments and (ii) NRC staff's independent modeling (Stothoff and Walter, 2013aa). Further, because saturated zone model results are less sensitive to uncertainty in net infiltration estimates than unsaturated zone model results, NRC staff's conclusions on adequacy of net infiltration results for unsaturated zone flow and transport also apply to the saturated zone. Additionally, available information shows that DOE's representation of spatial and temporal variability in net infiltration is likely to have a small influence on release, transport, and expected dose calculations compared to areal-average infiltration.

2.2.1.3.5.4 Evaluation Findings

The NRC staff has reviewed the SAR and other information submitted in support of the license application, which includes information required by 10 CFR 63.21(c)(1), (9), (10), and (15), and finds, with reasonable expectation, that the relevant requirements of 10 CFR 63.114, 10 CFR 63.305, and 10 CFR 63.342 are satisfied regarding the abstraction of climate and infiltration.

In particular, the NRC staff finds that DOE has adequately

- Included data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the surface and subsurface from the site and the region surrounding Yucca Mountain to define parameters and conceptual models used in the performance assessment calculation, in compliance with 10 CFR 63.114(a)(1)

- Accounted for uncertainty and variability in the parameter values used to model climate and infiltration, in compliance with 10 CFR 63.114(a)(2)

- Considered and evaluated alternative conceptual models that are consistent with currently available data and scientific understanding, in compliance with 10 CFR 63.114(a)(3)

- Provided a technical basis for the inclusion of FEPs affecting climate and infiltration, consistent with the limits on performance assessment in 10 CFR 63.342(a),(b), and evaluated in sufficient detail those processes that would significantly affect repository performance, in compliance with 10 CFR 63.114(a)(4–6)

- Provided technical bases for the models of climate and infiltration used in the performance assessment to represent the 10,000 years after disposal, in compliance with 10 CFR 63.114(a)(7)

The NRC staff finds that, with respect to the requirements of 10 CFR 63.305, for consideration of climate and infiltration, DOE has

- Used FEPs to describe the reference biosphere that are consistent with present knowledge of conditions in the region surrounding the Yucca Mountain site, in compliance with 10 CFR 63.305(a)

- Not projected changes in society, the biosphere (other than climate), human biology, or increases or decreases of human knowledge or technology, and has assumed that all of those factors remain constant as they are at the time of submission of the license application, in compliance with 10 CFR 63.305(b)

- Varied factors related to the geology, hydrology, and climate based upon cautious but reasonable assumptions of the changes in these factors that could affect the Yucca Mountain disposal system during the initial 10,000 years of the period of geologic stability in compliance with 10 CFR 63.305(c)

- Used biosphere pathways consistent with arid or semi-arid conditions, in compliance with 10 CFR 63.305(d)

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CHAPTER 9

2.2.1.3.6 Unsaturated Zone Flow

2.2.1.3.6.1 Introduction

Safety Evaluation Report (SER) Section 2.2.1.3.6 provides the U.S. Nuclear Regulatory Commission (NRC) staff's evaluation of the U.S. Department of Energy's ("DOE" or "applicant") abstraction of groundwater flow in the portion of the repository system above the water table. DOE presented this information in its Safety Analysis Report (SAR) of June 3, 2008 (DOE, 2008ab) and subsequent update of February 19, 2009 (DOE, 2009av). Although information from other sections of the SAR is cited in the NRC staff's review of the unsaturated zone flow abstractions, the primary SAR Sections used in the review are 2.3.2 (Unsaturated Zone Flow), 2.3.3 (Water Seeping into Drifts), and 2.3.5.4 (In-Drift Thermohydrological Environment). In SER Section 2.2.1.3.6, unsaturated zone flow abstractions include unsaturated zone flow, thermal conditions in the host rock, and in-drift thermohydrological conditions, excluding conditions for the engineered components.

The proposed Yucca Mountain repository site has up to 400 m [1,300 ft] of variably saturated rock between the ground surface and the repository and at least 200 m [650 ft] between the repository and the underlying water table (SAR Sections 2.1.1.1 and 2.1.1.3). Water percolating through the unsaturated zone may enter the drifts, thereby providing the means to interact with and potentially corrode the waste packages. DOE defined seepage as the water entering the drifts via dripping from the drift ceiling (SAR Section 2.3.3.2.1). Water percolating through the unsaturated zone below the repository also provides flow pathways for transporting radionuclides downward to the water table. Once radionuclides pass below the water table, they may move laterally within the saturated zone to the accessible environment. In this section (SER Section 2.2.1.3.6), the term "unsaturated zone flow" includes not only flow processes in the host rock under ambient and thermally perturbed conditions, but also in-drift hydrological processes related to flow through natural rubble and in-drift convection and condensation. Unsaturated flow, both above and below the repository horizon, is addressed in SER Section 2.2.1.3.6.

The unsaturated zone plays a role in two of the DOE-defined barriers: the Upper Natural Barrier and the Lower Natural Barrier (SAR Section 2.3.2). These barriers are reviewed in SER Section 2.2.1.1.3.2. Together with Climate and Infiltration (reviewed in SER Section 2.2.1.3.5), processes in the unsaturated zone above the repository comprise the Upper Natural Barrier. They influence system performance through the amount of water reaching the Engineered Barrier System and their control on hydrological conditions in the drift. In DOE's model of the nominal scenario, the Quantity and Chemistry of Water Contacting Engineered Barriers and Waste Forms (reviewed in SER Section 2.2.1.3.3), Degradation of Engineered Barriers (reviewed in SER Section 2.2.1.3.1), and Radionuclide Release Rates and Solubility Limits (reviewed in SER Section 2.2.1.3.4) abstractions use in-drift liquid water, relative humidity, and temperature to assess the potential for corrosion of waste packages, release of waste, and transport to the natural system. In the disruptive scenarios of seismic and igneous intrusion (reviewed in SER Sections 2.2.1.3.2 and 2.2.1.3.10), DOE's model uses the flux of water to assess the movement of radionuclides to the natural system below the repository. The portion of the unsaturated zone below the repository is part of the Lower Natural Barrier. The magnitude and distribution of flux in the unsaturated zone below the repository are used to determine the flow pathways for Radionuclide Transport in the

Unsaturated Zone (reviewed in SER Section 2.2.1.3.7). The unsaturated zone below the repository links the repository Engineered Barrier System to the Saturated Zone Flow and Transport System (reviewed in SER Sections 2.2.1.3.8 and 2.2.1.3.9) and ultimately to the biosphere in the accessible environment (reviewed in SER Sections 2.2.1.3.12 to 2.2.1.3.14).

DOE used several descriptions for the initial 10,000-year repository period in the SAR. The NRC staff determines that the difference between the phrase *the next [or initial] 10,000 years* and the phrases *10,000 years after (or following) disposal, emplacement, closure, or permanent closure* do not significantly affect estimates for deep percolation, seepage, condensation, and thermal perturbations, because the difference in the timing is small relative to the period of consideration.

The purpose of SER Section 2.2.1.3.6 is to evaluate repository performance with respect to DOE estimates of the

- Magnitude and distribution of the mass flux of water (percolation) moving through the unsaturated zone and reaching the drift

- Amount and distribution of liquid water seeping into the drift, contacting the engineered barriers (i.e., drip shield), and becoming available to carry radionuclides out of the drift and into the natural environment

- Environmental conditions inside the drift (i.e., temperature, relative humidity, and moisture redistribution and condensation)

- Magnitude and distribution of flux in the unsaturated zone below the repository, which is important for transport of radionuclides

2.2.1.3.6.2 Regulatory Requirements

The applicant is required to provide information specified in 10 CFR 63.21(c)(1), (9), (10), (15), and (19) that is related to the abstraction of unsaturated zone flow, thermal conditions in the host rock, and in-drift thermohydrological conditions excluding conditions for engineered components. The requirements in 10 CFR 63.114 (Requirements for Performance Assessments) and 10 CFR 63.342 (Limits on Performance Assessments) include postclosure performance assessment to demonstrate compliance with 10 CFR 63.113 (Performance Objectives for the Geologic Repository after Permanent Closure). Compliance with 10 CFR 63.113 is reviewed in SER Section 2.2.1.4.1.

The regulations in 10 CFR 63.114 require, in part, that a performance assessment

- Include data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the surface and subsurface from the site and the region surrounding Yucca Mountain [10 CFR 63.114(a)(1)]

- Account for uncertainty and variability in the parameter values [10 CFR 63.114(a)(2)]

- Consider and evaluate alternative conceptual models [10 CFR 63.114(a)(3)]

- Provide technical bases for either the inclusion or exclusion of features, events, and processes (FEPs), including effects of degradation, deterioration, or alteration processes of engineered barriers that would adversely affect performance of the natural barriers, consistent with the limits on performance assessment in 10 CFR 63.342, and evaluate in sufficient detail those processes that would significantly affect repository performance [10 CFR 63.114(a)(4–6)]

- Provide technical basis for the models used in the performance assessment to represent the 10,000 years after disposal [10 CFR 63.114(a)(7)]

The NRC staff's evaluation of inclusion or exclusion of FEPs is given in SER Section 2.2.1.2.1. Requirements for performance assessment for the initial 10,000 years following disposal are specified in 10 CFR 63.114(a). Requirements for the performance assessment methods for the time from 10,000 years through the period of geologic stability, defined in 10 CFR 63.302 as 1 million years following disposal, are specified in 10 CFR 63.114(b) and 10 CFR 63.342. These sections provide that through the period of geologic stability, with specific limitations, the applicant

- Use performance assessment methods consistent with the performance assessment methods used to demonstrate compliance for the initial 10,000 years after disposal [10 CFR 63.114(b)]

- Include in the performance assessment those FEPs used in the performance assessment for the initial 10,000-year period (10 CFR 63.342)

The requirements in 10 CFR 63.342(c)(2) pertain to the effects of climate change on performance for the period from 10,000 to 1 million years after disposal. In addition, the requirements in 10 CFR 63.342(c)(1) pertain to the effects of seismic and igneous activity on repository performance, subject to the probability limits in 10 CFR 63.342(a) and 10 CFR 63.342(b). Specific constraints on the analysis required for seismic and igneous activity are given in 10 CFR 63.342(c)(1)(i) and 10 CFR 63.342(c)(1)(ii).

The NRC staff's review of the SAR and supporting information follows the guidance in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa), Section 2.2.1.3.6, Unsaturated Zone Flow, as supplemented by additional guidance for the period beginning 10,000 years after disposal (NRC, 2009ab). The YMRP acceptance criteria for model abstractions that provide guidance for the NRC staff's review of the applicant's abstraction of climate and infiltration are

1. System description and model integration are adequate
2. Data are sufficient for model justification
3. Data uncertainty is characterized and propagated through the abstraction
4. Model uncertainty is characterized and propagated through the abstraction
5. Model abstraction output is supported by objective comparisons

In its review of the SAR and supporting information, the NRC staff used a risk-informed approach and guidance provided by the YMRP, as supplemented by NRC (2009ab), for aspects of unsaturated zone flow, thermal conditions in the host rock, and in-drift thermohydrological conditions (other than inside the engineered components) that are important to repository performance. The NRC staff considered all five YMRP criteria in its review of information provided by DOE. In the context of these criteria, only those aspects of the model abstraction that substantively affect the performance assessment results, as determined by the NRC staff, are discussed in detail in this SER section. The NRC staff's determination is based both on risk information provided by DOE and on the NRC staff's independent analyses and knowledge gained through experience.

2.2.1.3.6.3 Technical Review

The review of the technical information DOE provided for unsaturated zone, seepage, and in-drift hydrological conditions in this section is divided into six subsections. The first subsection is an overview of the DOE description of processes and models, and a summary of results for the entire unsaturated zone flow area of review. The overview (SER Section 2.2.1.3.6.3.1) provides context for and reviews the integration between models and results separately evaluated in the remaining five subsections (SER Sections 2.2.1.3.6.3.2–2.2.1.3.6.3.6) within the unsaturated zone. The remaining five subsections follow a natural flow through the unsaturated zone system: (i) ambient flow above the repository, (ii) thermohydrology, (iii) ambient and thermal seepage, (iv) in-drift hydrologic conditions, and (v) ambient flow below the repository.

2.2.1.3.6.3.1 Integration Within the Unsaturated Zone

In this section, the NRC staff's review covers a range of processes and features occurring at widely disparate spatial and temporal scales within the Upper and Lower Natural Barriers and, to a lesser extent, within the Engineered Barrier System (EBS). Within this area of review, the NRC staff evaluates aspects of repository performance pertaining to

- Unsaturated zone flow fields (SAR Section 2.3.2) above and below the repository

- Seepage into drifts (SAR Section 2.3.3)

- Hydrological aspects of the in-drift environment (SAR Section 2.3.5); in particular, the Multiscale Thermal-Hydrologic Model (SAR Section 2.3.5.4.1), the In-Drift Condensation Model (SAR Section 2.3.5.4.2), and the thermohydrologic response to the range of design thermal loadings (SAR Section 2.3.5.4.3)

The NRC staff evaluates the DOE treatment of interactions between liquid fluxes and engineered barriers inside drifts (i.e., drip shields, waste packages, and inverts) in SER Sections 2.2.1.3.3 and 2.2.1.3.4.

DOE's unsaturated zone flow models receive input from and provide output to several areas of review. SAR Section 2.3.1 provided spatially distributed net infiltration rates for the different predicted future climates for use as the top boundary flux of models in the unsaturated zone. For outputs, in-drift liquid-phase water, relative humidity, and temperature were used for abstractions of chemistry for the in-drift environment (SAR Sections 2.3.5.3, 2.3.5.5), corrosion of engineered components (SAR Section 2.3.6), and waste form degradation and in-drift transport (SAR Section 2.3.7) in the Total System Performance Assessment (TSPA). In SAR Section 2.3.5, feedback from thermal-hydrological-chemical models in the host rock during the thermal period provided information on the perturbation of hydrological properties caused by emplaced waste. The thermal period is the time period when the temperature of the host rock and EBS are above the ambient temperature of the host rock. The duration of the thermal period varies with the context because the extent of change for each hydrologic process or property is not expected to be the same as temperature changes. Output flow fields from the ambient unsaturated zone mountain-scale model (SAR Section 2.3.2), along with radionuclide flux from the EBS (SAR Section 2.3.7), were then used by the Radionuclide Transport in the Unsaturated Zone abstraction (SAR Section 2.3.8).

This SER subsection focuses on issues common to each of the unsaturated zone flow models, including (i) integration among those models, (ii) representative flow reduction through the mountain, and (iii) propagation of uncertainty in performance assessment calculations.

Integration of Unsaturated Zone Flow Models

DOE represented heat and mass transfer in the unsaturated zone using a variety of process-level models covering features and processes at a range of scales from millimeters to kilometers [fraction of inches to miles]. In addition to models that provide direct input, DOE used additional models to support aspects of conceptual model assumptions and parameter input. The models require different computational grids, different data needs, and different model support. Because the different models overlap in terms of features, processes, inputs, and outputs, the NRC staff reviewed the DOE integration between models. Because they reflect integration between models and affect performance, the NRC staff evaluated the spatial continuity of percolation flux (consistent propagation of high- and low-flux patterns) and quantification of barrier capability through the mountain.

The following list of DOE models, inputs, and outputs used in performance assessment calculations provides context for the NRC staff’s technical review that follows. The NRC staff’s evaluation in the next five subsections of SER Section 2.2.1.3.6.3 parallels this list, which follows the flow of water through the mountain.

Ambient Site-Scale Unsaturated Flow Model (SAR Section 2.3.2)

- Receives top flux boundary condition from net infiltration model
- Creates set of flow fields for TSPA
  - Above-repository flux distribution to the Multiscale Thermal-Hydrologic Model
  - Below-repository flow field for unsaturated zone transport (see last item in this list)
- Reviewed by the NRC staff in SER Sections 2.2.1.3.6.3.2 (above repository only)

Multiscale Thermal-Hydrologic Model (SAR Section 2.3.5.4)

- Composed of a set of five linked process-level thermal and thermohydrology models
- Creates a set of thermal response abstractions that provides
  - In-drift temperature and relative humidity for chemistry of seepage and corrosion of engineered barrier abstractions
  - Deep percolation field for seepage and chemistry models
  - Flux from host rock into invert for EBS transport
- Reviewed by the NRC staff in SER Section 2.2.1.3.6.3.3

Ambient and Thermal Seepage Models (SAR Section 2.3.3)

- Process-level models used to create seepage abstractions for TSPA that provide
  - Seepage fraction (number of waste packages getting wet) to EBS transport

    - Seepage flux to EBS transport
    - Temperature at drift wall, above which no seepage occurs
- Reviewed by the NRC staff in SER Section 2.2.1.3.6.3.4

In-Drift Convection and Condensation Models (SAR Section 2.3.5.4)

- Convection model provides dispersion coefficients to condensation model
- Condensation model provides flux rate and distribution to EBS transport
- Reviewed by the NRC staff in SER Section 2.2.1.3.6.3.5

Site-Scale Unsaturated Flow Below the Repository (see first item in this list)

- Provides flow field to unsaturated zone transport model
- Reviewed by the NRC staff in SER Section 2.2.1.3.6.3.6

The NRC staff reviews repository performance with respect to water flux and heat transfer within the unsaturated zone by separately evaluating the individual process models and their abstractions (SER Sections 2.2.1.3.6.3.2 through 2.2.1.3.6.3.6) and the overall integration among models (this section). The NRC staff recognizes that it may not be practical to use one process-level model to represent the entire suite of features and processes in the unsaturated zone for performance assessment calculations. The NRC staff notes that linking decoupled models is a standard practice for modeling complex systems that cover a wide expanse of scales where important dependencies from upstream to downstream models are adequately reflected in linked models. On the basis of NRC staff knowledge of general modeling practices and review of the information that links (is passed between) separate DOE models in SER Sections 2.2.1.3.6.3.2 through 2.2.1.3.6.3.6, the NRC staff concludes that the ensemble of DOE models is a reasonable approach for representing the unsaturated system for the performance assessment.

The NRC staff reviewed the information passed between the unsaturated zone models. DOE did not strictly follow continuity between models that relate to the transfer of water from the natural system into the emplacement drifts and back to the natural system. However, spatial continuity was approximated between DOE’s larger site-scale models and smaller drift-scale seepage models. DOE used five percolation bins to maintain continuity of flow above, through and below drifts for both ambient and thermal periods. Spatial detail is lost in the progression from the net infiltration model to the site-scale models and the seepage model, but use of the percolation bin approach ensures that high and low seepage zones correspond generally with high and low percolation zones above and below drifts. The NRC staff’s evaluations of the adequacy of spatial variability, upscaling, downscaling, and other linkages between models, as appropriate, are included in SER Sections 2.2.1.3.6.3.2 through 2.2.1.3.6.3.4. In those SER sections, the NRC staff concluded that DOE’s treatment of spatial variability is acceptable for performance assessment because the inclusion of more detailed variation did not significantly increase calculated dose. On the basis of the NRC staff’s review of the DOE technical bases and the NRC staff’s knowledge gained through experience, the NRC staff concludes that the flux of water is adequately integrated between the unsaturated zone models and between the natural system and the EBS.

The NRC staff’s review also considered the DOE implementation of barrier capabilities, represented by changes to percolation flux rates as water moves through the mountain from the ground surface into the drifts and onto the water table. Table 9-1 illustrates the quantitative reduction in flux from the ground surface to water entering the drift using flux averages over the repository footprint. Flux values in the table are from DOE (2010ai, Enclosure 1, Tables 1, 5, and 8), and seepage fraction values are from SAR Tables 2.1-6 to 2.1-9. The table also provides the component of the Upper Natural Barrier, primary features or processes, and the relevant SAR section for each step of the flux reduction.

DOE presented seepage flux values as volumetric flux over the area of a waste package, calculated by multiplying seepage flux values determined in units of volumetric flux per unit area by a waste package footprint that is 5.1 m [17 ft] long and 5.5 m [18 ft] wide (i.e., drift width).

To maintain consistent units of flux for this table, the NRC staff divided DOE-provided seepage flux values by the scaling factor described in the previous paragraph.
DOE (2010ai, Enclosure 1, Tables 5 and 8) used a million-year simulation to provide the seepage flux and fraction values at 10,000 years and for the 1-million-year period. The value for seepage fraction at 10,000 years in the million-year simulation may differ slightly from the simulation results used in TSPA calculations for 10,000-year dose estimates. Flux values of net infiltration through deep percolation retain the significant figures DOE presented. In Table 9-1, seepage fraction is that portion of the drifts where dripping is predicted to occur (also called the seeping environment).

Table 9-1. Quantitative Reduction in Flux From the Ground Surface to Water Entering the Drift Using Flux Averages Over the Repository Footprint

	Precipitation mm/yr*	Net Infiltration mm/yr*	Unsaturated Zone Site-Scale Top Boundary Net Infiltration mm/yr*	Deep Percolation mm/yr*	Seepage Repository Footprint	
					Flux mm/yr*	Fraction of Area
Component of Upper Natural Barrier	—	Topography and Soils	—	Unsaturated Zone	Unsaturated Zone	
Primary Feature or Processes	Semiarid Climate	Evapo-transpiration, Runoff, Infiltration	Uncertainty in Net Infiltration	—	Capillary Diversion and Vapor Barrier	
Section of SAR	2.3.1	2.3.1	2.3.2	2.3.2	2.3.3	
Thermal Period†	—	—	—	—	0	0
Initial 10,000 years, Nominal‡	296.7	38.88	21.37	21.74	2.0 {6.4}§	0.31
Initial 10,000 years, Seismic‡					2.3 {7.4}§	0.31
Post-10,000 years, Nominal	—	—	—	31.83	3.4 {8.5}§	0.40
Post-10,000 years, Seismic					15.5 {22}§	0.69

*Units: 25.4 mm/yr = 1 in/yr
†Thermal period defined by drift wall temperature > 100 °C [212 °F] (SAR Section 2.3.3.3.4).
‡Values of precipitation and percolation for initial 10,000 years are for glacial transition climate.
§Average flux for seeping environment is in brackets

The table includes DOE values for its nominal case (no disruptive events) and seismic ground motion scenario. In its igneous intrusion scenario (not included in the table), DOE stated that it conservatively assumed that seepage processes at the drift wall do not act as a barrier. Condensation rates are similarly not included in Table 9-1 because these fluxes are short lived. In DOE's model, condensation rates have no effect on performance because drip shields are predicted to remain intact beyond the several thousand years to 10,000 years of the thermal period. The NRC staff concluded in SER Sections 2.2.1.3.1.3.1 and 2.2.1.3.2.6 that DOE acceptably supported that drip shields are expected to remain intact for at least 12,000 years, and therefore will be able to divert all water away from waste packages. Representative thermal effects (peak waste package temperature, drift wall temperature, duration of drift wall temperatures at or above boiling) caused by waste package emplacement were summarized in SAR Tables 2.3.5-7 and 2.3.5-8 and are not reflected in Table 9-1.

NRC staff used the list of average flux values in Table 9-1 to confirm first-order continuity between net infiltration, deep percolation, and seepage in DOE's process models. The confirmation of first-order continuity by NRC staff is an evaluation of the magnitude of change in flux as new processes and features are introduced in the DOE models for the unsaturated zone. The NRC staff used DOE's glacial transition climate to represent the initial 10,000 years because (i) the monsoon and glacial transition climates have approximately equal flux rates and DOE's climate model represents these periods as encompassing most (94 percent) of the initial 10,000 years of performance and (ii) DOE described seepage as precluded by above-boiling conditions at the repository horizon during the preceding climate state. From the entries in Table 9-1, the NRC staff calculates that deep percolation is 7 percent of precipitation on average for the glacial transition climate in DOE's performance assessment model. This reduction comes from the two process-model steps that DOE used in its performance assessment: (i) net infiltration calculations, using the Mass Accounting System for Soil Infiltration and Flow (MASSIF) infiltration model, and (ii) net infiltration uncertainty calibration, using the unsaturated zone model top boundary net infiltration.

For the initial 10,000 years of performance

- The net infiltration flux calculated by DOE for the initial 10,000 years is a small fraction (approximately 13 percent) of total precipitation. The NRC staff concluded in SER Section 2.2.1.3.5 that the net infiltration estimates are adequately supported because (i) the estimates are generally consistent with infiltration for areas throughout the desert Southwest of the United States with similar climates and (ii) the infiltration rates were consistent with the NRC staff's independent analyses conducted over the past 15 years.

- DOE compared subsurface observations with simulations based on the site-scale ambient unsaturated flow model to calibrate the uncertainty in net infiltration, leading to a further reduction of 45 percent in the percolation flux. NRC staff concludes this further reduction of net infiltration flux is acceptable because DOE demonstrated that the reduction did not lead to an underestimation of dose (see SER Section 2.2.1.3.6.3.2).

- The average seepage flux calculated by DOE is approximately 10 percent of average percolation. Using this value, DOE calculated that 31 percent of waste package locations would become wet. The NRC staff concluded in SER Section 2.2.1.3.6.4 that an underestimate of seepage has a low consequence for repository performance because DOE predicts that the drip shields remain intact and prevent seepage water from contacting the waste packages well beyond 10,000 years (the NRC staff's evaluation of drip shield performance is in SER Sections 2.2.1.3.1.3.1 and 2.2.1.3.2.6).

For the post-10,000-year period, the NRC staff's review of percolation and seepage includes the seismic ground motion and igneous intrusion modeling scenarios because these two scenarios contribute most to DOE's predicted dose (see SER 2.2.1.4.1.3.3.1.1.2). In particular,

- Deep percolation values used for the post-10,000-year period adequately reflect those specified in the regulation (SER Section 2.2.1.3.6.3.2)

- The average seepage fraction for the seismic ground motion modeling case (69 percent) is sufficiently large such that increases are not expected to significantly increase expected dose (see SER Section 2.2.1.3.6.3.4)

- The average seepage flux estimated for the seismic ground motion modeling case is sufficiently large (47 percent of percolation) such that larger values would not significantly affect expected dose (see SER Section 2.2.1.3.6.3.4)

- DOE represented emplacement drifts as not forming a barrier with respect to seepage for the igneous intrusion modeling case; therefore, seepage is not underestimated (see SER Section 2.2.1.3.6.3.4)

On the basis of these considerations, the NRC staff concludes that the average values of flux reduction are consistent with the NRC staff's understanding of the features, processes, barrier capabilities of the unsaturated zone, and sensitivity to repository performance.

In summary, the NRC staff concludes that first-order flux variability between models is adequately incorporated throughout the set of unsaturated zone flow models discussed previously. Flux variability was evaluated both in terms of spatial continuity of high and low flux zones and in terms of reductions in average flux through the mountain. Because continuity of flux through the mountain and average reductions in flux are acceptable, the NRC staff concludes that DOE's collection of linked process models and abstractions for site-scale unsaturated flow and seepage is adequate for the performance assessment. Detailed NRC staff review of repository performance with respect to DOE results for net infiltration, percolation, seepage, and condensation is given in SER Sections 2.2.1.3.5.3 and 2.2.1.3.6.3.2 through 2.2.1.3.6.3.6.

Propagation of Uncertainty in Performance Assessment

The NRC staff evaluates propagation of uncertainty in DOE performance assessment calculations by examining (i) the technical basis for selecting parameters and uncertainties for sensitivity analyses and (ii) DOE's sensitivity analyses for the selected parameters with respect to TSPA intermediate results and calculated doses.

DOE conducted a series of analyses to determine the sensitivity of model outputs to input parameter uncertainties in its performance assessment calculations (SAR Section 2.4.2.3.3.3). The applicant treated a subset of parameters in the unsaturated zone and EBS as uncertain in the analyses. These parameters relate to infiltration (linked to percolation), seepage into the drift, host rock thermal and hydrogeologic parameters, and in-drift thermal processes that affect the release of radionuclides. In SER Sections 2.2.1.3.5.3.3 and 2.2.1.3.6.3.2 through

2.2.1.3.6.3.6, the NRC staff evaluated (i) the analyses used to identify uncertain parameters carried into performance assessment calculations and (ii) the technical basis for describing uncertain parameters when evaluating individual models and abstractions. In each of these sections, the NRC staff concluded that for use in the corresponding abstraction, the applicant adequately addressed model and parameter uncertainty in performance assessment calculations. On the basis of a comparison with important parameters identified in NRC (2005aa, Appendix D), the NRC staff concludes that DOE identified a reasonable set of the most important parameters to include in performance assessment calculations of dose. Because the uncertainty for the individual models is acceptable, the NRC staff concludes that the applicant adequately propagated uncertainty throughout the unsaturated zone.

The NRC staff notes that DOE sensitivity analyses indicate that the unsaturated zone is less important for performance compared to EBS components. DOE evaluated the sensitivity of intermediate results and expected mean annual dose as calculated by TSPA with respect to the selected parameters. In general, DOE's analyses identified key uncertain input parameters associated with waste package failure as the dominant factors affecting performance (SAR Table 2.4-12); saturated zone transport and net infiltration were also identified as key uncertain input parameters under some scenarios.

The applicant's analyses consistently identified uncertainty in net infiltration, which is closely related to percolation fluxes within the unsaturated zone, as significantly affecting uncertainty in intermediate results (e.g., drift seepage, drift wall temperatures, radionuclide releases from the EBS, unsaturated zone radionuclide transport rates) and expected mean annual doses (SNL, 2008ag). DOE's analyses also identified (i) relatively smaller contributions to uncertainty in seepage into drifts arising from uncertainty in host rock permeability and capillary strength and (ii) contributions to uncertainty in in-drift temperature and relative humidity from host rock thermal conductivity, as outlined in SNL (2008ag, Section K4).

The NRC staff examined DOE's sensitivity analyses with respect to intermediate results and expected mean annual dose by comparing the sensitivity results with the DOE description of the physical processes governing barrier function as represented in the models used for performance assessment. The NRC staff's review focused on parameters that DOE identified as systematically affecting either intermediate results or expected mean doses. On the basis of the NRC staff comparison, the NRC staff concludes that DOE's sensitivity analysis is consistent with the DOE description of the physical processes embodied in models used for performance assessment calculations with respect to net infiltration, percolation fluxes, thermal responses in the host rock, seepage, and in-drift temperature and humidity.

The NRC staff's conclusions on DOE's procedures for propagating uncertainty in performance assessment consider the reasonableness of overall performance assessment results. This is based in part on the NRC staff review of the performance assessment calculations (see SER Section 2.2.1.4.1). DOE's rankings of key parameter inputs are consistent with its representation of engineered barrier characteristics and are derived from a variety of approaches for sensitivity analyses, as described in SNL (2008ag, Appendix K9). The rankings suggest that failure and release mechanisms are more important to repository performance than natural system factors of the unsaturated zone because of the longevity of the engineered barriers. The NRC staff found DOE rankings of key parameter inputs to be similar to the NRC staff's parameter rankings derived from uncertainty analyses performed using independent methods and models (NRC, 2005aa).

On the basis of these considerations, the NRC staff concludes that DOE has adequately propagated data uncertainty associated with infiltration; seepage into the drift; host rock thermal and hydrogeologic parameters; and in-drift thermal, chemical, and mechanical processes that affect the release of radionuclides in its performance assessment.

2.2.1.3.6.3.2 Ambient Mountain-Scale Flow Above the Repository

The applicant represented the unsaturated zone and EBS using a hierarchy of far-field, near-field, and in-drift models. DOE uses the term far-field models to focus on features and processes of the natural system sufficiently distant from the repository to be unaffected by excavation and emplacement of waste. Similarly, near-field models focus on features and processes in the region affected by excavation and emplacement of waste (these terms are further described in the glossary). In-drift models focus on features and processes inside the disposal drift. DOE used its site-scale unsaturated zone flow model to represent far-field ambient mountain-scale flow from the ground surface to the water table (i.e., above, within, and below the proposed repository). In DOE documentation, site-scale and mountain-scale flow are interchangeable terms referring to the large scale of the computational grid, and ambient flow is the percolation flux that occurs without the flow-diverting effects of drifts and waste-produced thermal boiling fronts. DOE used output from the site-scale unsaturated zone flow model to account for far-field effects. This SER section evaluates repository performance with respect to ambient water flow within the upper unsaturated zone between the ground surface and the proposed repository horizon, focusing on the site-scale unsaturated zone flow model and an intermediate-scale model that links mountain-scale flow to seepage models. The NRC staff evaluates repository performance related to the effects of drift openings and thermal perturbation on flow patterns within and below the proposed repository horizon in SER Sections 2.2.1.3.6.3.3 through 2.2.1.3.6.3.6.

In its review, the NRC staff considered the site-scale unsaturated-zone flow model from two perspectives: (i) in the context of flow within the unsaturated zone as a whole and (ii) in the context of the applicant-defined upper and lower natural barriers. The NRC staff's evaluation in this SER section focuses on aspects of repository performance primarily related to the upper natural barrier, in particular, aspects of flow that affect seepage. The evaluation presented in SER Section 2.2.1.3.6.3.6 addresses aspects of repository performance primarily related to the lower natural barrier; in particular, aspects of flow that affect transport. The present section also considers some aspects of repository performance that apply to the entire unsaturated zone, such as estimation of parameter value sets for models. This SER section also describes the NRC staff's evaluation of aspects of repository performance specifically related to DOE's implementation of 10 CFR 63.342(c)(2), regarding percolation during the period from 10,000 to 1 million years after disposal.

Conceptual Model

The applicant's conceptual model for flow in the unsaturated zone is based on the primary hydrogeologic units within a column (SAR Section 2.3.2.2.1). The uppermost unit, the Tiva Canyon welded unit, features vertical episodic and fracture-dominated flow strongly influenced by episodic infiltration pulses. The underlying Paintbrush Tuff nonwelded (PTn) unit exhibits essentially vertical matrix-dominated flow with a strong potential for dampening and smoothing flows, thereby buffering lower units from episodic and localized infiltration pulses. The repository host horizon is within the Topopah Spring welded (TSw) unit, which features essentially vertical fracture-dominated flow in equilibrium with decadal-average net infiltration. In the TSw, mountain-scale flow patterns are controlled by mountain-scale infiltration patterns and fine-scale flow patterns are controlled by the TSw rock properties. The DOE conceptual model for the units underlying the repository host horizon [Calico Hills nonwelded (CHn) and Crater Flat undifferentiated (CFu) units] is evaluated in detail in SER Section 2.2.1.3.6.3.6.

The NRC staff concludes that DOE acceptably described its conceptual model of the physical phenomena affecting unsaturated flow under ambient conditions because DOE described in appropriate detail (i) the features of the site geology that affect unsaturated zone flow; (ii) the most important processes affecting the performance of the upper natural barrier, such as lateral diversion, spatial variability, and temporal variability; and (iii) the body of laboratory and field data supporting the conceptual model. The NRC staff also finds DOE's description of its conceptual model of the physical phenomena to be acceptable because DOE's description is consistent with the NRC staff understanding of the Yucca Mountain natural system, obtained from extensive prelicensing experience and independent analyses of unsaturated zone flow processes at Yucca Mountain (e.g., NRC, 2005aa).

Implementation of the Conceptual Model

DOE used several models to represent various aspects of flow within the upper unsaturated zone, each considering different processes and scales. Because several DOE abstractions consider flows in the upper unsaturated zone, and multiple downstream models depend on the calculated flows, the NRC staff considered how the flow abstractions interact with the other calculations affecting repository performance by (i) separately evaluating individual flow abstractions, (ii) evaluating downstream uses of the flow abstractions for consistency between abstractions, and (iii) evaluating the effects of the abstractions on performance assessment calculations as a whole. In this SER section, the NRC staff evaluates the numerical ambient mountain-scale flow model, focusing on the linkages between the model and both upstream models (i.e., the infiltration model) and downstream models (e.g., the Multiscale Thermal-Hydrologic Model and seepage models).

In performing a risk-informed, performance-based review of the DOE's representation of ambient flow in the upper unsaturated zone, the NRC staff considered the DOE-defined function of the Upper Natural Barrier as preventing or substantially reducing water seepage into emplacement drifts. On the basis of the function of the barrier, the NRC staff identified several aspects of the DOE abstractions as having primary importance for representing flow in the upper unsaturated zone. These aspects include

- Integration of flow models; in particular, integration between the mountain and seepage scales, because integration directly affects modeling approaches in upstream and downstream models

- Mountain-scale flow patterns within the upper unsaturated zone, because this characteristic affects the amount of water diverting away from the repository as a whole

- Site-scale unsaturated zone flow model parameters, because model properties may affect flow patterns and thereby affect the rate of water seepage into drifts or the thermal response of the natural system

- Systematic effects of infiltration uncertainty on percolation fluxes, because percolation rates at the repository horizon directly affect the rate of water seepage into drifts, and percolation rates below drifts directly affect the rate of radionuclide transport from the EBS to the water table

- Temporal and spatial variability; in particular, flow patterns at the drift walls, because these flow characteristics affect the rate and patterns of water seepage into drifts

The following sections describe the NRC staff's evaluation of these aspects of repository performance with respect to ambient flow in the upper unsaturated zone.

Integration of Flow Models

The NRC staff considered the applicant's approach to integrating the site-scale unsaturated zone flow model representation of ambient flow above the repository with (i) the MASSIF infiltration model (SAR Section 2.3.1), (ii) the Multiscale Thermal-Hydrologic Model (SAR Section 2.3.5.4), (iii) seepage models, and (iv) ambient flow below the repository. The NRC staff identified the integration of ambient flow above the repository with seepage models as the most risk significant among these for the following reasons:

- The MASSIF net infiltration model, site-scale unsaturated zone flow model, and Multiscale Thermal-Hydrologic Model are decoupled models that are linked in only one direction (see SER Section 2.2.1.3.6.3.1); the site-scale unsaturated zone flow model seamlessly considers ambient flow above and below the repository, and the model linkages locally conserve mass.

- Performance assessment calculations are sensitive to seepage, which is in turn sensitive to the representation of fluxes near emplacement drifts at scales smaller than those represented by the site-scale unsaturated zone flow model.

The NRC staff concludes that DOE adequately integrated the site-scale unsaturated zone flow model representation of ambient flow above the repository with the MASSIF infiltration model and the Multiscale Thermal-Hydrologic Model and ambient flow below the repository because the linkages locally conserve mass and the overlapping input properties are generally consistent between the models. The NRC staff reviewed input properties that are not consistent (e.g., hydraulic properties of bedrock near the ground surface) between the infiltration and site-scale ambient unsaturated flow models. On the basis of the NRC staff's knowledge and experience with developing input property sets for the different needs of infiltration and unsaturated zone models (Stothoff, 2013aa,ab; Manepally, et al., 2004aa) and knowledge of scaling of properties between different spatial scales, the NRC staff concludes the differences are technically justified or not significant for the infiltration and unsaturated zone flow models and their integration.

In considering integration between models, the NRC staff identified a potential concern with integration between the infiltration and seepage models (even though the two models are not directly linked) because (i) the MASSIF calculations are performed at a finer spatial and temporal resolution than the site-scale unsaturated zone flow model, (ii) modeled spatial and temporal variability in flow is reduced in the transfer of information between the two models, and (iii) spatial and temporal variability in flow near emplacement drifts affects the DOE seepage calculations.

The NRC staff evaluates DOE's integration between flow at the mountain and seepage scales, also considering integration between infiltration and seepage, in the remainder of this subsection. The NRC staff evaluated DOE's approach to modeling seepage in SER Section 2.2.1.3.6.3.4. For the reasons discussed in the following paragraph, the NRC staff concludes that DOE adequately integrated these models.

DOE used a hierarchical approach to link percolation fluxes within the repository horizon at the mountain scale to seepage, progressing from mountain-scale flow to the smaller scales of intermediate- or drift-scale flow and fine-scale flow (SAR Section 2.3.3.2). The site-scale unsaturated zone flow model, which calculates mountain-scale flow, represents flow averaged over an area approximately corresponding to the combined area of a drift and the pillar between drifts. Intermediate-scale flow represents average flow over an area approximately corresponding to the width of a drift. Fine-scale flow represents flow at scales smaller than a drift wall. DOE considered all three scales in calculating average seepage into drifts.

DOE represented both intermediate-scale and fine-scale flow statistically, as described in SAR Section 2.3.3.2.3 and DOE (2009bo, Enclosure 4). DOE derived an abstraction for the statistical distribution of intermediate-scale fluxes from numerical model calculations considering hydraulic-property heterogeneity in the densely welded TSw unit above the proposed repository using model parameters based on field observations. DOE provided simulation results demonstrating that the statistical patterns of predominantly vertical unsaturated zone flow at the intermediate scale stabilize within a short vertical distance from the top boundary and geological unit changes, using several representations of the top boundary flux (SAR Section 2.3.3.2.3.5). DOE also performed sensitivity studies considering alternative statistical representations of intermediate-scale variability. The NRC staff evaluated these sensitivity studies (see subsection on Temporal and Spatial Variability in the Upper Unsaturated Zone later in this section) and concluded that DOE demonstrated that the dose consequences from alternative statistical representations of intermediate-scale variability (including potential links between infiltration and seepage) are not significant to performance.

The NRC staff concludes that DOE adequately integrated mountain-scale flow and seepage for the following reasons:

- The procedure DOE used to pass flow from the mountain scale to the intermediate scale to the fine scale does not propagate flow diversion near the drift wall to larger scales, so seepage calculations use upper-bound estimates for local percolation flux.

- In performance assessment calculations, DOE used statistical distributions of relevant hydrological input parameters at a smaller scale that conserve fluxes calculated at the larger scale (e.g., mountain-scale flow model to intermediate-scale model, intermediate-scale model to seepage model).

- DOE used site observations to derive the statistical distributions for rock properties used as input to the intermediate-scale model linking the mountain-scale and seepage models.

- The performance assessment calculations are not strongly sensitive to the representation for intermediate-scale fluxes. DOE demonstrated that, for a given areal-average seepage flux, calculated peak cumulative areal-average radionuclide releases would not be underestimated using the performance assessment procedure outlined in DOE (2009bo, Enclosures 2 and 4). Further, DOE demonstrated that alternative statistical representations for the intermediate-scale flux linking mountain-scale and seepage fluxes do not significantly affect performance (see subsection on Temporal and Spatial Variability in the Upper Unsaturated Zone later in this section).

Mountain-Scale/Site-Scale Flow Patterns

The site-scale unsaturated zone flow model represents mountain-scale flow as steady state in equilibrium with climatic conditions, with water flowing from the upper boundary to the water table without lateral inflow or outflow. The NRC staff identified the key uncertainties in mountain-scale flow as (i) the magnitude of areal-average percolation flux through the repository footprint and (ii) the spatial patterns of percolation flux through the repository footprint. The NRC staff evaluates the site-scale model separately for the Upper Natural Barrier portion of the unsaturated zone, above the proposed repository (this section), and the Lower Natural Barrier portion of the unsaturated zone, below the repository (SER Section 2.2.3.6.3.6). The NRC staff evaluates DOE's representation for the spatial patterns of percolation flux in this subsection and evaluates DOE's representation for the magnitude of areal-average percolation flux through the repository footprint, which is determined by net infiltration, later in this section (see subsection on Infiltration Uncertainty).

DOE represents spatial variability in percolation fluxes as dominated by intermediate- and fine-scale variability in the seepage calculations for performance assessment. In the preceding subsection of this section (Integration of Flow Models), the NRC staff concluded that DOE acceptably integrated mountain-scale flow and seepage, and that DOE acceptably accounted for alternative intermediate- and fine-scale spatial patterns of percolation fluxes within the repository footprint (and as a logical consequence acceptably accounting for alternative mountain-scale spatial patterns that preserve the areal-average percolation within the repository footprint). Therefore, the NRC staff's review of the site-scale unsaturated zone flow model representation of mountain-scale flow patterns focuses on the applicant's representation of potential systematic mountain-scale flow diversion above the proposed repository horizon.

DOE demonstrated that the site-scale unsaturated zone flow model calculates mountain-scale spatial patterns of percolation fluxes, combining matrix and fracture flows, that are essentially vertical from the ground surface through the repository horizon within the repository footprint, in accordance with DOE's conceptual model, as shown in SNL (2007bf, Figures 6.1-2 through 6.1-5 and 6.6-1 through 6.6-4). DOE's numerical analyses indicate that substantial lateral diversion (associated with the PTn unit) away from the repository footprint may occur, which would reduce the amount of water passing through the footprint at the repository horizon (SAR Section 2.3.2.2.1.1).

The NRC staff concludes that DOE adequately considered uncertainty in ambient mountain-scale flow patterns above the repository horizon in performance assessment calculations for the following reasons:

- The site-scale unsaturated zone flow model represents mountain-scale percolation fluxes consistent with DOE's conceptual model and the NRC staff's understanding of the physical system. Further, DOE demonstrated that alternative flow patterns with the same areal-average percolation within the repository footprint do not significantly affect repository performance.

- DOE demonstrated that alternative representations for flow above the repository horizon tend to increase lateral diversion out of the repository footprint. Diverting percolation from the repository footprint would reduce percolation fluxes through the repository horizon and thereby reduce calculated seepage estimates.

Site-Scale Unsaturated Zone Model Parameters

The NRC staff's risk-informed, performance-based review of the site-scale unsaturated zone model parameters includes consideration of (i) the technical bases for the parameters and (ii) the use of the parameters in the site-scale unsaturated zone model and downstream models used for performance assessment calculations. DOE does not base input parameters for the downstream seepage models on site-scale unsaturated zone model parameters; the NRC staff evaluates the seepage model input in SER Section 2.2.1.3.6.3.4.

DOE's site-scale unsaturated zone flow model represents the stratigraphy at Yucca Mountain using 32 homogeneous material property layers, with the same set of material properties assigned to every grid cell in a property layer (SAR Section 2.3.2.4.1.2.2). The material property layers were developed from geologic layers described in the Geologic Framework Model. DOE justified the use of layerwise homogeneous material properties on numerical simulations by comparing outputs from simulations with different levels of heterogeneity, concluding that similarities in fracture flux patterns and tracer transport times demonstrate that heterogeneities within units have only a minor effect on site-scale flow processes (SAR Section 2.3.2.4.1.1.4). DOE developed some model parameters directly from site observations (e.g., porosity) and others from model calibration to field measurements of saturation, potential, pneumatic pressure, and perched water elevations. DOE used a set of *in-situ* observations not used for model calibration (e.g., calcite, Carbon-14, and strontium) to support the flow model (SAR Section 2.3.2.5.1). DOE further used a set of *in-situ* observations not used for either model calibration or support of the flow model (chloride and temperature observations) to calibrate the uncertainty in infiltration (SAR Section 2.3.2.4.1.2.4.5).

The NRC staff concludes from review of the SAR and supporting documentation, as discussed in the previous paragraphs, that (i) the model parameters are based on site-specific information and (ii) DOE clearly documented the procedures and bases for developing the parameters.

The NRC staff finds that DOE's performance assessment explicitly considered uncertainty in model parameters arising from uncertainty in infiltration flux, using a separate set of calibrated properties for the 10^{th}, 30^{th}, 50^{th}, and 90^{th} percentiles of the infiltration map uncertainty distribution. DOE ensured that the flow fields resulting from each set of calibrated properties featured predominantly vertical flow from the ground surface through the repository horizon within the repository footprint, in accordance with the DOE conceptual model. DOE did not propagate the full range of parameter uncertainties into the TSPA on the basis that sensitivity analyses demonstrated that the model results, including performance-affecting results such as flow pathways, are insensitive to the parameter values in the range that DOE considered (SAR Section 2.3.2.4.2.2).

The NRC staff concludes that DOE adequately considered site-scale unsaturated zone model parameter uncertainty with respect to the Upper Natural Barrier representation in performance assessment calculations for the following reasons:

- The downstream feed for the site-scale unsaturated zone flow model as used in performance assessment calculations (percolation flux at the base of the PTn unit) is approximately equal to net infiltration under predominantly vertical flow.

- The Multiscale Thermal-Hydrologic Model (SER Section 2.2.1.3.6.3.3) and the near-field chemistry model (SER Section 2.2.1.3.3.3.1) use hydraulic parameters based on the site-scale unsaturated zone model that were derived from measurements at Yucca Mountain and on equivalent units from nearby locations.

Infiltration Uncertainty

The NRC staff evaluates infiltration uncertainty by considering (i) the technical bases for DOE's approach and (ii) consequences of the approach in performance assessment calculations. DOE addressed uncertainty in infiltration estimates both derived from the net infiltration model using input parameter uncertainty and from infiltration uncertainty from deep subsurface observations. The NRC staff evaluates the latter type of infiltration uncertainty in this section. DOE also referred to the results from incorporating this uncertainty as the unsaturated zone top (upper) boundary net infiltration to distinguish it from results derived from the net infiltration model.

DOE described three climate states for the initial 10,000 years. These climates, present-day, monsoon, and glacial transition, are evaluated in SER Section 2.2.1.3.5.2. For each climate state, DOE selected four infiltration maps calculated by the MASSIF infiltration model, out of the 40 equally likely realizations for the climate state (SNL, 2007az), to represent the uncertainty in infiltration. The initial probabilities of 0.2, 0.2, 0.3, and 0.3 were assigned to the four selected maps under each climate state, on the basis of the percentile rankings for the infiltration maps. Because the site-scale unsaturated zone model uses larger grid cells than the MASSIF model, DOE created upper boundary net infiltration maps for the site-scale unsaturated zone model grid. DOE transferred the infiltration values for each site-scale unsaturated zone model grid cell by accumulating infiltration fluxes from nearby MASSIF grid cells. For the post-10,000 year climate, DOE assigned weights to the upper boundary net infiltration maps to be consistent with the NRC staff's proposed rule for deep percolation (NRC, 2005af) describing deep percolation in the post-10,000 year period. The NRC staff evaluates DOE's treatment of net infiltration during the post-10,000-year period later in this section (see subsection on Post-10,000-Year Approach), which also includes the NRC staff's evaluation of DOE's results considering changes between the proposed and final rule.

Another area of potential uncertainty concerned unsaturated zone chloride distributions. DOE found that model predictions for temperature and chloride distributions in the unsaturated zone, using models based on the site-scale unsaturated flow model, did not reflect average measured quantities in the unsaturated zone. DOE used a version of the Generalized Likelihood Uncertainty Estimation methodology (Beven and Binley, 1992aa) to update the initial probability weights assigned to each flow field (SAR Section 2.3.2.4.1.2.4.5). The Generalized Likelihood Uncertainty Estimation methodology uses likelihood functions to revise initial probability weights on the basis of differences between field observations and numerical model predictions. A likelihood function is a statistical model that is used to describe how likely a population represents observed values. DOE updated the probability weights using temperature observations from 5 boreholes and chloride observations from 12 boreholes, the Exploratory Studies Facility, and the Enhanced Characterization for the Repository Block Cross Drift. DOE used four different likelihood functions to compare the model predictions with observations.

SAR Section 2.3.2.4.1.2.4.5.5 reported estimated final weighting factors of 0.619, 0.157, 0.165, and 0.0596 for the 10th, 30th, 50th, and 90th percentile upper boundary net infiltration maps, respectively, by combining the analyses considering temperature and chloride data. These final weights, when applied to the upper boundary condition infiltration maps of the ambient site-scale unsaturated zone model, result in weighted-average infiltration over the repository footprint of 8.46, 16.00, and 21.37 mm/yr [0.33, 0.63, and 0.84 in/yr] under present-day, monsoon, and glacial transition climate states as recorded in DOE (2010ai, Enclosure 1, Table 1). For comparison, the initial weights applied to the upper boundary condition infiltration maps result in weighted-average infiltration over the repository footprint of 17.31, 38.12, and 38.88 mm/yr [0.68, 1.50, and 1.53 in/yr] for the same climate states, also recorded in DOE (2010ai, Enclosure 1, Table 1). The values using the initial weights are larger than the Generalized Likelihood Uncertainty Estimation-weighted infiltration by factors of 2.0, 2.4, and 1.8, respectively.

The NRC staff concludes from review of the SAR and supporting documentation, summarized in the preceding paragraphs, that (i) the representation of infiltration uncertainty is based on site-specific information and (ii) DOE documented transparently the procedures and bases for developing the infiltration uncertainty representation.

DOE identified several uncertainties in interpreting field observations using the temperature and chloride models (SAR Section 2.3.2.4.1.2.4.5.2). These uncertainties may affect the calculated probability weights for the assigned flow fields. DOE also demonstrated that there is uncertainty associated with the likelihood functions used to determine the weights by showing that the calculated weights varied depending on the selection of a likelihood function (SAR Tables 2.3.2-25 through 2.3.2-27).

DOE considered expected doses in the initial 10,000 years of performance for the seismic ground motion and igneous intrusion scenarios, which collectively account for approximately 97 percent of the calculated peak expected mean annual dose, as described in DOE (2009bo, Enclosure 5). DOE compared the performance assessment dose calculations of expected doses with and without the Generalized Likelihood Uncertainty Estimation weighting, finding that the original weighting scheme (without the Generalized Likelihood Uncertainty Estimation procedure) results in a 29 percent greater peak expected dose than the Generalized Likelihood Uncertainty Estimation weights. DOE concluded that, because the calculated doses are so similar, adjusting the infiltration uncertainty with the Generalized Likelihood Uncertainty Estimation procedure does not significantly affect performance assessment results.

The NRC staff concludes that the applicant's performance assessment calculations adequately represent the uncertainty in net infiltration for the following reasons:

- The NRC staff concludes in SER Section 2.2.1.3.5 that DOE's representation of net infiltration obtained from the MASSIF net infiltration model is adequate for use in performance assessment calculations.

- The NRC staff concludes in a preceding subsection (Integration of Flow Models) that the MASSIF net infiltration model results are adequately integrated with the site-scale unsaturated zone flow model.

- DOE demonstrated that the consequence of adjusting net infiltration uncertainties using the Generalized Likelihood Uncertainty Estimation procedure is not significant for predicted repository performance.

Temporal and Spatial Variability in the Upper Unsaturated Zone

DOE described a primary role for the upper unsaturated zone (i.e., above the proposed repository) as strongly dampening and smoothing episodic infiltration pulses, to the extent that flows below the PTn within the proposed repository footprint are essentially steady, in equilibrium with the long-term climate. This is a screening argument for FEP 2.2.07.05.0A, Flow in the Unsaturated Zone (UZ) from Episodic Infiltration (SNL, 2008ab). DOE identified the consequence of this dampening effect as reducing time-averaged seepage rates into intact or degraded drifts, with the reduction effect becoming less significant as the drift degrades, in DOE (2009an, Enclosure 2). NRC staff concluded in SER Section 2.2.1.2.1 (see FEP 2.2.07.05.0A) that episodic flow could be excluded on the basis of low consequence. The NRC staff's evaluation of temporal variability in the following paragraphs addresses longer temporal scales than the event-based or seasonal to annual episodic infiltration considered for FEP 2.2.07.05.0A.

The NRC staff's evaluation of DOE's treatment of temporal and spatial variability considered the potential consequences arising from partial breaches in the barrier capability represented by the upper unsaturated zone. The NRC staff identified effects on cooling of the EBS, transport, and seepage as potential consequences resulting from a partial breach. The NRC staff focused on the consequences with respect to seepage because in DOE's performance assessment calculations (i) conduction is the dominant thermal transport mechanism within the host rock, (ii) water moves from the repository horizon to the water table quickly relative to the performance period, and (iii) changes in percolation flux may have a nonlinear effect on seepage and release calculations. The NRC staff recognizes that temporal and spatial variability in percolation flux may nonlinearly affect release calculations in counteracting ways, because seepage may disproportionately increase with percolation. However, because of protection from the drip shield and waste package barriers, release rates may increase less than proportionately with seepage.

The NRC staff identified potential concerns with DOE's representation of temporal and spatial variability in percolation flux at the repository horizon for use in performance assessment stemming from (i) long-term (decadal to centennial) fluctuations around the mean climate, (ii) drift-scale percolation flux variability, and (iii) short-term (episodic or seasonal) fluctuations in deep percolation fluxes. The NRC staff was concerned that DOE may not have considered the full range of uncertainty for these processes in its performance assessment calculations. Each of these three items is discussed in the following paragraphs.

The NRC staff considered the effects of DOE's representation of long-term fluctuations about the mean climate by considering the consequences of those fluctuations on seepage. DOE considered decadal to centennial variability in percolation likely to occur below the PTn because the PTn has a finite storage capacity but expects that the increase in calculated average seepage would be small if decadal to centennial variability was explicitly included. DOE demonstrated that fluctuations in percolation flux of 20 and 50 percent about the mean (i.e., coefficients of variation of 0.2 and 0.5) yielded a systematic increase in Generalized Likelihood Uncertainty Estimation-weighted seepage of approximately 2.7 and 17 percent, respectively, under glacial transition conditions, as outlined in DOE (2009cc, Enclosure 1). DOE demonstrated that a systematic increase in Generalized Likelihood Uncertainty Estimation-weighted seepage by a factor of approximately 2.5 has a negligible effect on the expected dose (DOE, 2009bo, Enclosure 5). DOE concluded by analogy that the smaller systematic increases in seepage induced by decadal to centennial climatic fluctuations also would have a negligible effect on performance assessment results. The NRC staff concludes that DOE acceptably represented long-term (decadal to centennial) climatic fluctuations in performance assessment calculations because DOE demonstrated that such fluctuations have a negligible effect on the expected dose.

The NRC staff determines that the approach in DOE (2009bo, Enclosure 5), along with independent analysis (described in Stothoff, 2010aa, Section 3) that considered a more extreme range of decadal- to centennial-scale fluctuations, provides support for the conclusion that DOE acceptably represented long-term fluctuations in the performance assessment. Using a coefficient of variation of 0.8, which roughly corresponds to infiltration variability over a glacial cycle, as shown in Stothoff and Walter (2013aa, Table 4-2), the increase in seepage is smaller than the systematic increase in seepage that is attributed to the Generalized Likelihood Uncertainty Estimation approach, as detailed in DOE (2009cc, Enclosure 1).

DOE relied on numerical simulations to justify the representation of spatial variability in performance assessment calculations and to demonstrate that episodic flow is expected to be rare below the PTn. In these simulations, the NRC staff notes that DOE’s calculated spatial variability in percolation is dependent on the estimated heterogeneity of input parameters, which DOE did not explicitly tie to all sources of spatial variability potentially indicated by site observations. DOE described several mechanisms to explain why some field evidence, such as modern tritium observations below the PTn, may not completely support the DOE assumptions related to drift-scale variability and episodic flow. These mechanisms include transport through faults, episodic flow, spatially variable infiltration patterns, and heterogeneous hydrologic properties, as outlined in DOE (2009cc, Enclosure 1).

To evaluate uncertainty in spatial variability, the NRC staff considered possible sources of spatial variability in percolation between the mountain- and drift-scale that may not have been incorporated into the DOE models, such as intermediate-scale structural and stratigraphic features. The NRC staff considered the performance consequence of alternative representations of spatial variability using DOE’s analyses reflecting intermediate-scale percolation flux variability. In those DOE analyses, the effect on seepage was considered for several alternative statistical relationships for the effect of rock heterogeneity on flow focusing, as shown in SNL (2007bf, Section 6.8.2, Case 6), ranging from no intermediate-scale focusing (Case 6a) to flow focusing that is more extreme than those used for performance assessment calculations (Case 6c). DOE characterized the Case 6c distribution, which has a maximum flow-focusing factor more than six times larger than the distribution used in performance assessment calculations, as unrealistically extreme to represent rock heterogeneity, as described in DOE (2009bo, Enclosure 4). The NRC staff treated the Case 6c distribution as a surrogate that bounds spatial distribution not included in DOE’s model. DOE demonstrated that incorporation of the Case 6c distribution yields a minor increase in expected dose for the million-year seismic ground motion modeling case, as shown in DOE (2009cx, Enclosure 1). DOE (2009cx, Enclosure1) described the seismic ground motion modeling case as the dominant contributor to calculated expected dose (see SER Section 2.2.1.4.1.3.3.1.1.2). Because DOE demonstrated that the calculated peak cumulative areal-average releases are not strongly affected by an extreme increase in spatial variability, the NRC staff concludes that DOE acceptably represents spatial variability between mountain and drift scales for performance assessment calculations.

The NRC staff evaluated the consequences of DOE’s representation of flow as steady at the repository horizon, instead of episodic, by considering the field and modeling information DOE presented as a whole. DOE represented the PTn unit as a barrier with a strong potential to dampen episodic pulses below the PTn. This conceptual model was supported by numerical modeling results interpreting chloride, temperature, and radioisotope data obtained from within and below the PTn. DOE estimated that approximately 1 percent of the repository is affected by fast pathways, predominantly in faults (SAR Section 2.3.2.2.1.1), and argued that these fast pathways are not necessarily a consequence of episodic flow, but instead would represent spatial variability. The NRC staff finds that the information DOE presented (i) demonstrates that the PTn has the potential to strongly dampen episodic pulses below the PTn and (ii) suggests that any areas at the repository horizon that exhibit episodic pulses are likely to require a combination of flow focusing and fast pathways through the PTn to overcome the PTn dampening potential. As a consequence, DOE's data and modeling results indicate that, if episodic pulses do penetrate the PTn, relatively few waste packages are likely to be affected by these episodic pulses. Therefore, the NRC staff finds that the effect of episodic percolation can be evaluated by an analysis of the effect of the spatial extent of fast pathways that breach the barrier capability of capillary diversion at the drift (i.e., seepage becomes equal to percolation).

The NRC staff considered potential consequences of episodic flow using information in DOE (2010ai, Enclosure 1, Tables 4 and 5) to perform a calculation that assumes an extreme representation of the extent of episodic flow (Stothoff, 2010aa, Section 3). In the NRC staff analysis, NRC staff assumes that (i) episodic flow completely avoids the capillary barrier and becomes seepage, (ii) episodic flow pathways occur under the largest fluxes, and (iii) the area experiencing fast pathways is five times larger than DOE estimated. Each of these assumptions conservatively represents an aspect of the DOE seepage abstraction. Under these assumptions, all of DOE's Percolation Bin 5 (highest flux bin) is modeled as completely ineffectual at preventing seepage (seepage rates are equal to percolation rates). For comparison, seepage rates are less than 7 percent of infiltration rates for Percolation Bin 5 in DOE's model under glacial-transition conditions. With these assumptions, the NRC staff calculates an areal-average seepage rate 2.4 to 3 times larger than DOE used in performance assessment calculations, which is less than the 3.9 times increase in total seepage that DOE calculated in its analysis of the Case 6c flow-focusing distribution. As previously stated, DOE demonstrated that this order of increase in total seepage does not significantly increase calculated dose.

The NRC staff, therefore, concludes that repository performance is not strongly affected by the DOE representation of episodic flow for performance assessment calculations because (i) DOE demonstrated that repository performance is not strongly affected by increased seepage from increased infiltration or increased spatial variability and (ii) NRC independent analysis of an extreme case demonstrated that the increase in seepage that might occur from episodic flow results in a similarly small increase in seepage.

Therefore, the NRC staff concludes that the applicant adequately represented temporal and spatial variability within the upper unsaturated zone, and repository performance with respect to expected dose is not strongly affected by alternative representations.

Post-10,000-Year Approach

Consistent with 10 CFR 63.342(c)(2), DOE chose not to model climate or infiltration for the post-10,000-year period. Instead, DOE (i) selected four of the upper boundary condition net infiltration maps used to represent a climate state in the initial 10,000 years of performance; (ii) scaled these maps to create four new maps that would achieve areal-average deep percolation flux target values within the repository footprint consistent with the probability distribution in the proposed revision to 10 CFR 63.342 (NRC, 2005af) when using the infiltration-map probability values implemented for the initial 10,000 years; and (iii) created four site-scale unsaturated zone model flow fields on the basis of these infiltration boundary conditions (SAR Section 2.3.2.4.1.2.4.2).

DOE used the percolation distribution (SAR Section 2.3.2.4.1.2.4.2) in the proposed rule (NRC, 2005af) because the final rule was promulgated only a few months before DOE submitted the license application. Reflecting the difference between the draft and final percolation distributions, the mean percolation in the final rule [10 CFR 63.342(c)(2)] is 16 percent larger than that in the proposed rule. In DOE (2009cb, Enclosure 6), DOE performed sensitivity analyses that showed no significant effect on repository performance when using the distribution from the final rule instead of that from the proposed rule.

The NRC staff, based upon the evaluation in the previous paragraphs, concludes that the applicant acceptably considered the requirements of 10 CFR 63.342(c)(2) for the following reasons:

- DOE complied with the regulation by testing repository performance using a probability distribution of areal-average deep percolation fluxes to represent the effect of climate change that acceptably reflects the distribution in the regulation.

- The upper boundary condition net infiltration maps and deep percolation fluxes for the post-10,000-year period are based on FEPs considered in the initial 10,000 years of performance, consistent with 10 CFR 63.342.

- DOE represents infiltration uncertainty in the initial 10,000 years of performance and in the post-10,000-year period using a consistent approach.

The NRC staff evaluated the spatial distribution of net infiltration results for the post-10,000-year period by considering the potential effects of features and processes affected by climate change, such as alterations in soil depth, soil profile development, and changes in caliche volume. The NRC staff evaluated in SER Section 2.2.1.3.5.3.3 the influence of both temporal and spatial aspects of features and processes on the spatial distribution of net infiltration results and concluded that the effect of changes to properties over time can be evaluated in terms of alternative spatial distribution of net infiltration. The NRC staff finds, as described in the previous subsection, that DOE's performance assessment calculations would not be strongly affected by including systematic changes in seepage resulting from alternative representations of spatial variability, long-term climatic variability, and episodic fluctuations during the initial 10,000 years of performance. The NRC staff finds that these effects would be no more important in the post-10,000-year period because (i) these effects have a lesser effect on DOE's seepage model calculations as percolation fluxes increase, and the specified percolation fluxes in the post-10,000-year period are generally larger than those used during the initial 10,000 years and (ii) these effects have a lesser effect on seepage for degraded drifts, and DOE includes drift degradation from accumulated seismic events in performance assessment calculations for the post-10,000-year period.

Summary

The NRC staff reviewed the model conceptualization, the underlying assumptions of the ambient site-scale unsaturated zone flow model and other relevant abstractions with which the ambient site-scale unsaturated zone flow model exchanges data and information, and the alternative model conceptualizations DOE used to analyze model uncertainties. The NRC staff concludes that

- DOE appropriately considered the important flow-affecting features and processes and provided adequate technical bases for their inclusion in the abstracted ambient site-scale unsaturated zone flow model and downstream abstractions used in performance assessment calculations

- DOE adequately integrated intermediate-scale and fine-scale flow processes to represent the Upper Natural Barrier

- DOE adequately represented ambient mountain-scale percolation fluxes above the repository horizon and ambient drift-scale percolation fluxes at the repository horizon

- DOE adequately represented spatial and temporal variability of model parameters and boundary conditions to estimate flow paths and percolation flux with respect to the Upper Natural Barrier

- DOE provided sufficient data for ambient site-scale unsaturated zone model justification

- DOE adequately characterized and propagated ambient site-scale unsaturated zone model parameter uncertainty in process-level and performance assessment models with respect to the Upper Natural Barrier

- DOE adequately considered the requirements of 10 CFR 63.342(c)(2) regarding the use of the distribution of deep percolation rates to represent the effect of climate change during the period from 10,000 to 1 million years

2.2.1.3.6.3.3 Thermohydrologic Effects of Waste Emplacement

DOE represented the unsaturated zone and EBS using a hierarchy of far-field, near-field, and within-drift models to account for thermal effects due to emplacement. DOE used a conceptual and numerical model, the Multiscale Thermal-Hydrologic Model, to represent near-field and in-drift thermohydrologic conditions. DOE used input from the site-scale unsaturated zone flow model to account for far-field effects and output from the Multiscale Thermal-Hydrologic Model in downstream models that estimate the effects of in-drift thermohydrologic conditions on thermal seepage, quantity and chemistry of water contacting engineered barriers and waste forms, degradation of engineered barriers, and radionuclide release rates and solubility limits. This SER section describes the NRC staff's evaluation of the effects of waste emplacement on near-field and in-drift thermohydrologic conditions. The NRC staff evaluated repository performance related to the effects of thermal load on (i) seepage, (ii) quantity and chemistry of water contacting engineered barriers and waste forms, (iii) degradation of engineered barriers and radionuclide release rates, and (iv) solubility limits in SER Sections 2.2.1.3.6.3.4, 2.2.1.3.3, 2.2.1.3.1, and 2.2.1.3.4, respectively.

DOE passed Multiscale Thermal-Hydrologic Model output to several downstream models in its performance assessment calculations. DOE used drift-wall temperature to switch between two limiting conditions for seepage assuming that (i) no seepage occurs where drift-wall temperature is greater than 100 °C [212 °F] {the boiling temperature of water at the repository elevation is approximately 96 °C [205 °F]} and (ii) seepage occurs at the ambient rate for lower drift-wall temperatures (SAR Section 2.3.3.4). DOE modeled waste-package corrosion rates as depending on waste-package temperature and relative humidity and temperature-dependent chemistry of seeping water (SAR Section 2.3.6). DOE modeled a diffusive-release pathway forming within failed waste packages once a continuous liquid film forms at elevated waste-package relative humidity levels. DOE used invert saturation, invert temperature, and imbibition fluxes into the invert to estimate properties and fluxes affecting released radionuclide transport from the waste package to the host rock (SAR Section 2.3.7).

DOE's repository design basis places limits on the (i) peak waste package temperature {300 °C [572 °F] for 500 years, followed by 200 °C [392 °F] for 9,500 years} to reduce the potential for degradation of Alloy 22 waste packages; (ii) peak postclosure drift wall temperature {200 °C [392 °F]} to reduce thermal effects on drift stability; and (iii) peak mid-pillar temperature {96 °C [205 °F]} to facilitate drainage of percolation water between emplacement drifts (SAR Section 2.3.5.4.3; SAR Table 1.3.1-2). DOE's analysis indicated that the peak temperature limits can be accomplished through thermal loading criteria for waste packages (SAR Section 1.3.1.2.5). DOE used the Multiscale Thermal-Hydrologic Model to demonstrate that these design basis temperature limits can be achieved using (i) a stylized postclosure reference case based on expected waste package receipts over the emplacement period and (ii) two estimated limiting waste streams developed with different management options (SAR Section 1.3.1.2.5).

In evaluating repository performance with respect to near-field and in-drift thermohydrologic conditions, the NRC staff reviewed (i) the DOE conceptual model, (ii) the process-level implementation of the DOE conceptual model, (iii) data support and propagation of uncertainty, (iv) abstraction of the process-level model into performance assessment calculations, and (v) use of the model outputs in downstream models. The NRC staff's review of these topics focuses on aspects of repository performance that affect (i) duration of above-boiling temperatures at the drift wall, (ii) waste-package temperatures, (iii) in-drift humidity after onset of seepage, and (iv) seepage fluxes into inverts. These aspects of the Multiscale Thermal-Hydrologic Model are outputs that are used as input for downstream models that calculate seepage, corrosion of waste packages, and release pathways within the EBS. The NRC staff review also focuses on the DOE representation of collapsed drifts, because burial of waste packages by an insulating rubble layer could result in elevated waste-package temperatures.

Conceptual and Implemented Numerical Models

The NRC staff compared DOE's description of the conceptual model for thermohydrologic effects of waste emplacement provided in SAR Sections 2.3.3.1.1 and 2.3.5.4 (and selected references) with the NRC staff understanding of the Yucca Mountain natural system, obtained from extensive prelicensing experience and independent analyses of thermohydrologic processes at Yucca Mountain (e.g., NRC, 2005aa; Painter, et al., 2001aa; Manepally, et al., 2004aa). The NRC staff concludes that DOE adequately described its conceptual model of the thermohydrologic effects of waste emplacement because DOE adequately described (i) the features of the site geology and engineered barriers that affect thermohydrologic processes, and (ii) the most important thermohydrologic processes and parameters affecting the performance of the upper natural barrier, such as in-drift temperature and relative humidity, host rock drying and rewetting, thermal conduction, and thermal radiation, as discussed next. Furthermore, the NRC staff finds that DOE continued unchanged the features and processes, except for percolation flux as reviewed in SER Section 2.2.1.3.6.3.2, from the initial 10,000-year period through the period of geologic stability because the same model implementation was used for both periods.

DOE implemented the conceptual model into the four submodels of the Multiscale Thermal-Hydrologic Model. The submodels were linked through superposition, representing different aspects of coupled thermohydrologic processes at different spatial scales (SAR Section 2.3.5.4.1.3.1). These submodels consider (i) three-dimensional mountain-scale conduction, (ii) two-dimensional drift-scale thermohydrology in cross-sections (chimneys) perpendicular to the drift axis, (iii) links between the mountain-scale and chimney submodels, and (iv) effects of discrete waste packages. On the basis of the descriptions supplied in SAR Section 2.3.5.4 and SNL (2008aj), the NRC staff concludes that, with respect to each submodel, DOE (i) acceptably documented the procedures used to develop and support the submodel, (ii) developed the submodels of the Multiscale Thermal-Hydrologic Model in accordance with the corresponding conceptual model for the submodel, and (iii) used standard numerical approaches applied appropriately for the thermohydrologic processes considered in the submodel. The NRC staff concludes that while DOE did not transparently describe the overall Multiscale Thermal-Hydrologic Model methodology linking the submodels, it acceptably demonstrated that the linked submodels of the Multiscale Thermal-Hydrologic Model adequately represent thermohydrologic processes by using comparisons to alternative models. Further support for the NRC staff's conclusion on the acceptability of the completeness and representativeness of DOE's description of the technical basis for the Multiscale Thermal-Hydrologic Model, as shown in SAR Section 2.3.5.4 and SNL (2008aj, Section 6.2), was gained by comparing the applicant's technical basis with the NRC staff's understanding of numerical simulation approaches, obtained from extensive prelicensing experience and independent analyses of thermohydrologic processes at Yucca Mountain (e.g., NRC, 2005aa; Painter, et al., 2001aa; Manepally, et al., 2004aa).

DOE used its Drift Scale Test to validate the conceptual model underlying the drift-scale thermohydrologic submodel, as described in SAR Section 2.3.5.4.1.3.3 and SNL (2008aj, Section 7.4), and used its Large Block Test, shown in SNL (2008aj, Section 7.3), to demonstrate the Multiscale Thermal-Hydrologic Model's predictions of thermohydrologic processes in the host rock. DOE (i) simulated the thermohydrologic behavior observed in the Drift Scale Test and Large Block Test using the same modeling techniques included in the thermohydrologic submodel of the Multiscale Thermal-Hydrologic Model; (ii) compared modeled and observed temperature, relative humidity, and liquid-phase saturation values; and (iii) concluded that the differences were within the parametric uncertainty of Multiscale Thermal-Hydrologic Model results. On the basis of these considerations, the NRC staff concludes that DOE demonstrated that thermohydrologic processes observed in these heater tests are adequately included in the Multiscale Thermal-Hydrologic Model.

DOE identified several assumptions and limitations associated with linking the submodels (SAR Section 2.3.5.4.1.3.1), including restrictions related to mountain-scale and along-drift convection. DOE considered alternative conceptual models by comparing Multiscale Thermal-Hydrologic Model results with those of (i) an east-west cross section from a smeared-heat-source mountain-scale model; (ii) a three-dimensional, mountain-scale, nested-grid thermohydrologic model for a three-drift test case; and (iii) a three-dimensional, pillar-scale model with different axial in-drift vapor transport assumptions (SAR Section 2.3.5.4.1.3.3). DOE concluded that (i) the differences in predicted temperatures between the Multiscale Thermal-Hydrologic Model and the east–west model are within the range of temperature differences resulting from parametric uncertainty; (ii) the differences in predicted temperatures and relative humidity with the nested-grid model shows that the Multiscale Thermal-Hydrologic Model results envelope the nested-grid results; and (iii) the Multiscale Thermal-Hydrologic Model temperature, relative humidity, and saturation results are similar to those of the pillar-scale model. Based on assumptions related to lateral and axial vapor transport, the pillar-scale model predicts drier conditions than the Multiscale Thermal-Hydrologic Model. The NRC staff notes that wetter conditions in the Multiscale Thermal-Hydrologic Model results indicate a conservative approach because it results in earlier rewetting of the invert and the potential for more seepage locations. The NRC staff concludes that (i) DOE provided acceptable information to assess the model implementation uncertainty, because these comparisons range from mountain-scale to detailed pillar-scale simulations and (ii) DOE used a modeling approach that conservatively represents the effects of in-drift vapor transport on in-drift relative humidity, because in-drift axial transport would tend to systematically delay rewetting compared to the Multiscale Thermal-Hydrologic Model results. Therefore, the NRC staff concludes that DOE acceptably demonstrated that the Multiscale Thermal-Hydrologic Model results are comparable to alternative conceptual models in the intended thermal regime for repository operation. DOE assumed that mobilized water predominantly moves perpendicular to the axis of the emplacement drifts within the host rock. The NRC staff concludes this is consistent with the intended use of the Multiscale Thermal-Hydrologic Model under the design thermal regime for repository operation because mobilized water can drain within the mid-pillar region without systematic mountain-scale lateral redistribution of water when boiling conditions do not occur throughout the pillar.

In summary, the NRC staff concludes that DOE adequately provided the technical bases for the individual submodels and the linkages among the submodels supporting the intended use of the Multiscale Thermal-Hydrologic Model under the design thermal regime for repository operation, as discussed previously.

Abstraction of the Multiscale Thermal-Hydrologic Model for TSPA

The Multiscale Thermal-Hydrologic Model calculates time-dependent thermohydrologic variables for each of the 8 waste packages simulated for the 3,264 subdomains, each representing a 20-m [66-ft] segment of an emplacement drift. DOE referred to this as the comprehensive set of Multiscale Thermal-Hydrologic Model outputs. DOE performed the calculations for 7 parameter uncertainty cases, representing 12 combinations of infiltration uncertainty and thermal conductivity uncertainty, as described in the next subsection. DOE mapped each of the 3,264 subdomains to one of the five percolation bins abstracting the effects of seepage.

DOE calculated a comprehensive set of Multiscale Thermal-Hydrologic Model outputs for waste package temperature and relative humidity for different waste package types and drift-wall temperature for all Multiscale Thermal-Hydrologic Model subdomains in each percolation bin. The calculated outputs were provided as input to the Waste Package Degradation Model Component, the Drift Seepage Submodel, and the Drift Wall Condensation Submodel within the TSPA model, as shown in SNL (2008ag, Section 6.3.2.2). These downstream models use Multiscale Thermal-Hydrologic Model outputs to calculate waste package failure rates. The NRC staff concludes that DOE acceptably abstracts Multiscale Thermal-Hydrologic Model output for these downstream models because the entire set of process-level output is provided to these downstream models.

DOE abstracted the Multiscale Thermal-Hydrologic Model results by selecting a single representative codisposal waste package and commercial spent nuclear fuel waste package for each percolation bin. DOE used the codisposal and commercial spent nuclear fuel waste packages with peak waste package temperature and drift-wall boiling duration closest to the median values to represent thermohydrologic conditions for all waste packages in the percolation bin. DOE calculated the time-dependent output values for waste package surface temperature and relative humidity, drift wall temperature, invert temperature, invert saturation, and flux into the invert for the selected representative waste packages SNL (2008ag, Section 6.3.2.2). These values were provided to the Waste Form Degradation and Mobilization Model Component, the Engineered Barrier System Flow and Transport Model Component, the Engineered Barrier System Chemical Environment Submodel, and the Drift Wall Condensation Submodel within the TSPA model. These downstream models use the Multiscale Thermal-Hydrologic Model outputs to calculate radionuclide release rates from failed waste packages and radionuclide transport within the EBS. DOE provided analyses in SNL (2008ag, Section 7.3.4.3.1) comparing estimates of cumulative radionuclide releases from a single failed waste package using the representative location with estimates calculated using the comprehensive set of Multiscale Thermal-Hydrologic Model output. These analyses demonstrated essentially identical cumulative release of representative radionuclides from the EBS after both 10,000 and 1 million years. The NRC staff concludes that DOE acceptably abstracted Multiscale Thermal-Hydrologic Model process-level results in performance assessment calculations with respect to these downstream models because DOE demonstrated that the abstraction used for performance assessment calculations does not lead to an underestimate of radionuclide releases from the EBS for both the initial 10,000-year and million-year periods.

Data Support and Uncertainty Propagation in the Multiscale Thermal-Hydrologic Model

The NRC staff reviewed the information provided in SAR Section 2.3.5.4.1.2 and selected references therein to evaluate DOE's supporting data and characterization of uncertainties in the Multiscale Thermal-Hydrologic Model. Because of the importance of the following three topics in performance assessment calculations, the NRC staff focuses its review on (i) input parameters that significantly affect Multiscale Thermal-Hydrologic Model results, (ii) data support for the range of uncertainty in input parameters, and (iii) uncertainty propagation in the Multiscale Thermal-Hydrologic Model.

Multiscale Thermal-Hydrologic Model simulations require input to describe the EBS and natural barriers (SAR Section 2.3.5.4.1.2.2). DOE identified the design control parameters and associated design constraints in SAR Table 2.2-3. DOE derived EBS parameters from the design information. DOE based natural barrier parameters on the site-scale unsaturated zone flow model (evaluated in SER Section 2.2.1.3.6.3.3), including hydrologic properties of the unsaturated zone, natural system geometry, and percolation fluxes below the PTn.

DOE explicitly represented the drift, drip shield, and invert components of the EBS [SNL, 2008aj, Section 4(a)]. DOE screened out thermohydrologic processes related to other engineered components, such as rock bolts and associated boreholes used for ground support, on the basis of low consequence for performance assessment calculations. The NRC staff evaluated the applicant's screening arguments with respect to FEP 1.1.01.01.0B (Influx through Holes Drilled in Drift Wall or Crown) and 2.1.06.04.0A (Flow Through Rock Reinforcement Materials in Engineered Barrier System) in SER Section 2.2.1.2.1.1, finding both screening arguments acceptable, based on the small effect on performance assessment calculations of these FEPs. DOE described the design information representing engineered features in the Multiscale Thermal-Hydrologic Model as consistent with the design of subsurface structures, systems, and components (SAR Section 2.3.5.4.1.2.1). In SNL (2008aj, Section 4.1), DOE described repository subsurface and waste package design information as obtained from controlled sources and based on the repository design.

The NRC staff concludes that DOE adequately represented the input parameters describing engineered components with respect to the thermohydrologic parameters the Multiscale Thermal-Hydrologic Model calculated because DOE (i) identified the EBS components predominantly influencing in-drift thermohydrologic conditions, (ii) included these components in the Multiscale Thermal-Hydrologic Model, (iii) acceptably screened out other components, and (iv) described the included EBS components using parameter values consistent with the repository design.

DOE identified host rock thermal conductivity and percolation flux as the dominant parameters responsible for variability and uncertainty in simulated thermohydrologic conditions (SAR Section 2.3.5.4.1.3.2). The NRC staff evaluated the adequacy of percolation flux in SER Section 2.2.1.3.6.3.2; incorporation of uncertainty in percolation is described in the paragraphs that follow. DOE considered the uncertainty in thermal conductivity of the host rock using a geostatistical model supported by laboratory measurements and core samples to constrain and condition the geostatistical model (BSC, 2004bf). DOE extracted host rock thermal conductivity values from the geostatistical model, evaluated the influence of thermal conductivity on peak waste package temperatures and duration above boiling, and assigned weights for implementation in TSPA, as outlined in SNL [2008aj, Section 6.2.13.3(a)]. DOE averaged the thermal properties of nonrepository units to facilitate computational efficiency, using a sensitivity analysis to demonstrate that the averaging does not affect the Multiscale Thermal-Hydrologic Model results, as shown in SNL [2008aj, Section 6.2.13.4(a)]. The percolation flux applied at the top boundary in the thermohydrologic submodel is the percolation flux at the base of the PTn unit calculated in the site-scale unsaturated zone flow model for the nominal 10th, 30th, 50th, and 90th percentile scenarios. The NRC staff notes that the uncertainty in percolation flux in the site-scale unsaturated zone flow model at the mountain scale is propagated consistently in the Multiscale Thermal-Hydrologic Model. The NRC staff concludes that DOE acceptably propagated the uncertainty in host rock thermohydrologic properties with respect to observations into Multiscale Thermal-Hydrologic Model input because DOE appropriately identified the dominant thermohydrologic property affecting uncertainty and used appropriate methods to propagate the uncertainty from the available laboratory and field data into the Multiscale Thermal-Hydrologic Model input.

DOE propagated uncertainty in host rock thermal conductivity and percolation flux into simulations using 7 of the 12 combinations of 10th, 30th, 50th, and 90th percentile flux scenarios with low, mean, and high thermal conductivities as input (SAR Section 2.3.5.4.1.3.2). For each of the remaining five combinations, DOE used the results from one of the seven simulations as a surrogate on the basis of similarity in boiling duration, as described in SAR Section 2.3.5.4.1.3.2 and SNL [2008aj, Section 6.2.12.3(a)]. DOE propagated the full set of 12 combinations into performance assessment calculations. The NRC staff concludes that DOE acceptably propagated uncertainty in thermohydrologic conditions using the key performance-affecting result (duration of drift-wall boiling) to select surrogate simulations and consider the full range of the key thermohydrologic parameters.

In summary, the NRC staff concludes that DOE acceptably propagated uncertainty in thermohydrologic parameters into performance assessment calculations because

- DOE identified the key performance-affecting thermohydrologic parameters and factors

- DOE adequately propagated uncertainty from observations into Multiscale Thermal-Hydrologic Model input

- DOE represented uncertainty in the key performance-affecting parameters using appropriate combinations of the uncertain parameters

Use of Multiscale Thermal-Hydrologic Model Results in Downstream Models

The NRC staff evaluated predictions of thermohydrologic conditions in part by considering how DOE used Multiscale Thermal-Hydrologic Model results in downstream models. The downstream models using Multiscale Thermal-Hydrologic Model results include (i) thermal seepage, (ii) quantity and chemistry of water contacting engineered barriers and waste forms, (iii) degradation of engineered barriers, and (iv) radionuclide release rates and solubility limits. The NRC staff evaluates these downstream models in SER Sections 2.2.1.3.6.3.4, 2.2.1.3.3, 2.2.1.3.1, and 2.2.1.3.4, respectively.

On the basis of the NRC staff's review of the downstream uses of abstracted results produced by the Multiscale Thermal-Hydrologic Model, the NRC staff concludes that DOE acceptably used those results and parameters in downstream models because they are generally used in a manner consistent with Multiscale Thermal-Hydrologic Model assumptions. However, DOE chose to not represent the full range of local variability in Multiscale Thermal-Hydrologic Model results in some downstream model calculations. But on the basis of DOE sensitivity analyses [e.g., SNL (2008aj, Section 6.3)], the NRC staff concludes that DOE acceptably represented local variability in Multiscale Thermal-Hydrologic Model results (e.g., variability in drift wall temperature, waste package temperature and relative humidity, imbibition fluxes) because DOE demonstrated that local variability on radionuclide releases minimally affects duration or magnitude of radionuclide releases.

Uncertainty in Thermal Loading

DOE described the loading strategy implemented in the Multiscale Thermal-Hydrologic Model as a stylized postclosure reference case based on expected waste package receipts over the emplacement period (SAR Section 1.3.1.2.5). DOE considered the actual future waste stream to be uncertain and considered flexibility in emplacement strategies necessary to manage acceptance of a wide spectrum of waste streams (SAR Section 1.3.1.2.5). For example, the design heat load DOE described in SNL (2008ai) updated the heat load used for Multiscale Thermal-Hydrologic Model calculations (SAR Section 2.3.5.4.1). In SAR Section 2.3.5.4.3, DOE described how temperature estimates using the design heat load compared with the temperature estimates from the Multiscale Thermal-Hydrologic Model results. DOE stated that it would develop an emplacement drift plan (SAR Section 1.3.1.2.5; DOE, 2009ct, Enclosure 1) for each drift, or set of drifts, that will (i) provide specific information, such as waste characteristics, waste package emplacement locations, and ventilation duration and (ii) describe how preclosure and postclosure performance requirements will be met using the selected emplacement strategy.

DOE demonstrated that thermal management approaches using temperature index functions representing three- and seven-package segments are capable of achieving performance targets for mid-pillar, drift-wall, and waste-package peak temperatures for the two estimated limiting waste streams (SAR Section 2.3.5.4.3). The thermal management approaches utilize extended duration of ventilation beyond 50 years or different surface handling facilities and aging capacities. DOE concluded that (i) only minor modifications to the TSPA model inputs are needed to represent the anticipated range of thermal loading; (ii) the geomechanical, hydrogeologic, and geochemical system responses for the two estimated limiting waste streams are each within the range of applicability for the respective models, as shown in

SNL (2008ai, Section 6.4); and (iii) the changes in system responses arising from future waste streams different than the reference case do not significantly affect the screening justifications for excluded FEPs or the modeling basis for included FEPs, as outlined in SNL (2008ai, Section 6.5). The NRC staff concludes that DOE acceptably demonstrated that practical thermal management strategies can be implemented to achieve postclosure design basis targets in intact drifts using surface aging (limited to the 50-year emplacement period) and flexibility in ventilation duration beyond the minimum length of 50 years.

DOE expects that, under bounding assumptions, peak waste package temperatures for some waste packages may exceed the design basis temperature by nearly 100 °C [180 °F] if drifts were to collapse within the first 90 years after closure (SAR Section 2.3.5.4.3, SAR Figure 2.3.5-37). DOE considered several mechanisms for drift collapse, including seismic-induced ground motion, thermally induced stresses, and gravitational stresses. DOE screened out all mechanisms for drift collapse other than seismic-induced ground motion (FEP 2.1.07.02.0A, Drift Collapse) on the basis of low consequence (SNL, 2008ab). The NRC staff, in SER Section 2.2.1.2.1.3.2 (FEP 2.1.07.02.0A), concludes that DOE provided an adequate basis for these screening decisions. DOE screened in mechanisms of drift degradation from seismic-induced ground motion, including in-drift temperature and relative humidity consequences from seismic-induced drift collapse. DOE performed a bounding probabilistic risk analysis considering uncertainty in seismic events and the key thermohydrologic parameter, thermal conductivity. This risk analysis used methods and assumptions consistent with DOE's performance assessment calculations to calculate a probability of approximately 1 in 10,000 that the hottest waste package in the stylized postclosure reference case exceeded the 300 °C [572 °F] waste package temperature design basis because of drift collapse during the initial 10,000 years, as described in SAR Section 2.3.5.4.3, SNL [2008aj, Section 6.3.17(a)], and SNL (2008ai, Section 6.4.2.5). DOE screened out consideration of peak waste package temperatures exceeding the established temperature limits in performance assessment calculations on the basis of the probabilistic calculation and additional information describing the nature of the bounding assumptions (SAR Section 2.3.5.4.3). The NRC staff concludes that DOE adequately justified screening out the risk from EBS temperatures exceeding the established temperature limits as a result of drift collapse because DOE (i) used appropriate methods consistent with performance assessment calculations to develop risk estimates and (ii) applied a screening criterion for the probability of occurrence consistent with 10 CFR 63.342(a).

The NRC staff recognizes that the actual waste sent to Yucca Mountain, flexible emplacement strategies, and natural system and modeling factors may change the ensemble thermal response at the repository scale or the local thermal response at a small scale from those predicted in the SAR, which used a stylized reference case [e.g., SNL (2008ai, Figure 6.4.2-28)] for performance assessment and screening calculations. DOE stated that it would develop an emplacement drift plan prior to waste emplacement that specified for each drift, or set of drifts, the (i) waste characteristics, (ii) waste package emplacement locations, (iii) ventilation duration, and (iv) how preclosure and postclosure performance requirements will be met using the selected emplacement strategy, as described in SAR Section 1.3.1.2.5 and DOE (2009ct, Enclosure 1). Therefore, the NRC staff finds the uncertainty in heat load is adequately addressed. DOE stated that the emplacement drift plan would be updated prior to waste emplacement, and the performance requirements would be re-assessed. The NRC staff notes that its evaluation of the thermal response would be one element of an NRC review of an application to receive and possess of waste.

Summary

In this SER section, the NRC staff evaluated the in-drift and near-field thermohydrologic conditions estimated using the Multiscale Thermal-Hydrologic Model (SAR Section 2.3.5.4.1). The NRC staff concludes that, with respect to the Multiscale Thermal-Hydrologic Model, DOE adequately described the (i) conceptual models, (ii) implementation, (iii) performance assessment abstraction, (iv) data used to derive inputs, and (v) uses of output in performance assessment calculations. The NRC staff concludes that DOE adequately considered (i) conceptual model uncertainty, (ii) alternative conceptual models, and (iii) data variability and uncertainty, with respect to the application of the Multiscale Thermal-Hydrologic Model for performance assessment calculations. On the basis of the discussion set forth previously, the NRC staff concludes that DOE adequately integrated the Multiscale Thermal-Hydrologic Model with downstream models in performance assessment calculations. The NRC staff concludes that DOE adequately demonstrated the capability for devising emplacement loading strategies that achieve postclosure design basis targets.

2.2.1.3.6.3.4 Ambient and Thermal Seepage Models

This section contains the NRC staff's review of DOE's model and results for water seeping into drifts. Seepage into drifts encompasses a subset of processes in the unsaturated flow system that occurs in the vicinity of the drift wall. DOE described seepage as a component of the unsaturated zone above the repository and within the Upper Natural Barrier (SAR Section 2.1.2.1). DOE considered seepage (SAR Section 2.3.3) separately from unsaturated zone flow (SAR Section 2.3.2) because of the smaller scale of analysis needed for the processes important for seepage and, consequently, the need for a different set of data and models to produce results for use in the performance assessment.

DOE strictly defined seepage as liquid water that drips from the drift ceiling and therefore could potentially contact engineered barrier components. Two primary processes provide barrier capability in DOE's seepage model: capillary diversion of liquid water around large openings (drifts in this case) and vaporization in the host rock that creates a dry zone around the drifts (SAR Section 2.1.2.1). Capillary forces may make drifts barriers to flow by inducing water to laterally flow (divert) around the large opening. During the thermal period, the vaporization barrier refers to the boiling of water in the host rock and migration of the resultant vapor to locations away from the heat source. In DOE's abstraction, the resultant creation of a dryout zone surrounding a drift leads to elimination of liquid flux at the drift wall.

Three inputs are provided to the seepage abstraction from other areas of the natural systems. First, the Multiscale Thermal-Hydrologic Model provides the distribution of percolation rates across the repository to the seepage abstraction (SAR Section 2.3.5.4.1). The values used are consistent with those from the ambient, site-scale unsaturated zone model (SAR Section 2.3.2). Second, the Multiscale Thermal-Hydrologic Model provides the temperature history for the drift wall to the seepage abstraction (SAR Section 2.3.5.4.1). Third, the thermal-mechanical model abstraction (SAR Section 2.3.4) provides the accumulated amount of rubble to the seepage abstraction, which uses it to reflect the degradation state of drift openings for seepage calculations.

DOE's seepage abstraction provides two outputs to the EBS models: the seepage rate and the fraction of drift segments where liquid water seeps into drifts. The total dripping flux (SAR Section 2.3.3), which is the sum of the seepage and condensation flux (SAR Section 2.3.5.4.2), is the flux of liquid water leaving the drift wall and contacting engineered components. The NRC staff reviewed DOE's estimates of condensation flux in SER Section 2.2.1.3.6.3.5.

In its review of DOE's seepage estimates, the NRC staff considered how and to what extent seepage affects performance. The fraction of waste package locations getting wet and seepage flux in those wet areas are both passed directly from DOE's seepage abstraction model to the EBS models. The condition of the drip shields (intact or degraded) plays a critical role in the DOE performance assessment. When intact, drip shields prevent liquid water from contacting the waste package. In this case, seepage has no influence on the release of radionuclides, and transport rates out of failed waste packages are constrained to the slower diffusive rate of radionuclide movement rather than the faster advective rate. When degraded, drip shields do not divert all water away from waste packages. In this case, the EBS models use seepage estimates, which may influence the (i) corrosion of engineered components; (ii) number of waste packages contacted by water; and (iii) dissolution, mobilization, and transport of radionuclides to the unsaturated zone below the drifts. The NRC staff reviews these three areas, which include processes and features from the drip shield to the invert/host rock interface, in SER Sections 2.2.1.3.1, 2.2.1.3.3, and 2.2.1.3.4.

The NRC staff organized its review of the information and bases DOE provided for ambient and thermal seepage into the five following areas: (i) development of the ambient seepage abstraction, (ii) capillary diversion for intact drifts, (iii) capillary diversion for degraded drifts, (iv) seepage fraction (the fraction of repository that is in the seeping environment), (v) spatial variability of flow, and (vi) thermal seepage.

Development of Ambient Seepage

This section reviews DOE's description of seepage processes, field tests, and measured data and how they are incorporated into the seepage abstraction.

DOE stated that capillary diversion of liquid water around large openings is the dominant seepage process providing barrier capability during the ambient period (SAR Section 2.1.2.1). In the context of seepage, DOE defined the thermal period as the time when the drift wall temperature exceeds 100 °C [212 °F]. Therefore, DOE used the ambient seepage model to estimate water flux entering the drift from approximately the first few thousand years through 1 million years (SAR Section 2.3.3.1).

For ambient seepage, DOE described (i) the theoretical treatment of seepage into circular openings in porous media; (ii) its choice of a fracture-only, stochastic, equivalent continuum seepage model; and (iii) field tests at Yucca Mountain used to calibrate seepage models (SAR Section 2.3.3.1). Because drifts are approximately circular in cross section, DOE drew on the theoretical treatment derived from an analysis in Philip, et al. (1989aa) of water diversion around large circular openings in homogeneous porous media. Because water is more likely to drip from fractures than from matrix, and to simplify the numerical models, DOE developed seepage models that include only the fracture network as the porous media. DOE described field tests at Yucca Mountain and observations from analog sites that illustrated the capability of circular openings in fractured rock to divert water.

DOE implemented a seepage approach predicated on continuum models on the basis of the Richards equation (Richards, 1931aa) for granular porous media and the representative element volume assumption. Flow in fractures and through the fracture networks, however, may not satisfy these assumptions. For example, the density of fractures with flowing water is small relative to the grid size of the seepage model. Also, in addition to capillary-based flow in small-aperture fractures, flow regimes in fractures also include adsorbed films, sliding drops, rivulet flow, stable thick films, and unstable (laminar or turbulent) films (Ghezzehei, 2004aa); none of these would follow classical Richards-equation-based models. Instead of directly modeling these complexities, DOE's approach calibrated the seepage model to field injection tests that would inherently incorporate small-scale processes and flow regimes in fractures. DOE's models solve the Richards equation for saturated-unsaturated flow through porous materials, with the van Genuchten–Mualem relations describing the capillary pressure and relative liquid permeability in the fracture continuum as a function of liquid saturation (van Genuchten, 1980aa). Because DOE used *in situ* field tests to measure seepage, and field tests are a well-recognized method of capturing ensemble behavior of processes, the NRC staff finds that DOE used an acceptable approach capable of capturing unsaturated flow processes in fractures in the seepage model.

DOE used two separate numerical seepage models: one for calibration to field tests and the other to generate ambient seepage abstraction lookup tables for the performance assessment model. Injection tests at Yucca Mountain, as described in BSC (2004av, Section 6.2), form the basis of DOE's calibrated seepage model that was designed specifically for the injection test domains (SAR Section 2.3.3.2.3.3). The key parameter for estimating seepage is the unsaturated zone property of capillary strength. It is the inverse of the van Genuchten α term (van Genuchten, 1980aa) and reflects the ability of the fracture continuum to offset gravity for water dripping into drifts. DOE conceptually separated percolating water reaching the drift ceiling into (i) water diverted by capillarity, which remains in the host rock; (ii) water dripping from the ceiling, which is defined as the seepage flux; and (iii) water entering the drift but not dripping, which includes along-wall flow and evaporation. Because the capillary strength parameter is calibrated from the injection tests, it encompasses the effects of evaporation, along-wall flow, and capillary diversion.

The seepage abstraction was developed using the second seepage model, which kept the same grid characteristics as the seepage calibration model, but was designed for the geometry of emplacement drifts (SAR Section 2.3.3.2.3.4). DOE used the second seepage model to generate two tables: one for intact drifts and one for collapsed drifts for the seismic modeling cases. These tables covered a wide range of percolation rates and permeability and capillary strength parameter values. To estimate seepage at any location using the abstraction, DOE sampled capillary strength and permeability from uncertainty distributions (SAR Section 2.3.3.2.4.1) and used a spatially dependent local percolation rate.

DOE adjusted the local percolation flux input for the seepage lookup table by a flow-focusing factor (SAR Section 2.3.5.4.1). Flow-focusing factors increase local percolation in some areas and decrease it in other areas, but the flow-focusing factor does not modify the total flux over the entire area. DOE used the flow-focusing factor to incorporate intermediate-scale heterogeneity (e.g., nonvertical small faults) that might lead to convergence or divergence of flow in the rock layers immediately above drifts. Spatially variable net infiltration and other large-scale heterogeneities from the ambient site-scale unsaturated flow model were propagated to the Multiscale Thermal-Hydrologic Model, and therefore were brought into the seepage abstraction. DOE described intermediate scale as falling between the grid scale of the ambient site-scale unsaturated flow model {approximately on the order of 100 m [330 ft]} and several drift diameters. For spatial variability below the scale of several drift diameters, DOE incorporated heterogeneity directly into the seepage numerical model input.

For the seismic ground motion, seismic fault displacement, and igneous intrusion modeling cases, DOE predicted changes to the drift opening that might lead to changes in seepage rate and distribution. To account for changes in the drift wall caused by seismic events that lead to changes in dripping, DOE utilized the second seepage table for collapsed drifts. The degree of drift degradation controls the switch from the intact to the collapsed seepage table. For lithophysal rocks only, values from both tables are obtained and some intermediate value is calculated on the basis of scaling to the volume of rubble detached from the drift ceiling. For nonlithophysal rocks, accumulated rockfall above a specified threshold causes the seepage to be set equal to the percolation rate. The two seismic modeling cases are treated slightly differently. For the seismic ground motion modeling case, all drifts are shifted to a degraded state. For the seismic fault displacement, only a small number of drifts and waste package sections are affected by the seismic event. As with the seismic scenario for nonlithophysal rocks, the DOE seepage abstraction for the igneous scenario is simplified by neglecting the effect of capillary diversion.

The NRC staff compared DOE's description of ambient seepage processes and the incorporation of those features and processes into models and abstractions, with the NRC staff's understanding of seepage-related features and processes at Yucca Mountain obtained from field observations and independent analyses (NRC, 2005aa; Leslie, et al., 2007aa; Basagaoglu, et al., 2007aa; Or, et al., 2005aa). On the basis of this understanding, the NRC staff concludes that DOE adequately described features and processes at Yucca Mountain and that DOE included the important features and processes in its conceptual and numerical models for ambient seepage for the nominal, seismic, and igneous intrusion scenarios.

Capillary Diversion around Intact Drifts

DOE described the effectiveness of the unsaturated zone in the Upper Natural Barrier in terms of two important metrics that affect performance: seepage flux and seepage fraction (SAR Section 2.1.2.1.6). This subsection focuses on the seepage flux, which is calculated as the amount of water percolating through the host rock above the drift that the capillary diversion process does not divert around the drift by capillary diversion process. Capillary strength is the key parameter in DOE's seepage model for estimating the amount of water diverted around drifts, both because of the sensitivity of seepage estimates to this input parameter and because of the uncertainty in estimating representative values of this parameter. Percolating water (i) drips from the drift ceiling (seepage); (ii) flows laterally around the drift in the host rock; (iii) enters the drift, but flows along the drift wall; or (iv) enters the drift in the gas phase (vapor flux). DOE defined seepage as only the water dripping from the drift ceiling.

The NRC staff reviewed the information provided in the SAR to evaluate the adequacy of DOE's estimate of seepage during the ambient period considering data and model support. The NRC staff relied on integrity and performance of engineered components for conclusions after evaluating the information DOE provided for the capability of capillary diversion to reduce the flux of water entering drifts. This review approach was driven by the NRC staff's difficulty in assessing the effect of uncertainties in DOE's model due to (i) alternative interpretations of injection tests used to calibrate the seepage model, as described in DOE (2009ct, Enclosure 4) and Or, et al. (2005aa); (ii) representativeness of the locations of injection tests for estimating distribution of the capillary strength parameter, as described in DOE (2009ct, Enclosures 2 and 3); (iii) lack of literature support for the capillary strength parameter because of grid dependency and nonstandard inclusion of other processes in this term; (iv) an alternative conceptual model for water entering drifts suggested by observations in the East-West Cross Drift Passive Test, as shown by Salve and Kneafsey (2005aa) and DOE (2009bo, Enclosure 1);

and (v) lack of quantitative support from analog sites and problems and inconsistencies from other analog sites [e.g., lithophysae as outlined in DOE (2009ct, Enclosure 6) and ancient artifact sites].

For the initial 10,000 years after disposal, the NRC staff concludes that underestimates of seepage are not important to 10,000-year performance. This conclusion is based on the NRC staff's evaluations in SER Sections 2.2.1.3.1.3.1 and 2.2.1.3.2.6, which conclude that corrosion and mechanical processes do not degrade drip shields during the first 12,000 years after repository closure. Intact drip shields divert all seepage from contacting waste packages. The NRC staff evaluates seepage rates for the million-year period in its review of the seepage for degraded drifts in the next subsection.

Capillary Diversion for Degraded Drifts

NRC staff reviewed DOE's approach and estimate of seepage flux that account for the disruptive modeling cases of seismic ground motion, seismic fault displacement, and igneous intrusion. In DOE's model, capillary diversion remains the predominant barrier for seismically degraded drifts. This evaluation focuses on seepage in lithophysal units reflecting seismically degraded drifts.

To address drift collapse, DOE developed a collapsed drift seepage table similar to the intact drift seepage lookup table. The table was developed using an enlarged drift opening, with an 11-m [36-ft] instead of 5.5-m [18-ft] diameter. DOE selected a perfectly circular 11-m [36-ft]-diameter drift opening, as outlined in BSC (2004be, Section 6.6.3), based on inspection of simulation results from rock mechanics modeling, as described in BSC (2004al, Appendix R). NRC staff notes that DOE's calculated increase in seepage for the collapsed drift seepage table is solely due to the larger opening and does not reflect irregular drift shapes, as illustrated in BSC (2004al, Appendix R). The NRC staff notes that both the size and shape of a drift affect capillary diversion. The irregular shape of degraded drifts may lead to increased values of seepage. As explained in the following paragraphs, however, the NRC staff found that uncertainty of seepage for collapsed drifts, such as that due to irregular shapes, is not significant to performance.

A tiered abstraction was used to account for the degree of drift degradation (SAR Section 2.3.3.4.1.1). For nonlithophysal rock, seepage estimates from the intact seepage table were used with estimated accumulated rubble less than 0.5 m^3 per meter of drift length [5.4 ft^3 per foot of drift length]. Otherwise, seepage was set to the percolation rate, including adjustments from the sampled focusing factor. For lithophysal rock, the intact seepage table was used for accumulated rubble less than 5 m^3 per meter of drift length [54 ft^3 per foot of drift length]. For accumulated rubble greater than 60 m^3 per meter of drift length [650 ft^3 per foot of drift length], the collapsed drift seepage table was used. Seepage was interpolated from the entries in both the intact and collapsed drift seepage tables for intermediate values of rubble accumulation {between 5 and 60 m^3 per meter of drift length [54 and 650 ft^3 per foot of drift length]} in lithophysal rock.

The NRC staff concludes that DOE estimates of seepage for seismic ground motion, seismic fault displacement, and igneous intrusion modeling cases during the initial 10,000 years are acceptable for performance assessment calculations because of the low probability of occurrence for disruptive events. For the seismic fault displacement modeling case, only a small number of waste packages can be affected by each event, as outlined in SNL (2008ag, Section 6.1.2.3.4). Therefore, uncertainty in seepage rates has little effect on total estimated dose. For the igneous intrusion modeling case and for seismically degraded drifts in nonlithophysal rock, seepage is not underestimated, because DOE conservatively sets seepage rate equal to percolation rate.

For the million-year period, DOE's model determined disruption by seismic ground motion to be the most important contributor to dose estimates. To evaluate DOE's estimates of seepage rate for the million-year period, the NRC staff considered the following:

- DOE calculated an average seepage rate for the million-year period to be 49 percent of percolation rate, shown in DOE (2010ai, Table 8). On average, seepage is unlikely to be greater than percolation (i.e., the NRC staff knows of no process that focuses water to drifts on an average basis for the repository). Hence, doubling the seepage rate so that seepage is equal to percolation is a maximum bound. Information from sensitivity analyses in DOE (2009cx, Enclosure 1) illustrated that increases in seepage flux by factors slightly larger than two do not significantly affect estimates of total dose.

- The NRC staff expects that degraded drifts provide some measure of barrier capability for seepage, though the value of the percentage of percolation is uncertain. This expectation is based on the NRC staff's understanding of the physics of flow at sharp boundaries and modeling efforts completed during the prelicensing period (Basagaoglu, et al., 2007aa; Leslie, et al., 2007aa; Or, et al., 2005aa). The NRC staff models generate seepage rates approximately the same as the DOE estimates for the million-year period.

Therefore, the NRC staff concludes that DOE's treatment of capillary diversion and estimates of seepage for the million-year period are acceptable for performance assessment.

Seepage Fraction

DOE described the effectiveness of the unsaturated zone in the Upper Natural Barrier in terms of two metrics: seepage flux and seepage fraction (SAR Section 2.1.2.1.6). This subsection focuses on the latter. DOE defined seepage fraction as the number of drift segments where seepage occurs, divided by the total number of drift segments (SAR Section 2.3.3.1). DOE defined a drift segment as an approximation to the average waste package length, including the gap between waste packages. Therefore, seepage fraction is essentially the same as the fraction of waste packages getting wet. The NRC staff's evaluation of DOE's estimates of seepage fraction focuses on the adequacy of average values used for the million-year period for realizations in which the seismic ground motion and igneous intrusion modeling cases dominate estimates of dose (see SER Section 2.2.1.4.1.3.3.1.1.2). Estimates of seepage fraction for the initial 10,000 years are not significant because NRC concluded that the drip shields are expected to remain intact for this entire period (SER Sections 2.2.1.3.1.3.1 and 2.2.1.3.2.6) and will divert all seeping water away from waste packages.

Conceptually, DOE defined seepage fraction as the portion of the repository where dripping is expected to occur, using the footprint of a waste package (drift diameter and waste package length) to define a seepage area. Therefore, seepage fraction is linked to the number of waste packages that would get wet if no drip shields were present, divided by the total number of waste packages. The remainder of the repository is the nonseeping environment, where the flux of liquid water potentially dripping in a waste package location is set to zero in the DOE abstraction.

In the DOE abstraction, seepage fraction is important because releases of radionuclides in the seeping environment are transported by advection. In that portion of the repository where the liquid flux is zero, any released radionuclides are transported by diffusive processes out of the waste package, which are slow compared to advective transport rates. Releases in the nonseeping environment rely on transport by diffusion along stagnant water films. Therefore, determination of the threshold at which seepage occurs can impact radionuclide transport.

DOE used the seepage model to predict seepage at all locations. At locations where calculated seepage was less than 0.1 kg/yr per waste package (DOE, 2009ct, Enclosure 5), DOE set the value to zero in the performance assessment. Because the seepage fraction is sensitive to the selection of a value for the seepage threshold, DOE (2009ct, Enclosure 5) performed sensitivity analyses showing that reducing the seepage threshold value to zero led to a negligible change in predicted performance.

In the abstraction, seepage fraction is fixed at a constant value for any particular TSPA realization. To determine the constant value, DOE selected the highest calculated seepage fraction that would occur at any time during the simulation period (excluding the effect of igneous events). This value of seepage fraction was then applied throughout the simulation. Separate TSPA realizations were run for the 10,000-year (using a 20,000-year simulation period) and million-year calculations. DOE provided average values for TSPA realizations in SAR Tables 2.1-6 through 2.1-9. DOE estimated an average seepage fraction of 0.10 for the initial 10,000 years when seismic and igneous scenarios do not influence seepage. Similarly, DOE estimated an average seepage fraction of 0.69 for the million-year period when the seismic ground modeling case is the dominant dose contributor. Igneous intrusion and seismic fault displacement scenarios do not influence the seepage fraction used in the million-year calculation because (i) the abstraction for igneous intrusion sets seepage fraction to one after the occurrence of an igneous event, according to SAR Section 2.3.11.6.5 and DOE (2009ct, Enclosure 8), although the probability for an igneous event is low for any particular realization and (ii) seismic fault displacement only affects a small number of drifts.

The NRC staff concludes that seepage fractions are not underestimated on the basis of the following

- For the 10,000-year period, DOE adequately demonstrated that drip shields are estimated to remain intact significantly beyond the initial 10,000 years (SER Sections 2.2.1.3.1.3.1 and 2.2.1.3.2.6). Therefore, no liquid is predicted to reach the waste packages because drip shields divert all water regardless of whether the drift segment is a seeping or nonseeping environment. Therefore, the average value, and any uncertainty, in the value of seepage fraction are not important for performance during the initial 10,000 years.

- For the million-year period, the NRC staff notes that the calculated seepage fractions cannot increase much before reaching the bounding value of one (i.e., all waste packages get wet). Increasing the seepage fraction from 0.69 to the maximum of 1 at most results in a 44 percent increase in dose, assuming all aspects of release and travel paths are the same for the additional area compared to the original area. This increase in dose is not significant to performance results as total dose is not close to the criteria specified in 10 CFR Part 63; the NRC staff found in SER Section 2.2.1.4.1 that estimated dose is significantly less than the standards in 10 CFR Part 63.

The NRC staff expects that seepage fraction during the million-year period will likely be less than the bounding case of one (i.e., all waste packages get wet). The potential for a seepage fraction to be less than one is uncertain, but any value less than one would lead to a lower estimate of dose compared to the bounding case. The NRC staff expects seepage fraction to be less than one because

- DOE's observations during a natural seepage event in the South Ramp of the Exploratory Studies Facility tunnel, which started in February 2005 and continued for several months, supported a seepage fraction significantly smaller than one. Results of a DOE simulation using its seepage model qualitatively reproduced the seepage fraction deduced from observations in the tunnel (SAR Section 2.3.3.4.3).

- The NRC staff's analysis of site features that may reflect the spacing of flowing fracture suggests the seepage fraction is likely less than one (Basagaoglu, et al., 2007aa). The NRC staff's analysis suggests that the average seepage fraction is uncertain, but like DOE's estimate, is likely less than one. The features considered in the NRC staff's analysis include (i) fracture spacing, including long through-going fractures; (ii) structural features such as highly fractured zones and faults; and (iii) spacing of features with secondary mineralization.

Representation of Spatial Variability

The NRC staff reviewed the representation and propagation of spatial variability across the repository to determine whether DOE's model underestimated seepage. DOE incorporated spatial variability at several levels in developing its seepage results for performance assessment, including (i) integration of variability from upstream model results, (ii) variability of permeability and capillary strength in the seepage model, (iii) incorporation of a flow-focusing factor, and (iv) abstraction of spatial variability for performance assessment. The key aspect for the NRC staff's evaluation of adequacy of spatial variability reflected in the performance assessment is the upscaling of results to five percolation bins for the entire repository.

DOE incorporated spatial variability in the seepage by

- Integrating aspects of spatial variability related to net infiltration and large-scale heterogeneities from the ambient site-scale unsaturated-flow model (and propagated to the Multiscale Thermal-Hydrologic Model) directly into the seepage model through the input of percolation distribution across the repository. This aspect of variability is evaluated in SER Section 2.2.1.3.6.3.2.

- Incorporating spatial variability and uncertainty in permeability and capillary strength directly into the seepage model used to create the seepage lookup tables. This incorporated variability at the scale of several drift diameters. Permeability was stochastically varied across the seepage model grid. Capillary strength was treated as an upscaled parameter for the model domain.

- Incorporating a flow-focusing factor in the seepage abstraction that addressed the possibility of convergence or divergence of flow in the rock layers above drifts. The flow-focusing factor represents intermediate-scale heterogeneity, which DOE described as falling between the grid scale of the ambient site-scale unsaturated flow model {approximately on the order of 100 m [330 ft]} and several drift diameters (seepage model grid). Flow-focusing factors increased local percolation in some areas and decreased it in other areas, but the total flux over the entire area remained constant in DOE's performance assessment. The resulting values of flow-focusing factors reflect spatial variability and range from 0.116 to 5.016, as outlined in SNL (2007bk, Section 6.6.5.2.3). To support estimates of the distribution of flow-focusing factors, DOE performed additional modeling exercises using different assumptions for calculating the focusing factor (SAR Section 2.3.3.2.3.7.6). Results of alternative flow-focusing distributions led DOE to use a narrower range for the distribution of flow-focusing factors.

- Using five percolation bins, and therefore five seepage histories, to address spatial variability in the performance assessment. The use of average seepage histories for a percolation bin represents an upscaling of spatial variability.

Because DOE upscaled seepage estimates for its performance assessment to five percolation bins, therefore losing details of spatial variability, the NRC staff considered whether the upscaling was important to performance. If the upscaling of spatial variability does not lead to underestimates of dose, then the representation and supporting basis of detailed spatial variability are not important for performance. DOE used five percolation bins to separate the repository into areas of similar percolation rates. The areas of any one bin are not necessarily contiguous. The binning of percolation rates roughly ensured spatial continuity of flow zones above and below the repository (i.e., high percolation and therefore high seepage zones correspond with high flow zones for transport below the repository). DOE (2009bo, Enclosure 2) compared calculated release results using the five percolation flux bins with results using the 3,264 locations. The analysis demonstrated that the two approaches have similar time histories of radionuclide release, but the bin approach tended to estimate larger repository-wide cumulative release of radionuclide mass to the lower unsaturated zone at 10,000 years and 1 million years. Comprehensive model results, however, predicted higher doses at intermediate times during the initial 10,000 years because of radionuclides that are solubility limited and have intermediate to high sorption coefficients, as shown in DOE (2009bo, Enclosure 2, Figures 2 and 3). However, higher predicted dose for the comprehensive model compared to results from the percolation bin approach is not significant, because the NRC staff concluded in SER Sections 2.2.1.3.1.1 and 2.2.1.3.2.6 that the drip shields will remain intact beyond 10,000 years. The NRC staff concludes that DOE adequately represented spatial variability of percolation and seepage flux because DOE demonstrated that the upscaled percolation bin approach in the TSPA abstraction does not underestimate cumulative radionuclide release rates compared to estimates derived from representations with detailed spatial variability.

Thermal Seepage

DOE described two important features created by the thermal perturbation that affects seepage into drifts: the dryout zone around a drift and a reflux zone at the outer edge of the dryout zone. DOE's abstraction for thermal seepage sets seepage to zero for drift wall temperatures exceeding 100 °C [212 °F]. This temperature threshold for seepage is the focus of the NRC staff's evaluation of thermal seepage in the following paragraphs. The NRC staff reviewed the description of features and processes incorporated into the conceptual and numerical models that DOE used to develop the seepage threshold of thermal seepage. Considering uncertainty derived from observations used to develop the thermal seepage abstraction, the NRC staff focused its evaluation on the effect of thermal seepage estimates on performance.

DOE described the predominant seepage barrier capability for the thermal period as the elimination of liquid flux at the drift wall due to the dryout zone. DOE referred to this as the vaporization barrier (SAR Section 2.1.2.1.6.2). Flow diversion due to capillary forces (capillary diversion) remains a relevant process at all temperatures. DOE indicated that this vapor barrier would eliminate liquid water reaching the drift wall at temperatures exceeding 100 °C [212 °F] (SAR Section 2.3.3.4). In DOE's thermohydrological characterization, two-phase flow (liquid and vapor) in the host rock occurs at the outer edge of the dryout zone. Referred to as the reflux zone or heat pipe, because of increased heat transfer, evaporated liquid water rises and condenses in a continuous cycle. This zone of elevated liquid saturation above the dryout zone can serve as a supply of water added to the local percolation that potentially may breach the dryout zone as focused flow in large fractures, possibly reaching the drift wall and dripping into the drift.

DOE separated the thermal evolution into three regimes: dryout, transition, and low temperature (SAR Section 2.3.5.4.1.1.3). DOE asserted that no water enters the drift during the dryout period, and seepage may occur during the transition period and continue into the lower temperature period transitioning to ambient temperature conditions. DOE defined the dryout period as the time when drift wall temperatures are estimated to exceed 100 °C [212 °F]. DOE eliminated seepage into drifts at a threshold value of 100 °C [212 °F] for intact and partially degraded drifts, but no threshold was implemented for fully collapsed drifts in the seismic scenario. Drift wall temperature is provided to the thermal seepage abstraction from the Multiscale Thermal-Hydrologic Model, which incorporates host rock heat transport, dryout, and rewetting (presented in SAR Section 2.3.5.4; reviewed by the NRC staff in SER Section 2.2.1.3.6.3.3).

DOE derived the seepage threshold value of 100 °C [212 °F] from process-level thermohydrological modeling exercises to evaluate the possibility of preferential flow breaching the dryout zone under different realistic and bounding flow conditions. DOE described the thermal seepage model as a dual-continuum (matrix and fracture) representation with coupled heat and mass transport. The model necessarily uses a different property set than that used for the fracture-only continuum models for ambient seepage. DOE assumed that hydrologic properties need not incorporate the effect of thermal-mechanical and thermal-chemical processes. This assumption is based on results of DOE's thermal-mechanical and thermal-hydrological-chemical modeling of the heated field tests, which suggest that changes to the flow patterns are smaller than the variability and uncertainty already considered for seepage. In addition, DOE indicated these changes may be transient and likely would disappear with the decay of the thermal pulse. Generally, the modeling exercises included pulses of water applied to a single fracture and assessment of whether the pulse would evaporate before reaching the drift ceiling. Thermal aspects of the numerical model were supported by field and laboratory observations from thermal tests DOE performed. However, hydrological aspects, particularly preferential flow, are difficult to observe or measure in field tests.

The NRC staff compared DOE's description of thermohydrological features and processes during the thermal period, which is summarized in the previous paragraphs, with the NRC staff's knowledge and experience (NRC, 2005aa; Green, et al., 2008aa, and references contained therein) gained from observations and modeling of field and laboratory thermal tests. On the basis of this comparison, the NRC staff concludes that DOE acceptably described features and processes important for thermal seepage and developed a framework to adequately incorporate those features and processes into the thermal seepage model.

DOE stated that the value of 100 °C [212 °F] for the thermal seepage threshold temperature at the drift wall, being several degrees above the ambient boiling point {96.3 °C [205 °F]}, accounts for modeling uncertainties and the possibility of a heat pipe occurring near the drift wall (SAR Section 2.3.3.3.4). To determine whether the seepage threshold adequately accounted for uncertainty, the NRC staff considered field test observations reflective of temperatures at which preferential flow may have occurred.

The NRC staff noted that observations at thermal field tests could indicate preferential flow occurring in the dryout zone where measured temperature exceeded 100 °C [212 °F] (Green, et al., 2008aa). Although DOE provided reasonable explanations for the observations, the NRC staff believes the difficulty in collecting relevant observations and the uncertainty in interpretations of observations both support a larger uncertainty factor than reflected by the DOE estimate of the seepage threshold temperature value.
DOE (2009bo, Enclosure 7) provided information on drip shield integrity during the thermal period and noted that intact drip shields will divert any dripping water away from waste packages. On the basis of DOE's prediction that the drip shields will remain intact and divert all seeping water away from waste packages throughout the thermal period, the NRC staff concludes that DOE's treatment of data uncertainty and model support for the 100 °C [212 °F] thermal seepage threshold are adequate for performance assessment. The NRC staff concluded in SER Sections 2.2.1.3.1.3.1 and 2.2.1.3.2.6 that the drip shields will remain intact during the first 12,000 years, which is beyond the time when elevated temperatures significantly affect seepage.

Summary

The NRC staff has reviewed the information DOE provided and concludes that the performance and treatment of the uncertainty for the seepage rate and fraction are reasonable and acceptable because they are consistent with the technical justification provided for the model abstractions and the barrier capabilities. In evaluating seepage rates for performance assessment, the NRC staff notes that the 10,000-year and million-year periods are considered separately. In particular

- For the initial 10,000 years after disposal, mean values of seepage fraction and rate and their uncertainty are not important for performance assessment, because drip shields are predicted to remain intact well beyond 10,000 years. Intact drip shields divert seeping water away from waste packages.

- The thermal seepage abstraction that shows no seepage occurring when the drift wall temperature exceeds 100 °C [212 °F] is acceptable because drip shields remain intact well beyond the thermal period and therefore divert any seeping water away from waste packages.

- For the period from 10,000 years to 1 million years, DOE's seepage and dose calculations are dominated by the seismic ground motion seepage scenario. Average seepage fraction (69 percent) and rate (49 percent of percolation) are acceptable because reasonable increases accounting for uncertainty would not significantly affect performance assessment calculations and dose results.

- Seepage estimates for the igneous intrusion and seismic fault displacement modeling cases are acceptable because DOE uses conservative assumptions for estimating seepage.

2.2.1.3.6.3.5 In-Drift Convection and Moisture Redistribution

This section contains the NRC staff's evaluation of DOE's models, data, and results representing in-drift convection and moisture redistribution. Condensation flux, which results from moisture redistribution via vapor movement, is added to the seepage flux to obtain the total flux of water that may reach the engineered barrier components.

In SAR Section 2.3.5.4.2.1 and SNL (2007bl, Section 6.1), DOE described in-drift convection and moisture redistribution as driven by temperature differences between the waste package, drift wall, and other engineered components. In DOE's conceptual model, decay heat from emplaced waste will create large temperature differences, both radially and axially within a drift. The temperature differences will produce buoyancy-driven natural convective flow of air inside the drift opening that will increase heat transfer and redistribute moisture. Convective air flow will cause water evaporation at warmer locations in the host rock and subsequent transport by in-drift convection to cooler locations where it may condense on cooler surfaces. DOE described the ensemble of evaporation, convective moisture redistribution, and condensation on cooler surfaces inside the drift as the cold trap phenomenon.

The in-drift convection and condensation models provide two outputs. First, the convection model provides support for the effective thermal conductivity used in the thermohydrological model, reviewed in SER Section 2.2.1.3.6.3.3. Second, the condensation model, using dispersion coefficients calculated from the convection model, provides probability and flux of condensation to the performance assessment. Condensation is added to the dripping flux to obtain a total seepage flux entering the drifts. Condensation is linked to DOE's abstraction for chemistry of liquid water contacting engineered barrier components (reviewed in SER Section 2.2.1.3.3) and to flux of water in the invert, which influences radionuclide transport (reviewed in SER Section 2.2.1.3.4).

The NRC staff's evaluation of the convection and condensation models and results are divided into two parts: (i) in-drift heat transfer and convection and (ii) moisture redistribution and condensation.

2.2.1.3.6.3.5.1 In-Drift Heat Transfer and Convection

The NRC staff reviews DOE's conceptual model for in-drift heat transfer and the implementation of the numerical model in this section. The review considers the adequacy of the heat transfer model to estimate representative dispersion coefficients and effective thermal conductivity.

In-drift heat exchange processes involve conduction, convection, radiation, and phase-change (latent) heat transfer (SAR Section 2.3.5.4.2.3.1). Heat transfer processes reduce temperature differences created by emplacing heated waste packages in a drift (SAR Section 2.3.5.4.2.3). Though the heat transfer model generally will be referred to here as the convection model, radiative and conductive heat transfer processes are also included in DOE's model and analyses.

DOE implemented the convection model using the commercial computational fluid dynamics solver FLUENT®, a code commonly used in industry and academia. DOE set up FLUENT® to solve the steady-state form of the Navier-Stokes equation for selected times during the thermal pulse with a model domain that is divided into a large number of computational cells. DOE models incorporate, as appropriate, the complex arrangement of engineered components and take advantage of vertical axial symmetry to reduce computational effort. Radiation and conduction are included, but latent heat transfer is excluded because DOE demonstrated in SNL (2007bl, Sections 6.3.7.2.4 and 6.3.5.1.2) that it does not significantly affect overall heat transfer and convection. Independent NRC experiments and analyses (Fedors, et al., 2004aa) similarly conclude that latent heat transfer is a small component of overall heat transfer. For radiation and conduction, DOE used standard heat transfer models and relevant parameter values (e.g., Incropera and DeWitt, 2002aa; Kreith and Bohn, 2001aa).

The NRC staff focused its review of the heat transfer on the convection aspect of the DOE model because convection most directly affects moisture redistribution and is the most complex of the heat transfer processes in the drifts. Asymmetric geometry inside the drift leads to complex convective flow patterns at different scales. SNL (2007bl, Sections 6.1 and 6.2) used numerical models at local and drift scales to represent heat transfer at scales ranging from large-scale heat transfer along drifts (center to repository edge) to small-scale heat transfer across boundary layers at solid-air interfaces. DOE's local-scale model emphasizes cross-sectional patterns in its simulations of temperature gradients between the waste package, drip shield, and drift wall. DOE's drift scale models address temperature gradients between the hot repository center and cooler edges. Model support was provided by DOE's laboratory convection experiments and other experiments in the scientific literature using similar geometries (e.g., Kuehn and Goldstein, 1978aa).

The NRC staff concludes that DOE has adequately incorporated convective patterns in its conceptual and numerical models for convection. The NRC staff notes that analysis at two different scales is a generally accepted scientific technique for simulating large systems covering a wide range of scales and incorporating complex geometries. Furthermore, the NRC staff concludes that DOE adequately described and incorporated the important heat transfer processes and engineering design features in developing its conceptual and numerical convection models. Independent NRC experiments and associated modeling projects (Das, et al., 2007aa; Green and Manepally, 2006aa; Manepally, et al., 2007ab) support the NRC staff's conclusion.

In its two- and three-dimensional convection models, DOE used dimensions and physical properties of waste packages, drip shield, invert, and heat loads consistent with upstream models, as shown in SAR Sections 2.3.5.4.2.2 and 1.3.2 and SNL (2007bl, Section 4.1), and used values for physical properties of fluids and solids derived from standard heat transfer textbooks (i.e., Incropera and DeWitt, 1996aa; Kreith and Bohn, 2001aa). DOE assumed that convection is based on pure air (i.e., without water vapor) and demonstrated that this assumption would slightly underestimate in-drift vapor transport, as outlined in SNL (2007bl, Section 6.1.3.2.1). Independent NRC staff analyses similarly used and justified this assumption (Fedors, et al., 2004aa). On the basis of this assumption, DOE used a neutrally buoyant tracer gas in simulations and calculated dispersion coefficients using the resulting concentration gradients. Independent analyses reported in Fedors, et al. (2004aa) support use of this assumption. On the basis of these considerations, the NRC staff concludes that DOE adequately represented physical parameters for solids and fluids and for geometrical parameters in convection flow analyses.

The NRC staff reviewed DOE's use of the convection model results to estimate dispersion coefficients for the condensation model. DOE stated that dispersion coefficients are dependent on a number of factors, including axial drift wall temperature variation, convective flow pattern, presence of drip shields, and time, as outlined in SNL (2007bl, Section 6.2.7). DOE calculated dispersion coefficients at two locations in the simulated drift and at discrete timesteps during the thermal pulse. DOE addressed uncertainty in the dispersion coefficient using parametric studies and bounding analyses, as described in SNL (2007bl, Sections 6.1.7 and 6.2.7). The NRC staff reviewed the representativeness and uncertainty of the calculated dispersion coefficients for their intended usage with the condensation model. In particular, DOE did not address some uncertainty in the convection model output, including (i) the basis for the representative dispersion coefficients being spatially constant, rather than varying along drifts; (ii) justification for selecting the two specific locations in the analog 71-m [233-ft] drift for calculating dispersion coefficients and how the values are representative of an entire drift length; and (iii) the effect of the revised heat load scenario (SNL, 2008ai) on calculations. Although DOE's convection model is acceptable for calculating dispersion coefficients, the approach for estimating representative values from the model output is not fully supported for use in the condensation model because of the three aspects of uncertainty in representative dispersion coefficients described previously. However, the NRC staff concludes that the choice of dispersion coefficient values is not important for repository performance. This conclusion is supported by the NRC staff's conclusion that drip shields remain intact for 12,000 years (SER Sections 2.2.1.3.1.3.1 and 2.2.1.3.2.6) and therefore prevent direct contact of condensation with waste packages during the first 2,000 years, when DOE indicates that condensate formation is significant.

The NRC staff also reviewed DOE's use of the convection model results to derive estimates of effective thermal conductivity for the porous media representation of air gaps in the thermohydrologic submodel of the Multiscale Thermal-Hydrologic Model. DOE computed an effective thermal conductivity from output of the convection model between (i) drip shield and drift wall and (ii) drip shield and waste package, as shown in SNL (2007bl, Table 6.4.7-3). The calculated values supported the Francis, et al. (2003aa) correlations used in the Multiscale Thermal-Hydrologic Model submodels, as described in SNL (2007aj, Appendix I[a]). The NRC staff concludes that the use of the convection model to estimate effective thermal conductivity is acceptable because it follows a widely accepted technical approach used in engineering analyses (Kuehn and Goldstein, 1976aa, 1978aa), and the inputs are acceptable for the purpose of the model, as concluded in the previous paragraph.

In summary, the NRC staff concludes that DOE provided acceptable support for estimates of in-drift heat transfer because (i) the convection model is based on an acceptable technical approach, (ii) inputs are supported by textbook values, (iii) data and model uncertainty were addressed with parametric studies and bounding analyses, and (iv) model inputs and results are supported by laboratory experiments. The NRC staff also concludes the approach DOE used to estimate that thermal conductivity is adequate. The NRC staff found that uncertainty was not adequately addressed in the estimation of dispersion coefficients, which were used by DOE to estimate condensation. However, the NRC staff determined that uncertainty in dispersion coefficients is not important for estimations of dose because the drip shields inhibit condensation on waste packages and the drip shields are estimated to remain intact past the period of condensation.

2.2.1.3.6.3.5.2 Moisture Redistribution and Condensation

The NRC staff reviewed the information DOE presented to support estimates of condensation flux in drifts. DOE described the conceptual, numerical, and abstraction models for moisture transport and condensation in SAR Section 2.3.5.4.2. Treatment of data and model uncertainty was described in SAR Sections 2.3.5.4.2.2 and 2.3.5.4.2.3.3. The NRC staff focused its evaluation on the consequence of condensation flux on repository performance.

Condensation Approach

This section reviews DOE's description of the conceptual, numerical, and abstracted models used to estimate condensation rate for the performance assessment model.

Moisture redistribution and condensation inside the drift is also referred to as the cold trap process. The process involves water evaporation from hotter locations, convection to cooler locations, and the condensation of vapor on cooler surfaces. DOE considered surface condensation, which requires direct contact of the convecting gas-phase with a cooler surface, but did not provide information on the potential effects of dust or volumetric condensation (Cussler, 1995aa) in its conceptualization. DOE predicted that condensation will only occur on the drift wall because the drift wall will be cooler than the drip shield, waste package, or invert at each axial position along any drift. DOE added condensation flux to the dripping rate to obtain a total seepage rate contacting engineered barrier components and reaching the invert and, therefore, affecting advective radionuclide transport rates to the natural system.

DOE described the evolution of moisture transport and condensate formation using three stages controlled by drift wall temperature (SAR Section 2.3.5.4.2.1). In Stage 1, the initial cooling stage, the drift wall temperature exceeds boiling along the entire length of the emplacement drift. DOE stated that no condensate formation takes place during the initial stage. In Stage 2, the intermediate cooling stage, the drift wall temperature exceeds boiling in most of the drift, but the end of the drift (repository edge) is below the boiling temperature. For the intermediate stage, DOE performed a bounding analysis to calculate condensation flux occurring on codisposal waste packages at cooler locations. In Stage 3, the final cooling stage, the drift wall temperature is below boiling along the entire length of the drift. In the DOE abstraction, condensation occurs at both codisposal and spent nuclear fuel waste package locations, but all condensation ceases at 2,000 years. Results for process-level models for the intermediate and final cooling stages provide the basis for the abstraction model used in the performance assessment.

For the intermediate cooling stage, DOE estimated condensation using a three-dimensional, pillar-scale thermohydrological model (SAR Section 2.3.5.4.2.4). This is an alternative conceptual model supporting the thermohydrological results that the Multiscale Thermal-Hydrologic Model calculated (SAR Section 2.3.5.4.1.3.3; reviewed in SER Section 2.2.1.3.6.3.3). Described as a bounding approach, DOE used a range of dispersion coefficients and percolation values in the three-dimensional, pillar-scale model to determine that condensation occurs on codisposal waste packages, but not on spent nuclear fuel waste packages.

For the final cooling stage, DOE used a one-dimensional analytical moisture transfer model to estimate condensation occurrence and flux when drift wall temperatures along the entire length of the drift are below boiling. DOE's network model calculates quantity of condensate at a given location along the drift (SAR Section 2.3.5.4.2.3.1) for specified percolation rates and thermal input. The one-dimensional model is based on a diffusion-type equation and uses values of dispersion coefficients calculated by the convection model as an effective diffusion-type parameter. Conductive heat transfer in host rock is based on an analytical mountain-scale conduction model, as outlined in SNL (2007bl, Section 6.3.5.1.1), and in-drift heat transfer between components is calculated on the basis of correlations derived from simple systems and reported in open literature (Raithby and Hollands, 1985aa; Kuehn and Goldstein, 1976aa; Burmeister, 1993aa). DOE considered the supply of water for evaporation at drift walls to be bounded by the sum of capillary-pumping flow and local percolation flux intercepted by the emplacement drift footprint. DOE implemented a design feature in its model that allows axial convection to convey a portion of the moisture beyond the last waste package before condensation would occur. The NRC staff considers the control parameter that commits DOE to an unheated open length at the ends of emplacement drifts (SAR Section 1.9, Control Parameter Number 01-18, Table 1.9-9) to be an important design feature that promotes removal of potential condensate moisture from the emplacement area.

DOE implements the abstraction of condensation in the performance assessment using a three-step process (SAR Section 2.3.5.4.2.4). First, DOE used the process-level condensation models to generate a set of results for different parametric variations that account for dispersion coefficient, percolation rates, invert assumptions, and temporal variation of heat load. Second, DOE developed a set of regression curves that establishes a functional relationship between percolation flux, probability of condensation, and condensate mass. Third, DOE used the regression curves for each percolation subregion to determine the occurrence (fraction of area) and magnitude of condensation. DOE added condensate flux directly to dripping flux to obtain a total flux of water entering drifts.

In DOE’s model, condensation within emplacement drifts would be altered if a disruptive event occurs during the thermal period. DOE’s abstraction sets condensation to zero once an igneous intrusion or drift collapse event occurs (SAR Section 2.4.2.3.2.1.12.3) because these processes would fill drifts with rock and substantially reduce air gaps. However, such events have low probability and DOE expects drifts to remain intact throughout the first 2,000 years, as described in DOE (2009ct, Enclosure 7) (See SER Section 2.2.1.3.2.3.3).

The NRC staff concludes that DOE provided a transparent and acceptable description of conceptual, numerical, and abstracted models and parameter inputs. This conclusion is supported by staff knowledge gained from prelicensing interactions with DOE (NRC, 2005aa), and from independent laboratory experiments and numerical modeling of convection, vapor transport, and condensation in drift analogs (Das, et al., 2007aa; Manepally, et al., 2007ab).

Condensation Results

The DOE condensation model results can be summarized as follows:

- During the intermediate cooling period, all codisposal waste package locations receive condensation dripping from the drift ceiling at rates 8 to 35 times greater than the mean seepage rate [calculated from SAR Table 2.1-11 and DOE (2010ai, Table 7)]. DOE conservatively applied condensation to all codisposal waste package locations, but no spent fuel waste package locations receive any condensation.

- During the final cooling stage (after approximately 1,500 years), mean condensation rates are less than 1 percent of mean seepage rate, and condensation only occurs at a small fraction of locations for both codisposal and spent fuel waste packages, as shown in SAR Tables 2.1-10 and 2.1-11 and DOE (2010ai, Tables 6 and 7).

The average condensation rate in a percolation bin is calculated by multiplying the fraction of waste package locations receiving condensation times the condensation flux rate, which is then added to the seepage rate (SAR Section 2.3.3) to obtain a total dripping rate.

The NRC staff considered two types of information DOE provided to evaluate the condensation model and results used in the performance assessment. First, in developing the conceptual model, DOE stated that observations of vapor movement and condensation in response to small thermal gradients in the East-West Cross-Drift indicate the importance of the cold trap process in the repository (SAR Section 2.3.5.4.2). Although no quantitative estimate could be made from observations in the East-West Cross-Drift, the NRC staff believes the observations from near-ambient conditions point to possible condensation in emplacement drifts both prior to and beyond the 2,000-year cutoff used in the DOE condensation abstraction. Second, to support model results, DOE included evaluations by two university professors in the model validation section of SNL (2007bl, Section 7.6).

On the basis of its evaluation of condensation estimates on the consequence for repository performance, the NRC staff concludes that condensation estimates are not important for repository performance. This conclusion is derived from different considerations for different time periods following repository closure. During the first 2,000 years after closure, the presence of the intact drip shield ensures that the condensate will not directly contact waste packages. During this period, DOE asserts that drip shields will be sufficiently warm that any condensation will occur on drift walls, above or away from drip shields. On the basis of the evaluation of drip shield corrosion and mechanical degradation processes, the NRC staff concludes in SER Sections 2.2.1.3.1.3.1 and 2.2.1.3.2.6 that drip shields are likely to remain intact for at least 12,000 years, which is well beyond the thermal period DOE defined for significant condensation.

The NRC staff concludes that even if condensation continued past the 2,000-year limit DOE specified in the TSPA, it will not significantly affect repository performance for the following reasons:

- For condensation during the period from 2,000 years to 12,000 years after closure, condensation rates are not important because the drip shields will remain intact and therefore divert water from the waste package. The NRC staff does not expect dripping on waste packages to occur due to condensation underneath the drip shields because (i) the underside of the drip shield above the waste package is not a likely location for condensation to occur, because there are cooler surfaces under the drip shield that are not directly above the waste package, and (ii) the exchange of air between the inside and the outside of the drip shields reduces the likelihood of condensation under the drip shield. The NRC staff notes that DOE’s condensation rate estimate is zero during the 2,000 years to 12,000 years period, though the NRC staff considers the possibility that evaporation and condensation rates are nonzero based on observations in the East-West Cross-Drift Test. However, nonzero condensation rates are not important to repository performance because the drip shields will remain intact and therefore divert water from the waste package, and the NRC staff does not expect dripping on waste packages to occur due to condensation underneath the drip shields.

- For condensation during the period between 12,000 years and 255,000 years, the NRC staff expects the total amount of water entering drifts would be the same regardless of the mechanism for water entering drifts. Evaporation from drift walls and vapor flow from fractures into a drift are other mechanisms for water entering drifts. The total flux of water approaching a drift limits the total water entering a drift regardless of whether that water flux is due to dripping or to evaporation and condensation. The NRC staff concludes in SER Section 2.2.1.3.6.3.4.1 that the seepage percentage is already high for this period and that further increases would minimally affect performance. In addition, the NRC staff finds that any change in the fraction of waste package locations getting wet because of condensation, rather than dripping, will not adversely affect performance, because DOE already set the seepage fraction to be a high value (see SER Section 2.2.1.3.6.3.4.1) for this period.

- Beyond 255,000 years, DOE predicts that drifts will collapse (SAR Section 2.1.2.2.6) and axial convection along drifts will no longer occur. If portions of the drifts collapse, allowing for small convection cells and potential condensation, the NRC staff expects the fraction of waste packages that get wet to remain the same or decrease.

Summary

The NRC staff concludes that, for the condensation abstraction and results, condensate estimates are not important for performance during the first 2,000 years when DOE predicts significant condensation rates because drip shields are expected to remain intact. The NRC staff concludes in SER Sections 2.2.1.3.1.3.1 and 2.2.1.3.2.6 that drip shields are expected to remain intact and divert water away from waste packages for 12,000 years. Drip shield performance is evaluated in SER Sections 2.2.1.3.1 and 2.2.1.3.2. After the drip shields degrade, the NRC concludes that bounded estimates of condensation flux potentially contacting waste packages do not significantly affect repository performance, because total flux of water entering a drift as vapor or liquid and potentially dripping onto waste packages along a drift would not increase.

2.2.1.3.6.3.6 Ambient Mountain-Scale Flow—Below the Repository

The NRC staff's evaluation of the flow field in the unsaturated zone below the repository considers how the flow magnitudes and patterns affect radionuclide transport. Flow above the repository is evaluated in SER Section 2.2.1.3.6.3.2, and flow below the repository is evaluated in this section. The site-scale unsaturated flow model provides flow fields both above and below the repository for different climates (SAR Section 2.3.2.3). In SER Section 2.2.1.3.6.3.2, the NRC staff evaluated the use of site characterization data, development of a conceptual model, calibration procedure, and validation for this model. The following factors influence aspects of the ambient site-scale flow fields that are relevant to the flow fields below the repository: (i) the CHn influences flow in the southern and northern portions of the repository footprint, (ii) the active fracture model (AFM) influences the fracture-to-matrix flux, and (iii) the uncertainty of flow fields influences transport. Output of the ambient site-scale flow model (i.e., flow patterns, water saturations, and flow rates) is direct input to the radionuclide transport abstraction, which NRC staff reviews in SER Section 2.2.1.3.7.

Evaluation of the adequacy of flow fields below the repository is separated into three parts: (i) the NRC staff's review of the DOE description of the conceptual model for flow below the repository, (ii) the NRC staff's evaluation of information and observations supporting flow features in the southern vitric CHn zone and in the northern zeolitic CHn zone, and (iii) the NRC staff's evaluation of how uncertainty in flow fields can affect repository performance regarding radionuclide transport.

2.2.1.3.6.3.6.1 Flow Model Conceptualization

DOE described aspects of the flow below the repository in SAR Section 2.3.2.2.1.4 and how these aspects are related to the hydrogeologic units (SAR Table 2.3.2-2) below the repository. The hydrogeologic units include

- TSw (Topopah Spring welded tuff); dominantly fracture flow
- CHn (Calico Hills nonwelded hydrologic units);
  - Calico Hills Formation; nonwelded vitric and zeolitic zones
  - Prow Pass Tuff and the top of Bullfrog Tuff; devitrified and zeolitic horizons
- CFu (Crater Flats undifferentiated units); varied degree of welding
- Fault zones crossing all hydrologic units

DOE described flow in the first layer underlying the repository, the TSw, as occurring dominantly through fractures (SAR Section 2.3.2.2.1.3). DOE assumed steady-state flow was based on dampening of episodic infiltration pulses by the overlying PTn unit (SAR Section 2.3.2.2.1.2; reviewed by NRC staff in SER Section 2.2.1.3.6.3.2). Percolating water moves approximately vertically from the ground surface through the proposed repository to the base of the TSw. Below the TSw, DOE described flow patterns in the CHn that differ markedly between the northern and southern portions of the repository footprint (SAR Section 2.3.2.2.1.4). The CHn is the only unit where lateral variation has been incorporated into the ambient site-scale unsaturated flow model. DOE described portions of the originally vitric CHn layer as altered to zeolites, which strongly modifies hydraulic properties. The distribution of alteration is described as increasing with depth and increasing to the north and east across the repository footprint (SAR Section 2.3.2.2.1.4). Below the southern portion, DOE expects flow in the vitric CHn to be dominated by matrix flow, because matrix permeability is higher than percolation rates. DOE expects little fracture flow where the CHn is unaltered. Below the northern portion, where most of the CHn has been altered to zeolites, perched water occurs in overlying units due to low permeability of the zeolitic tuff. DOE described the perched water as affecting performance by causing lateral flow to faults, followed by fast vertical flow and transport down to the groundwater table (saturated zone). Because flow through the matrix of vitric CHn units is much slower than flow through fractures and faults, DOE predicted travel times in the southern portion to be much longer than in the northern portion of the repository (SAR Section 2.3.8.1).

Limited information was available to support DOE's model estimates of flow patterns in the underlying Prow Pass, Bullfrog, and CFu Tuffs. DOE described these units as layers of devitrified, zeolitic, welded, and nonwelded tuff (SAR Table 2.3.2-2). In SAR Section 2.3.2.2.1.4, DOE noted that these units comprise a small volume of rock above the water table. With contrasting hydraulic properties for layers in Prow Pass, Bullfrog, and CFu Tuffs, percolating waters would be expected to switch between fracture- and matrix-dominated flow several times between the repository horizon and the water table, with the matrix layers controlling sorption and travel time. The NRC staff concludes that DOE adequately described the flow magnitudes and paths below the repository, consistent with independent staff analyses (e.g., Leslie, et al., 2007aa; NRC, 2005aa).

2.2.1.3.6.3.6.2 Flow Features Below Southern and Northern Portions of Repository

The NRC staff reviewed the support DOE provided for flow features below the repository that may affect performance. Flow patterns below the repository can be described in terms of water velocity, which together with water saturation is directly tied to transport travel times. Flow patterns can be separated into three horizons, starting at the repository and proceeding downward: (i) fracture flow in welded tuffs of the TSw, (ii) influence of nonwelded tuffs of the CHn, and (iii) flow in the variably welded tuffs below the CHn. DOE described flow through the TSw as primarily vertical and rapid because of the pervasive fracture network but described rock alteration in the underlying CHn as causing different flow patterns in the southern and northern portions of the repository footprint (SAR Section 2.3.2.2.1.4 and supporting documents). In the southern portion, travel times are slow and sorption potential is high due to flow predominantly through the matrix of the unaltered, vitric CHn. This is contrasted with fast travel times for transport in the northern portion of the repository where fracture and fault flows dominate because the low permeability of the altered, zeolitic CHn led to the formation of perched water. Below the CHn, DOE described alternating layers of tuff with differing degrees of welding that host to the present-day and past fluctuations of the water table (General Information Section 5.2.2).

There is limited access to, and therefore limited direct observations of, these units because of their depth below the ground surface and below the existing tunnel and drift. Therefore, the NRC staff considered how uncertainty caused by sparse data may affect performance of the repository. To accomplish this, the NRC staff reviewed support for the conceptual model and estimation of parameter values of the numerical model that NRC staff determined to be important for flow patterns below the repository. These aspects include (i) fracture-matrix flow in the TSw immediately below the repository, (ii) properties and distribution of vitric CHn in the southern portion below the repository, (iii) influence of zeolitic CHn on perched water below the northern portion of the repository, and (iv) uncertainty of flow patterns below the CHn.

Flow in Welded Topopah Spring Tuff

The NRC staff's review of the basis provided for flow patterns through the fracture network of the welded tuffs of the TSw immediately below the repository focuses on support for the hydrologic properties of the fracture network, including support for the conceptualization and estimation of parameter values for the active fracture model (AFM).

DOE utilized air permeability and fracture data (SAR Section 2.3.2.3.3.2) to inform calibration of fracture hydrological properties. DOE assumed that fractures follow the van Genuchten–Mualem constitutive relations for saturation, water potential, and relative permeability, adjusted by the AFM. Support for the AFM parameter values is discussed later in this section. DOE based fault hydrological properties on air permeability measurements (SAR Section 2.3.2.3.3.3) and integrated these properties into the transition of one-dimensional calibration values to three-dimensional values across the site. DOE did not include sorption on fracture surfaces (SAR Section 2.3.8.1) and represented flow through welded-tuff fractures and faults as fast [e.g., tens to hundreds of years; SAR Figure 2.3.8-49 and DOE (2009an, Enclosure 6)]. The NRC staff concludes that DOE acceptably represented uncertainty in the fracture and fault input parameters of permeability, water retention, and relative permeability. Alternative representations either insignificantly reduce radionuclide travel times or improve repository performance.

Estimation of parameter values for the AFM may affect performance because the AFM controls the flux of water from fractures into the surrounding matrix. Increasing the flux of water moving from fractures to matrix increases the movement of radionuclides into the matrix, where slower travel times and increased sorption occurs relative to the fracture continuum. Therefore, the NRC staff reviewed the basis and uncertainty of parameter values used for the AFM.

DOE described and implemented the AFM of Liu, et al. (1998aa) in the site-scale unsaturated flow model to capture the effects of gravity-driven fingering flow through a limited number of water-conducting or active fractures. In the site-scale unsaturated zone flow model, DOE kept layerwise AFM parameters constant in TSPA realizations, but varied the values with the infiltration uncertainty scenario, as shown in SAR Tables 2.3.2-8 through 2.3.2-11 and SNL (2008an, Section 6.5.6). DOE estimated the AFM parameters for the three-dimensional model by calibrating one-dimensional flow simulations with field data (SAR Section 2.3.2.4.1.2.3.2). DOE adjusted the calibrated AFM parameter values to induce perching for model layers with observed perched water, thereby forming a fast pathway for water to flow into faults and bypass the underlying low-permeability and high-sorption units. The NRC staff concludes that DOE's AFM model is acceptable for representing flow patterns below the repository because DOE demonstrated it was capable of reproducing field-injection test data and natural geochemical and isotopic observations. Further support in this conclusion is derived from NRC staff's independent analysis and modeling studies (Basagaoglu, et al., 2009aa).

The NRC staff evaluated the support DOE provided for representative estimates of the AFM parameter values used in performance assessment calculations. The NRC staff considered the (i) uncertainty from using one-dimensional model sensitivity analyses to estimate three-dimensional behavior for a three-dimensional model result and (ii) uncertainty in interpretations of observations from natural isotopes, fracture coatings, and forced-injection field tests caused by multiple processes affecting transport of natural and injected tracers. The NRC staff concludes the DOE estimation of AFM parameter values is acceptable because DOE provided results of sensitivity analyses that demonstrated no significant changes to the flow fields for reasonable ranges of the AFM parameter values (SAR Section 2.4.2.3.1.7 and references cited therein).

The NRC staff also considered integration between the unsaturated flow and transport for the AFM parameters. DOE used different values and uncertainty distributions for the AFM parameter values for flow and transport simulations. DOE used fixed values of the gamma parameter in the AFM in flow simulations (SAR Tables 2.3.2-8 to 2.3.2-11, 2.3.2-13, and 2.3.2-21 to 2.32-24). DOE, however, treated the same parameter as uncertain in transport simulations by probabilistically sampling from a distribution of values (SAR Section 2.3.8.4.5.2). From a conceptual perspective, DOE, following Zhou, et al. (2007aa), suggested that advective flow occurs mostly in large-scale fractures, but transport through matrix diffusion takes place in both small- and large-scale fracture networks in real heterogeneous fractured continua. Therefore, conceptually, there may be additional contributions to the active fracture-matrix interfacial area, across which diffusive mass transfer occurs, by small-scale fracture networks that do not contribute to large-scale flows. The effect of transport in small-scale fractures of a network is not explicitly addressed in the DOE algorithm, as shown in SNL (2008an, Equation C–40), but the effect is incorporated in the sampling of gamma for transport calculations. NRC staff reviews the distribution of gamma parameter values for transport in SER Section 2.2.1.3.7.3.2.3. Because DOE provided a reasonable basis for using different values for the AFM parameter in the flow model compared to those in the transport model, the NRC staff concludes the flow and transport models are adequately integrated in regard to the AFM.

Influence of CHn in Southern Portion of Repository

DOE described the CHn in the southern portion of the repository as unaltered, though some zeolitic alteration is present and increases with depth. DOE identified the unaltered, or vitric, zones of the CHn as an important component of the Lower Natural Barrier in terms of water travel times and the unit's capability for delaying radionuclide movement (SAR Section 2.1.2.3.1). Because the travel times through the matrix of the vitric CHn are longer than in other units above and below the CHn where fast fracture flow may dominate, transport through the vitric CHn dominates the travel times of the entire sequence of hydrogeologic units below the southern portion of the repository to the water table. Therefore, the NRC staff included the uncertainty of the hydrologic properties and the spatial distribution of the vitric CHn in its focused review.

DOE characterized flow in the vitric CHn unit as matrix flow dominant (i.e., little or no fracture flow) due to the unit's relatively high matrix permeability and porosity (SAR Section 2.3.2.2.1.4). DOE provided information from boreholes near the repository and from the Busted Butte analog site to support its characterization of the vitric CHn below the repository. Busted Butte was the location of a field experiment DOE performed to support the importance of capillarity and matrix-dominated flow in the CHn vitric tuff using several injection tracer tests (SAR Section 2.3.2.3.2.4). Noting that the CHn at Busted Butte is a distal portion of the CHn found at Yucca Mountain, as described in BSC (2004av, Appendix H), DOE provided a lithologic and mineralogic comparison of the CHn near the repository with the units at Busted Butte but did not provide a hydrologic comparison. Therefore, the NRC staff reviewed measured and calibrated values of hydraulic conductivity and porosity from the boreholes near Yucca Mountain and the Busted Butte site. Values used in the NRC staff's review are from (i) SNL (2007bj, Tables 7-8, 7-9, 7-13, and 7-14); (ii) SAR Table 2.3.2-3; (iii) BSC (2004av, Table H-3); and (iv) Flint (1998aa, Table 7). The NRC staff notes that porosity values do not significantly differ between the two sites. However, hydraulic conductivity does differ between the sites. The NRC staff's analysis of the DOE data indicates uncertainty in hydraulic conductivity is two orders of magnitude larger or smaller than the values used in the performance assessment, when considering measured values in different layers and ignoring scale effects for calibrated values. The NRC staff concludes that differences between the borehole information at Yucca Mountain and Busted Butte are not important because

- If hydraulic conductivity of the CHn matrix near the repository footprint is two orders of magnitude smaller than estimated, the NRC staff expects matrix flow to dominate in the vitric CHn unit because the matrix can still accommodate typical percolation rates.

- If the hydraulic conductivity of the CHn matrix is larger than estimated by DOE, the difference in travel time and sorption would not lead to significant changes in the release of radionuclides from the unsaturated zone. Small relative changes to the long travel times in the CHn matrix, compared to the fast travel times in fractures and faults elsewhere below the repository, are not expected to significantly affect performance.

Because the uncertainty in the range indicated by the Busted Butte experiment is not important to performance, the NRC staff concludes that DOE estimates of hydrologic properties of the vitric CHn are acceptable for performance assessment.

DOE described the spatial distribution of the CHn both laterally and vertically in the southern portion of the repository footprint. DOE indicated there were few available data to constrain the spatial distribution of vitric and zeolitic zones below the repository. As a result, uncertainty about spatial variability of the CHn units (i.e., vitric versus zeolitic) may increase with depth and distance from the repository footprint (SAR Section 2.3.2.3.5.3). In its analysis, DOE incorporated data from surface mapping and 23 boreholes spread in and around Yucca Mountain, as described in BSC (2004bt, Section 6.2.3). Six of these boreholes lie within or near the edge of the repository footprint.

To assess the reasonableness of using a small number of boreholes to represent the extent of the vitric CHn, the NRC staff evaluation of the uncertainty of the spatial distribution of the vitric tuff considers potential alternative representations. The NRC staff notes that (i) the contact between the zeolitic and vitric zones of the CHn is not well constrained because of large distances between boreholes and the horizontal and vertical complexity of the contact itself and (ii) by fixing the distribution of CHn vitric and zeolitic zones in the site-scale unsaturated flow model grid, DOE did not quantify uncertainty in the extent of the vitric unit in performance assessment calculations. The NRC staff evaluated the risk-importance of the spatial distribution of vitric tuff using a bounding estimate for the performance consequence resulting from the presence of vitric tuff. Because of slower flows and greater sorption in matrix-dominated flows through the vitric tuff, which DOE represents as underlying approximately half of the repository, the vitric tuff provides a greater barrier capability than elsewhere. In the alternative representation, the NRC staff assumed that the southern vitric portion performed similarly to the northern zeolitic portion, thereby substantially reducing the performance capability of the vitric portion. If the entire repository was represented by the generally rapid travel times and minimal sorption DOE ascribed to the northern zeolitic half of the repository footprint, then dose would at most increase by a factor of two. With this extreme bounding assumption, alternative vitric distributions would either reduce calculated dose (if the vitric area were larger) or would increase by no more than a factor of two (if no vitric unit were present). This bounding analysis provides support for the conclusion that uncertainty in the location of the vitric/zeolitic contact, and therefore the spatial distribution of the vitric unit, has only a small consequence for performance relative to DOE's modeled results. Therefore, using a performance metric, the NRC staff concludes that the small number of boreholes available to constrain the distribution of the vitric CHn is acceptable. In addition to the bounding analysis, the NRC staff compared DOE's description, integration, and interpolation of sparse borehole data with the NRC staff's knowledge of site characteristics, as outlined in Leslie, et al. (2007aa, Section 6.4) and NRC (2005aa, Section 5.1.3.6.4). The NRC staff concludes that DOE acceptably represented the estimated areal extent and vertical variations of the vitric and zeolitic units because DOE represented the contact between vitric and zeolitic units as consistent with geological conceptualizations considering the interplay between faulting and stratigraphic dip of the layers.

Influence of CHn in Northern Portion of Repository

DOE described released radionuclides in the northern portion of the repository as starting out in the welded units of the TSw; proceeding vertically, predominantly in the fracture system to perched water above the zeolitic zones of the CHn; predominantly bypassing the low-permeability, high-sorptivity zeolitic zones by rapidly moving laterally to faults in the perched water body; and finally rapidly moving vertically within faults to the water table (SAR Section 2.3.2.2.1.4). DOE did not include sorption on fracture surfaces (SAR Section 2.3.8.1) and represented flow through welded-tuff fractures and faults as fast [e.g., tens to hundreds of years; SAR Figure 2.3.8-49 and DOE (2009an, Enclosure 6)]. DOE stated that neglecting sorption on fracture surfaces and representing fast flow in fractures and faults were conservative assumptions for estimating dose in the performance assessment. In this section, the NRC staff reviews DOE's treatment of model uncertainty for (i) perching of water and (ii) extent of the zeolitic unit that causes the perching to ensure that DOE's representations did not lead to underestimates of dose.

DOE implemented a permeability-barrier conceptual model for perched water (SAR Section 2.3.2.4.1.2.4.4) in which sufficient local percolation flux, poorly interconnected and conductive fractures, and locally low vertical and horizontal permeabilities contribute to the occurrence of perched water. DOE incorporated the conceptual model for flow in the perched water by adjusting the calibrated model parameters for the layers where perched water has been observed in the field by the applicant (SAR Section 2.3.2.4.1.2.4.4). DOE identified a reasonable alternative model for perching of water whereby slow vertical flow through the zeolitic portions of the CHn unit occurs at rates much smaller than percolation (SAR Section 2.3.2.4.2.1.3). DOE conservatively selected a perched water model that provides a small barrier capability for repository performance. Based upon the above, the NRC staff concludes the permeability-barrier conceptual model adequately represents field conditions for performance assessment.

DOE constrained the spatial extent of zeolitic CHn, and therefore perched water, in the site-scale unsaturated zone model using borehole data. Through hydraulic testing and interpretation of borehole observations, DOE suggested that the volume and extent of the perched-water bodies at Yucca Mountain may vary greatly (SAR Section 2.3.2.2.2.4). The extent of perched water is inversely related to the extent of vitric CHn, which the NRC staff reviewed in the previous subsection on the southern portion of the repository. The NRC staff considered the possibility that DOE underestimated the extent of perched water because this would lead to overestimates of average travel times, and therefore potentially lead to underestimates of dose. The NRC staff concludes that it is unlikely that the extent of perched water is significantly underestimated because of geospatial constraints informed by the locations of boreholes and DOE's reasonable, but conservative, placement of the zeolitic contact with the vitric CHn.

In summary, the NRC staff concludes that DOE acceptably represented the influence of the CHn unit in the northern portion of the repository footprint because alternative models for the cause of perched water and the extent of perched water would either increase travel times or insignificantly reduce travel times.

Flow Patterns below the CHn

DOE provided sparse observations related to hydrologic properties and flow patterns below the CHn in the northern and southern repository areas traversing the Prow Pass, Bullfrog, and CFu Tuffs. In its review of flow patterns below the CHn unit, the NRC staff considered uncertainty caused by this limited information.

The NRC staff reviewed the hydrologic characterization of layers below the CHn. To estimate the hydrologic properties of the lowermost layers to calibrate the site-scale unsaturated flow model, DOE supplemented the available information and observations with analog data from the PTn and TSw (SAR Section 2.3.2.3.5.3). The NRC staff considered the effect on uncertainty from the limited data available from scattered boreholes that reach deep enough to cross these units. The NRC staff concludes that uncertainty in hydrologic properties and flow patterns in the Prow Pass, Bullfrog, and CFu Tuffs does not significantly affect dose calculations because (i) flow paths in the northern portion of the repository bypass these units, (ii) travel time in the southern portion of the repository is dominated by transport time in the CHn vitric tuff horizons, and (iii) assigned hydraulic properties lead to more focusing of flow and therefore increase transport velocities through the highly sorbing matrix units in DOE's flow model.

The NRC staff considered model support for the spatial distribution of zones of focused flow potentially provided by the pattern of water table temperatures. The NRC staff expects that large-scale zones of focused flow may depress the geothermal gradient in the unsaturated zone and perturb the temperature at the water table. Temperatures at the water table might reflect large-scale flow features such as (i) localized and high flux rates predicted by the unsaturated zone model in faults; (ii) low flux reaching the water table below zeolitic rocks, which predominate in the northern half of the repository; or (iii) flux rates focused by the decreasing areal extent of vitric CHn with depth, which predominates in the southern half of the repository.

Alternative interpolations of water table temperature were presented in the SAR Figure 2.3.2-37, SNL (2008ag, Figure 6.3.1-7), and Sass, et al. (1988aa). Because the distribution of water table temperatures in any of the interpolations was not consistent with the large-scale spatial distribution of percolation in DOE's site-scale unsaturated zone flow model, DOE stated that water table temperature may not be a sensitive indicator to percolation rate, as outlined in DOE (2009cy, Enclosure 1). DOE stated that multiple factors make it difficult to interpret potential relationships between temperature and percolation, including (i) uncertainty in ground surface temperature, (ii) thickness of the unsaturated zone, (iii) uncertainty in thermal conductivity of unsaturated zone units, (iv) influence of vertical groundwater flow in the saturated zone, and (v) uncertainty in the deep subsurface heat flux. The NRC staff notes that complexity of processes and features in the unsaturated zone may negate the usefulness of temperature data to support the conceptual model and numerical modeling results that exhibit focused flow. Therefore, in the paragraph below, the NRC staff instead considers the consequences of uncertainty in focused flow caused by sparse data.

The locations of zones of focused flow are fixed in DOE's model. Because no other information was available to support predictions of the spatial distribution of flow reaching the water table, the NRC staff considered the consequences of uncertainty in the location of focused flow in the lower part of the unsaturated zone. DOE demonstrated with sensitivity analyses that the exact locations of radionuclide release from the unsaturated zone to the saturated zone are not important in the performance assessment (SAR Section 2.3.2.3.5.4; see SER Section 2.2.1.3.9.3.1.2). Because the radionuclide release distribution is not important to performance, the distribution of focused flow also is not important to performance. On the basis of DOE's sensitivity analyses and the fact that less focusing would improve predicted repository performance, the NRC staff concludes that DOE acceptably accounted for the sparse available data in model development for flow paths below the repository.

The NRC staff reviewed water table position because of its effect on unsaturated zone transport length. Future, wetter climates affect flow below the repository in two ways: (i) increased flow rates and (ii) water table rise. The increased percolation rates during future climates are evaluated as part of the site-scale unsaturated flow model in SER Section 2.2.1.3.6.3.2. The present-day water table is located in the dipping layers of the Prow Pass, Bullfrog, and Crater Flat Tuffs. Mineralogical and geochemical evidence suggests the water table occurred at higher elevations in the past. DOE stated that observations indicate that perched water elevations underneath the repository did not rise above the present levels; therefore, the NRC staff only considered the uncertainty of a higher water table. DOE accounted for water table rise under future, wetter climate conditions by raising the location to a uniform 850-m [2,790-ft] elevation, which is approximately 120 m [394 ft] higher than the present-day estimate for the water table position. DOE stated (SAR Section 2.3.2.5.2) that this rise in the water table is significantly greater than indicated by geologic evidence, which includes mineralogic alteration, isotopic ratios in secondary minerals and flow modeling exercises with increased precipitation and recharge. The NRC finds that the water table elevation DOE set for performance assessment is higher than that indicated by geologic, mineralogic, and isotopic data, or flow modeling exercises. Because a greater rise in the water table elevation reduces the transport path length for the unsaturated zone, and therefore shortens travel times and reduces potential sorption capacity of the unsaturated zone, the NRC staff concludes that DOE acceptably incorporated the effects of future, wetter climates on flow paths below the repository.

2.2.1.3.6.3.6.3 Adequacy of Flow Fields for Transport

The NRC staff reviewed the effect of uncertainty of flow fields on transport through the unsaturated zone. DOE passed the flow fields below the repository to the Unsaturated Zone Transport (SAR Section 2.3.8.5) portion of the DOE performance assessment. Overall, DOE considered advection to be the most important transport process in the unsaturated zone because the rate of water movement in the unsaturated zone largely controls radionuclide travel times, as outlined in SNL (2007bj, Section 6.1.2.1) and DOE (2009an, Enclosure 6). DOE also identified matrix diffusion and sorption as highly important for moderately to strongly sorbing radionuclides, particularly radionuclides with a short half-life that pass through a matrix unit, as described in DOE (2009an, Enclosure 6). DOE identified matrix diffusion and sorption as more important in the southern half of the repository because of the control from matrix transport in the CHn vitric facies, and more important for the 10,000-year period than the million-year period, as outlined in DOE (2009an, Enclosure 6). On the basis of these DOE assessments, the NRC staff focused its review on the flow fields with respect to transport for non-sorbing and moderately to strongly sorbing dissolved radionuclides. The NRC staff evaluated the repository performance with respect to unsaturated zone transport, including colloidal transport, in SER Section 2.2.1.3.7.3.2.4.

Flow path differences between the northern and southern portions of the repository influence the travel times of non-sorbing and sorbing radionuclides. DOE provided model results (SAR Figures 2.3.8-36 and 2.3.8-49) that showed three predominant types of transport pathways. These are (i) fast transport for fracture releases occurs in the northern half of the repository, with mean travel times of years to centuries; (ii) moderately slow transport pathways for both matrix and fracture releases go through the southern half of the repository, with mean travel times of centuries to millennia; and (iii) slow transport through the matrix for radionuclides released into the matrix of the TSw tuff with mean travel times of millennia, with a small percentage transferring to the fracture system and reaching the water table more rapidly. The DOE ambient site-scale unsaturated zone model includes perching below the repository horizon in the northern half of the repository. In the DOE implementation, perching diverts fracture waters into faults and thereby creates a large difference in travel times for the northern and southern halves of the repository.

The NRC staff first considered non-sorbing radionuclides. In DOE (2009am, Enclosure 1), several single-realization simulations with 30th and 50th percentile infiltration maps were used to illustrate the effect of transport properties on performance for the unsaturated zone. Using data from DOE (2009am, Enclosure 1, Table 4) for the seismic ground motion scenario, DOE calculated that the total activity released from the unsaturated zone in the initial 10,000 years is 73 percent of the total activity released from the EBS for Tc-99 (a non-sorbing radionuclide) in the northern half of the proposed repository and 78 percent in the southern half. DOE calculated that total Tc-99 activity released from the unsaturated zone during the 1-million-year period is at least 98 percent of that released from the EBS for the igneous intrusion and seismic ground motion scenarios, regardless of release location.

The NRC staff concludes that DOE acceptably represents flow in the unsaturated zone for non-sorbing radionuclides because the DOE transport model represents radionuclide transport processes through the unsaturated zone as not substantially reducing the activity of non-sorbing radionuclides released from the EBS.

The NRC staff evaluated the extent to which the unsaturated zone flow fields affect DOE’s performance assessment with respect to moderately to strongly sorbing radionuclides. DOE classified the porous matrix as either zeolitic, devitrified, or vitric and assigned all three classifications with sorptive capability. DOE explained that sorbing radionuclides are preferentially released to the fracture system as a result of sorption within the invert, as outlined in DOE (2009am, Enclosure 1, Section 1). As a result of the DOE release and flow models, which route fracture waters through the perched zone and into faults draining to the water table in the northern half of the repository, sorbing radionuclides predominantly bypass the matrix in the north. In the DOE flow model, both matrix and fracture waters pass into the matrix of the permeable and sorbing vitric CHn unit in the south.

The NRC staff concludes that, for transport calculations of sorbing radionuclides in the northern portion of the proposed repository, the applicant's conceptual model for perching represents a conservative approach for flow from the repository horizon to the water table because almost all fracture waters and some matrix waters experience short travel times from the top of the perched zone to the water table. For the southern portion of the repository, DOE demonstrated that reasonable changes to the hydraulic properties of the vitric CHn matrix will not significantly change the flow regime (SAR 2.3.8.5.2.2). The NRC staff notes that reasonable increases or decreases in hydraulic conductivity of the vitric CHn would minimally affect travel times of sorbing radionuclides. The NRC staff concludes that, for transport calculations of sorbing radionuclides in the southern portion below the proposed repository, the applicant's model represents a reasonable approach because the CHn vitric units have large matrix saturated hydraulic conductivity values large enough to transmit typical percolation rates at Yucca Mountain, so that matrix-dominated flows will occur.

2.2.1.3.6.3.6.4 Summary

On the basis of evaluations of the northern and southern portions of the proposed repository footprint, the NRC staff concludes that the range of flow fields generated from DOE's site-scale flow model adequately represents model and data uncertainty for performance assessment calculations. The NRC staff reaches this conclusion because (i) the resulting flow fields are unlikely to overestimate radionuclide travel times from the proposed repository to the water table and (ii) different parameter value sets either would minimally affect travel times or would increase travel times. Because the flow fields are directly used as input in the transport model abstraction, the NRC staff concludes integration between the unsaturated flow and transport abstractions is acceptable. The NRC staff evaluates DOE's overall approach to transport modeling in SER Section 2.2.1.3.7.

2.2.1.3.6.4 Evaluation Findings

The NRC staff has reviewed the SAR and other information submitted in support of the license application, which includes information required by 10 CFR 63.21(c)(1), (9), (10), (15), and (19), and finds, with reasonable expectation, that the relevant requirements in 10 CFR 63.114 and 10 CFR 63.342 are satisfied regarding the abstraction of unsaturated zone flow, thermal conditions in the host rock, and in-drift thermohydrological conditions excluding conditions for the engineered components.

In particular, the NRC staff finds that DOE has adequately

- Included data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the surface and subsurface from the site and the region surrounding Yucca Mountain and provided adequate information on the design of the

EBS to define parameters and conceptual models used in the performance assessment calculation, in compliance with 10 CFR 63.114(a)(1)

- Accounted for uncertainty and variability in the parameter values used to model unsaturated zone flow, in compliance with 10 CFR 63.114(a)(2)

- Considered and evaluated alternative conceptual models that are consistent with currently available data and scientific understanding, in compliance with 10 CFR 63.114(a)(3)

- Provided a technical basis for the inclusion of FEPs affecting unsaturated zone flow, consistent with the limits on performance assessment in 10 CFR 63.342, and evaluated in sufficient detail those processes that would significantly affect repository performance, in compliance with 10 CFR 63.114(a)(4-6)

- Provided technical bases for the models of unsaturated zone flow used in the performance assessment to represent the initial 10,000 years after disposal, in compliance with 10 CFR 63.114(a)(7)

- Used performance assessment methods for the period of geologic stability consistent with the methods used to demonstrate compliance for the initial 10,000 years after disposal, in compliance with 10 CFR 63.114(b)

- Included in the performance assessment for the period of geologic stability those FEPs used for the initial 10,000-year period, consistent with specific limits, in compliance with 10 CFR 63.342(c)

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CHAPTER 10

2.2.1.3.7 Radionuclide Transport in the Unsaturated Zone

2.2.1.3.7.1 Introduction

Safety Evaluation Report (SER) Section 2.2.1.3.7 provides the U.S. Nuclear Regulatory Commission (NRC) staff's evaluation of the U.S. Department of Energy's ("DOE" or "applicant") model abstraction for transport of radionuclides in the unsaturated zone. DOE's performance assessment analysis included the flow of water from precipitation falling on Yucca Mountain, its migration as groundwater through the unsaturated zone above and below the repository, and the flow of groundwater in the saturated zone to the accessible environment. Exposure to radionuclides in groundwater extracted by pumping is one of the principal pathways for radiological exposures to the reasonably maximally exposed individual and for releases of radionuclides into the accessible environment. Therefore, as required by 10 CFR 63.114, the performance assessment analysis included radionuclide transport in the unsaturated zone among those model components that significantly affect the timing and magnitude of transport for any radionuclides released from the repository.

In its Safety Analysis Report (SAR) Section 2.3.8 (DOE, 2008ab), DOE (i) described the features, events, and processes (FEPs) that DOE included to model the transport of radionuclides in groundwater in the unsaturated zone below the repository and (ii) provided the technical basis for DOE's implementation (or abstraction) of the unsaturated zone transport model in the Total System Performance Assessment (TSPA) model. The NRC staff's evaluation focuses on the following processes, detailed in subsequent sections, that DOE included in its SAR Section 2.3.8 as important for radionuclide transport in the unsaturated zone: (i) advection, because most of the radionuclide mass is carried through the unsaturated zone by water flowing downwards to the water table; (ii) sorption, because sorption in porous media in the southern half of the repository area has the largest overall effect on slowing radionuclide transport in the unsaturated zone; (iii) matrix diffusion in fractured rock, because matrix diffusion coupled with sorption slows radionuclide transport in the northern half of the repository area; (iv) colloid-associated transport, because radionuclides attached to colloids may travel relatively unimpeded through the unsaturated zone; and (v) radioactive decay and ingrowth, because these processes affect the quantities of radionuclides released from the unsaturated zone over time. The NRC staff's review of DOE's technical basis for excluding other FEPs is addressed in the SER Section 2.2.1.2.1 (Scenario Analysis).

DOE's radionuclide transport model abstraction for the unsaturated zone utilizes information on the magnitude and patterns of groundwater flow in the unsaturated zone and the flux of radionuclides released from the waste forms and engineered barrier system (EBS). In turn, the unsaturated zone radionuclide transport abstraction provides information about the mass flux of radionuclides released to the saturated zone.

2.2.1.3.7.2 Regulatory Requirements

The applicant is required to provide information specified in 10 CFR 63.21(c)(1), (9), (10), and (15) that is related to the abstraction of radionuclide transport in the unsaturated zone.
The requirements in 10 CFR 63.114 (Requirements for Performance Assessment) and 10 CFR 63.342 (Limits on Performance Assessments) include postclosure performance assessment to demonstrate compliance with 10 CFR 63.113 (Performance Objectives for the

Geologic Repository after Permanent Closure). Compliance with 10 CFR 63.113 is reviewed in SER Section 2.2.1.4.1.

The regulation at 10 CFR 63.114 requires, in part, that a performance assessment

- Include data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the surface and subsurface from the site and the region surrounding Yucca Mountain [10 CFR 63.114(a)(1)]

- Account for uncertainty and variability in the parameter values [10 CFR 63.114(a)(2)]

- Consider and evaluate alternative conceptual models [10 CFR 63.114(a)(3)]

- Provide technical bases for either the inclusion or exclusion of FEPs, including effects of degradation, deterioration, or alteration processes of engineered barriers that would adversely affect performance of the natural barriers, consistent with the limits on performance assessment in 10 CFR 63.342, and evaluate in sufficient detail those processes that would significantly affect repository performance [10 CFR 63.114(a)(4-6)]

- Provide a technical basis for the models used in the performance assessment to represent the 10,000 years after disposal [10 CFR 63.114(a)(7)]

The NRC staff's evaluation of DOE's inclusion or exclusion of FEPs is in SER Section 2.2.1.2.1. Requirements for performance assessment for the initial 10,000 years following disposal are specified in 10 CFR 63.114(a). Requirements for the performance assessment methods for the time from 10,000 years through the period of geologic stability, defined in 10 CFR 63.302 as 1 million years following disposal are in 10 CFR 63.114(b) and 10 CFR 63.342. These sections provide that through the period of geologic stability, with specific limitations, the applicant

- Use performance assessment methods consistent with the performance assessment methods used to demonstrate compliance for the initial 10,000 years following permanent closure [10 CFR 63.114(b)]

- Include in the performance assessment those FEPs used in the performance assessment for the initial 10,000-year period [10 CFR 63.342]

The NRC staff's review of the SAR and supporting information follows the guidance in the Yucca Mountain Review Plan (YMRP), NUREG–1804 (NRC, 2003aa), Section 2.2.1.3.7, Radionuclide Transport in the Unsaturated Zone, as supplemented by additional guidance for the period beginning 10,000 years after disposal (NRC, 2009ab). The YMRP acceptance criteria for model abstractions that provide guidance for the NRC staff's review of the applicant's abstraction of radionuclide transport in the unsaturated zone are

1. System description and model integration are adequate.
2. Data are sufficient for model justification.
3. Data uncertainty is characterized and propagated through the abstraction.
4. Model uncertainty is characterized and propagated through the abstraction.
5. Model abstraction output is supported by objective comparisons.

In its review of the SAR and supporting information, the NRC staff used a risk-informed approach and guidance provided by the YMRP, as supplemented by NRC (2009ab), for aspects of radionuclide transport in the unsaturated zone important to repository performance. The NRC staff considered all five YMRP acceptance criteria in its review of information provided by DOE. In the context of these criteria, only those aspects of the model abstraction that substantively affect the performance assessment results, as determined by the NRC staff, are discussed in detail in this SER section. The NRC staff's determination is based both on risk information provided by DOE and on the NRC staff's knowledge gained through experience and independent analyses.

2.2.1.3.7.3 Technical Review

The NRC staff reviewed information in SAR Section 2.3.8 and references therein that described how DOE predicted the transport of radionuclides in the unsaturated zone below the repository. The NRC staff's technical review focused on how DOE (i) developed a system description that incorporated site-specific transport-related geological, hydrological, and geochemical features of Yucca Mountain in the unsaturated zone transport abstraction, including a description of how the transport abstraction was integrated with other TSPA model components (Section 2.2.1.3.7.3.1), and (ii) established the technical basis for modeling the major risk-significant processes related to radionuclide transport in DOE's process-level models and in the unsaturated zone radionuclide transport abstraction (Section 2.2.1.3.7.3.2).

2.2.1.3.7.3.1 System Description and Model Framework

This section provides the NRC staff's review of DOE's overall system description as it relates to a conceptual model for radionuclide transport in the unsaturated zone. This section also provides the NRC staff's review of the model framework developed by DOE for the integration of radionuclide transport in the unsaturated zone as an abstraction in DOE's performance assessment model, TSPA.

System Description

DOE used the Yucca Mountain site data in analyzing the downward flow of water in the unsaturated zone, through fractures, major faults, and rock matrix from the repository drifts to the water table. DOE used the same analytical framework for its modeling of unsaturated zone transport of radionuclides as for its site-scale unsaturated zone flow model (SAR Section 2.3.2): a three-dimensional representation of layered volcanic tuff units with specified geological and hydrological properties, in which water and radionuclides move through fractures in the rock, through the rock matrix, and between fractures and the matrix. Major faults, which DOE assumed to provide fast transport pathways through the unsaturated zone, are represented in the model framework separately by a model with limited fracture–matrix interaction in fault zones, as documented in DOE (2009am, Enclosure 2).

DOE simulated the transport of radionuclides as (i) dissolved species and (ii) attached to mobile, colloid-sized particles. These two modes of transport are subject to various physical and chemical processes that affect radionuclide transport rates. DOE's conceptual model addresses how each of the transport-affecting processes influences the rate at which radionuclides travel through the unsaturated zone relative to the rate that water travels (SAR Section 2.3.8.2).

NRC Staff's Review

The NRC staff compared DOE's conceptual model and system description of radionuclide transport in the unsaturated zone in SAR Section 2.3.8 and references therein with the NRC staff's understanding of the Yucca Mountain natural system, obtained from prelicensing field observations and independent analysis of the unsaturated zone transport processes, as identified in NRC (2005aa, Section 5.1.3.7) and Leslie, et al. (2007aa). The NRC staff concludes that DOE's conceptual model is acceptable because it includes FEPs that are reasonably expected to affect radionuclide transport in the unsaturated zone over the period of geologic stability defined in 10 CFR 63.302.

The NRC staff also concludes that DOE has provided an acceptable system description for radionuclide transport in the unsaturated zone for the following reasons:

- DOE identified Yucca Mountain site characteristics that produce vertical and lateral unsaturated zone groundwater flow pathways, lateral variability within layers (e.g., differences in the properties of the Calico Hills tuff in the southern part of the repository area compared to the northern area), and the presence of fault zones.
- DOE used Yucca Mountain site characterization data to develop geologic and hydrologic parameter values for specific rock units or to define ranges of values for these properties to address uncertainty about the natural variability of the system.
- DOE adequately identified how and where the features of the unsaturated zone beneath the repository contributed to barrier capability in the performance assessment, as detailed in SAR Sections 2.3.8.1 and 2.1.2.3 and references therein and in DOE (2009am, Enclosure 1).

Model Integration for the TSPA Code

For TSPA calculations, DOE represented unsaturated zone transport as a model abstraction that simulates the transport of dissolved radionuclides and colloid-associated radionuclides through the unsaturated zone beneath the repository. This model generates breakthrough curves at the water table (i.e., at the contact between the unsaturated zone and the saturated zone) for the 27 aqueous species and 12 colloidal species DOE determined were the most representative and risk significant (SAR Sections 2.3.8.6, 2.3.7.4.1.2, and 2.3.8.5.4). DOE simulated radionuclide transport in the abstraction with a residence-time particle-tracking technique in an external process model, Finite Element Heat and Mass Transfer Code (FEHM), as identified in SNL (2008ag, Section 3.6). FEHM simulates flow and transport in three dimensions through porous media (e.g., transmissive media such as fractures or the rock matrix). The three-dimensional volume through which the water and radionuclides travel is subdivided into a three-dimensional grid of cells, each of which is assigned fracture and matrix properties specific to that cell's spatial location. The particle-tracking technique determines the amount of time a particle spends in each cell of the model and determines, on the basis of flow field information, which cell (fracture or matrix) the particle travels to next.

DOE integrated the unsaturated zone radionuclide transport abstraction with three other TSPA model components (SAR Figure 2.3.8-2): the EBS radionuclide transport abstraction (SAR Section 2.3.7.12), the site-scale unsaturated zone flow model (SAR Section 2.3.2.4.1), and the saturated zone radionuclide transport abstraction (SAR Section 2.3.9.3).

The release of radionuclides from the EBS is simulated in DOE's EBS transport abstraction and is provided as input to the unsaturated zone transport abstraction. In the EBS, radionuclides are transported out of breached waste packages by advection (flow) or by diffusion, travel through the crushed tuff invert, and exit into the unsaturated rock at the base of the repository drift (SAR Section 2.3.7.12.3.2). At the model exit boundary, the EBS transport abstraction uses a submodel, which DOE termed the Engineered Barrier System–Unsaturated Zone (EBS–UZ) interface submodel [SNL (2007aj, Section 6.5.2.6)], to distribute the flux of radionuclides between the fractures and the rock matrix according to modeled flow conditions and concentration gradients at the boundary between the EBS and the unsaturated rock beneath the repository. DOE provided information about the EBS–UZ interface submodel in SAR Section 2.3.7.12.3.2; SNL (2007aj, Sections 6.5.2.5. and 6.5.2.6); SNL (2008ag, Sections 6.3.8 and 7.7.1[a]); and DOE (2009am, Enclosures 1, 9, 11, and 12).

As stated in DOE (2009an, Enclosure 6), the overall result of the calculations in the EBS–UZ interface submodel is that most radionuclides released from waste packages in seeping drifts are transferred by advection into fractures, and most radionuclides released from waste packages in nonseeping drifts are transferred by diffusion into the rock matrix. DOE identified the release of radionuclides from the EBS into the rock matrix as a significant barrier mechanism because DOE models indicated that radionuclides travel more slowly in the rock matrix than they do in the fractures (SAR Section 2.3.8.5.4; SAR Figure 2.3.8-49; DOE, 2009am, Enclosure 9).

In the unsaturated zone, DOE's site-scale unsaturated zone flow model passes flow field information to the unsaturated zone transport abstraction, such that radionuclide transport through the fractures and rock matrix in the model grid depends on the percolation (downward flow) fluxes provided by the flow fields. In particular, the flow fields generated by the site-scale unsaturated zone flow model provide the transport abstraction with spatial distributions of fracture-to-fracture, matrix-to-matrix, fracture-to-matrix, and matrix-to-fracture flow rates and moisture contents in the three-dimensional model framework, as detailed in SNL (2008ag, Section 6.3.9.2). During a calculated TSPA realization, the unsaturated zone transport abstraction receives flow field information from a sequence of up to four steady-state flow fields associated with different climate states (i.e., present-day, monsoon, glacial-transition, and post-10,000-year period) to account for future changes in percolation flux at specified points in time (SAR Section 2.3.8.5.3). In DOE's unsaturated zone transport abstraction, the elevation of the water table beneath Yucca Mountain increases at the transition from the present-day climate state to a future, wetter climate state. The water table remains at the higher elevation for the remainder of the realization, effectively shortening the modeled thickness of the unsaturated zone transport path (SNL, 2008an, Section 6.4.8; SNL, 2008ag, Section 6.3.9.3). All other features of the unsaturated zone transport model grid and the sampled values of model parameters for the unsaturated zone transport abstraction remain constant throughout a TSPA realization.

The output of the unsaturated zone transport abstraction calculation provides time-dependent radionuclide mass flux at the water table as input to the saturated zone transport abstraction. Specifically, the unsaturated zone transport abstraction groups the radionuclide mass fluxes into four collection regions and transfers the grouped mass fluxes to the saturated zone transport abstraction, which then initiates radionuclide transport in the saturated zone transport abstraction at an arbitrarily selected location for each of the four regions (SAR Section 2.3.8.5).

NRC Staff's Review

In this section, the NRC staff evaluates the information DOE provided in SAR Section 2.3.8 and in Sections 6.3, 6.4, and 6.5 of SNL (2008an) and references therein about the unsaturated zone transport abstraction and its integration with related model abstractions in TSPA calculations. The NRC staff has reviewed the technical basis and model properties for DOE's EBS transport abstraction in SER Section 2.2.1.3.4.3.5. The NRC staff's review of the site-scale unsaturated zone flow model is documented in SER Section 2.2.1.3.6. The NRC staff has reviewed the saturated zone transport abstraction in SER Section 2.2.1.3.9.

In reviewing the EBS–UZ interface submodel used by DOE to integrate the EBS transport abstraction and the unsaturated zone transport abstraction, the NRC staff noted that DOE represents the near-field unsaturated zone below the repository drift as a localized, two-dimensional vertical array of overlapping fractures and rock matrix. The NRC staff accordingly compared DOE's model descriptions in SAR Sections 2.3.7.12.3.2 and 2.3.8.4.1 and verified that DOE's conceptual model for the EBS–UZ interface submodel was consistent with the modeling approach for fracture and matrix cells that DOE used for the unsaturated zone transport abstraction. The NRC staff also compared the conceptual basis of the EBS–UZ interface submodel with an independent model that relied on percolation flux rates and unsaturated matrix conductivity to simulate the transfer of radionuclides to either the unsaturated zone fractures or matrix, as identified in Leslie, et al. (2007aa, Chapter 11). The NRC staff confirmed that DOE's approach resulted in releases to fractures and rock matrix that were similar to those for the independent model. The NRC staff concludes that DOE's EBS-UZ interface submodel is adequately integrated with the unsaturated zone transport abstraction and the EBS transport abstraction for the following reasons:

- DOE developed fracture and matrix water saturation and fluxes for the EBS–UZ interface submodel that were consistent with flow fields calculated by the unsaturated zone flow model, as identified in SNL (2007aj, Section 6.5.2.6).
- DOE chose values for rock properties and radionuclide transport parameters (e.g., sorption coefficients and effective matrix diffusion coefficients) in the submodel that were based on unsaturated zone transport processes and reasonable model properties for the Topopah Spring tuff subunits from DOE's unsaturated zone transport model.
- DOE used transport properties for the crushed tuff invert for the EBS–UZ interface submodel that were based on a set of data for the crushed tuff invert from DOE's EBS transport model.

The NRC staff evaluated DOE's integration of the unsaturated zone transport abstraction (SAR Section 2.3.8.5 and references therein) and the site-scale unsaturated zone flow model (SAR Section 2.3.2.4.1 and references therein). The NRC staff verified, by examining results in SAR Section 2.4 and SNL [2008ag, Section 7.1(a)], that the suite of radionuclides DOE used for the unsaturated zone transport calculations was consistent with the radionuclides that DOE's radionuclide screening analysis identified for transport in groundwater pathways, as provided in SNL (2008ag, Section 6.3.7.1.2) and in SNL (2007au, Sections 6.2 and 6.3). In evaluating DOE's integration of the unsaturated zone transport abstraction and the site-scale unsaturated zone flow model, the NRC staff makes the following conclusions:

- DOE's integration of the unsaturated zone transport abstraction and the site-scale unsaturated zone flow model is acceptable because DOE's unsaturated zone transport abstraction used a model framework, technical bases, model properties, and assumptions that were consistent with the site-scale unsaturated zone flow model.

- DOE's representation of radionuclide transport in fracture-dominated and matrix-dominated unsaturated zone flow paths is acceptable because DOE used well-documented and peer-reviewed FEHM modeling approaches (e.g., Doughty, 1999aa; Robinson, et al., 2003aa) that were consistent with the conditions and assumptions of the site scale unsaturated zone flow model.

The NRC staff evaluated DOE's integration of the unsaturated zone transport abstraction (SAR Section 2.3.8.5 and references therein) and the saturated zone transport abstraction (SAR Section 2.3.9.3 and references therein). The NRC staff concludes that DOE's model integration is acceptable for the unsaturated zone transport abstraction because DOE adequately described how its model assumptions about the transfer of radionuclides from the fractures and rock matrix of the unsaturated zone to fracture-dominated flow paths in the saturated zone were consistent with conditions identified by DOE for Yucca Mountain site characteristics, flow model properties, and differences in scale between the unsaturated zone and the saturated zone transport paths.

2.2.1.3.7.3.2 Unsaturated Zone Radionuclide Transport Processes

In DOE's unsaturated zone transport abstraction, the migration of radionuclides through the unsaturated zone is influenced by the transport-affecting processes of (i) advection and dispersion, (ii) sorption, (iii) matrix diffusion, (iv) colloid-associated radionuclide transport, and (v) radioactive decay and ingrowth (SAR Section 2.3.8.1). Advection, dispersion, matrix diffusion, and colloidal transport are transport mechanisms that move radionuclides from one location to another. In contrast, sorption may delay the transport of a radionuclide by attachment to stationary surfaces such as the rock matrix. Radioactive decay removes a radionuclide permanently from the system. Ingrowth is the replacement of a decayed radionuclide with a newly formed (daughter) nuclide, which may have different radioactivity and transport properties than the parent.

2.2.1.3.7.3.2.1 Advection and Dispersion

In DOE's unsaturated zone transport model and abstraction, advection refers to the transport of radionuclides, as either dissolved or colloid-associated phases, by the bulk movement of water. DOE stated in SNL (2007bj, Section 6.1.2.1) that advection was probably the most important transport process in the unsaturated zone because the rate of water movement largely controls radionuclide travel times in the unsaturated zone. DOE coupled the advective transport of radionuclides with the bulk movement of water in fractures, in the rock matrix, and between fractures and matrix, using the groundwater flow rates and flow paths supplied by the site-scale unsaturated zone flow model, as detailed in SAR Section 2.3.8.5.2.1 and SNL (2008ag, Section 6.3.9.2). Because the unsaturated zone flow model predicts that water flows through the unsaturated zone at different rates in different rock units, the advective radionuclide transport rates vary correspondingly at different locations in the unsaturated zone. For example, in the fracture-dominated northern part of the repository area, DOE's unsaturated zone transport model predicts generally fast advective transport of radionuclides due to high modeled flow rates in fractures and fault zones. In the southern part of the repository area, advective transport of radionuclides in the unsaturated zone is slower due to low flow rates in the matrix-dominated flow system of the Calico Hills vitric tuff units (SAR Section 2.3.8.5.4).

DOE described dispersion as a spreading plume of dissolved radionuclides caused by localized differences in flow conditions, as identified in SAR Section 2.3.8.2.2.1 and SNL (2008ag, Section 6.3.9.1). As stated in SNL (2008an, AD01, Section 4.1.6), DOE did not identify dispersion as an important transport-affecting process at the scale of the unsaturated zone transport model. However, DOE chose to include a simple fixed-value longitudinal dispersion term in the transport model to support numerical analyses of breakthrough curves at the water table (SAR Section 2.3.8.5.2.2). To estimate a value for the dispersion term, DOE used results from saturated zone flow and transport tests at Yucca Mountain that were comparable in scale to site-scale unsaturated zone flow and transport paths (SAR Section 2.3.8.5.2.2).

NRC Staff's Review

In the NRC staff's review of advection, the NRC staff examined the information DOE provided about advection in SAR Section 2.3.8 and references therein. DOE identified that advective radionuclide transport in DOE's unsaturated zone transport abstraction is determined directly from the flow field calculations supplied by DOE's unsaturated zone flow model abstraction. As a result, the DOE's technical basis for advection in the site-scale unsaturated zone flow model also provides the technical basis for DOE's representation of advective radionuclide transport in the unsaturated zone transport abstraction. Therefore, on the basis of the NRC staff's review, evaluation, and acceptance of DOE's site-scale unsaturated zone flow model in SER Section 2.2.1.3.6, the NRC staff concludes that DOE has provided an adequate technical basis for modeling radionuclide transport by advection in the unsaturated zone transport abstraction.

In the NRC staff's review of dispersion, the NRC staff examined the information DOE provided about dispersion in SAR Section 2.3.8 and references therein. The NRC staff concludes that DOE provided an adequate technical basis for dispersion in the unsaturated zone transport abstraction for the following reasons:

- DOE provided adequate mathematical examples, field observations, and process-level modeling results to support DOE's statement that dispersion did not appreciably affect radionuclide travel times in the unsaturated zone transport calculations.
- DOE's representation of dispersion as a transport process was consistent with DOE's conceptual model of fracture and matrix flow conditions at Yucca Mountain.
- DOE estimated the value of the dispersion term from site-specific field tests that were representative of the expected scale of dispersion in the unsaturated zone transport abstraction.
- DOE addressed data and model uncertainty for the dispersion term with simplifying assumptions that were appropriate for the minor effect of longitudinal dispersion on radionuclide transport through the unsaturated zone.

2.2.1.3.7.3.2.2 Sorption

Sorption is a general term for chemical and physical processes that transfer a fraction of dissolved chemical species to the surface of a solid phase. Depending on specific properties of (i) the dissolved species, (ii) the solid phase, and (iii) the liquid phase, some dissolved species will sorb more readily onto solids than others will, and some will not sorb at all. In DOE's unsaturated zone transport model, sorption of radionuclides onto rock surfaces slows the transport rate of radionuclides through the rock relative to the flow rate of water, a delaying effect that is called retardation (SAR Section 2.3.8.2.2.2). Sorption potentially can retard the transport of moderately or strongly sorbing radionuclides in the unsaturated zone for thousands of years or longer, contributing more significantly to unsaturated zone barrier capability than any other retardation process, as identified in DOE (2009an, Enclosure 6). In contrast, sorption of radionuclides onto mobile colloids may decrease the overall retardation effect.

DOE represented sorption in the unsaturated zone rock matrix with a sorption coefficient (K_d), an empirically determined or modeled value that represents the ratio of the sorbed-phase radionuclide concentration to the dissolved-phase radionuclide concentration. Low or zero values of K_d indicate that little or no sorption occurs; higher values indicate moderate or strong sorption, and therefore retardation. Factors that influence K_d values include the (i) radionuclide chemistry and dissolved-phase concentration; (ii) solution pH and major ion water chemistry; (iii) temperature of the system; and (iv) physical and chemical properties of the solid phase, including its surface area. Retardation by sorption is expressed in transport calculations by a retardation factor that depends on the value of the sorption coefficient and the physical properties (porosity and density) of the solid medium through which the radionuclide is transported. Retardation calculations assume that (i) K_d does not vary with changes in radionuclide concentration, (ii) sorption and desorption reactions are fast relative to the flow rate, and (iii) bulk chemical composition of the water is constant (Davis and Curtis, 2003aa; Langmuir, 1997aa; Davis and Kent, 1990aa).

DOE modeled sorption of dissolved radionuclide species in the unsaturated zone rock matrix at Yucca Mountain but assumed that there was no sorption on fracture surfaces, except for those portions of the model framework that are designated as fault zones, as identified in SAR Section 2.3.8.5.2.3 and DOE (2009am, Enclosure 2). DOE modeled fault zones as a fracture continuum with low porosity where sorption can occur on rock surfaces (SAR Section 2.4.2.3.2.1.9). DOE also included sorption in modeling colloid-associated radionuclide transport (e.g., SAR Sections 2.3.8.4.3 and 2.3.8.5.2.5), as reviewed separately in SER Section 2.2.1.3.7.3.2.4.

In terms of the barrier capability of the lower unsaturated zone (SAR Section 2.1.2.3), DOE attributed a higher overall importance to sorption than to any other transport process, as identified in DOE (2009an, Enclosure 6). The NRC staff has reviewed the information DOE provided about sorption in SAR Section 2.3.8 and references therein with a particular focus on (i) development of sorption values (i.e., how DOE obtained sorption data and addressed data uncertainty for a sorption model) and (ii) sorption modeling for radionuclide transport (i.e., how DOE supported the sorption model as implemented in radionuclide transport calculations).

Development of Sorption Values

In developing an unsaturated zone transport model and abstraction for performance assessment purposes, DOE assumed that four radioelements are nonsorbing (carbon, chlorine, iodine, and technetium), and DOE assigned a fixed value of $K_d = 0$ for each. For the remaining 11 radioelements that DOE modeled in groundwater transport calculations (americium, cesium, neptunium, plutonium, protactinium, radium, selenium, strontium, thorium, tin, and uranium) (SNL, 2008ag, Section 6.3.7.1.2), DOE developed ranges and statistical distributions of K_d values for each radioelement and for each modeled rock unit from a combination of empirical data, process modeling, and professional judgment, as summarized in SAR Table 2.3.8-2. DOE detailed the K_d selection process in SNL (2007bj, Appendices A, B, I, and J and Addendum 1) and in DOE (2009am, Enclosure 3).

To obtain empirical estimates of K_d values for sorption modeling, DOE grouped the various rock units below the repository into three rock types that have different sorption characteristics—zeolitized tuff, devitrified tuff, and vitric tuff (SAR Section 2.3.8.3.1). DOE measured sorption data from batch experiments that used site-specific crushed samples of these three types of tuff and saturated zone water samples from two wells (J-13 and UE-25 p#1). DOE chose the water chemistries to bracket the major ion chemistry observed in the pore waters and perched waters of the unsaturated zone [SAR Section 2.3.8.3.1 and SNL (2007bj, Section A4)]. SNL (2007bj, Section A4) provided summaries of major ion chemistry (e.g., calcium, sodium, bicarbonate) for the unsaturated zone pore waters, which were sampled by extraction from rock cores, and for perched waters, which were sampled by wells in locally saturated regions above the regional water table. DOE compared these reported ranges of unsaturated zone water chemistry with the two water saturated zone chemistries that DOE used in the sorption experiments, as identified in SNL (2007bj, Section A4).

DOE identified mineral surface area and particle size as potential sources of data uncertainty related to the use of crushed tuff in sorption experiments. The DOE approach to addressing this uncertainty was to use results from batch experiments for a range of particle sizes and to bias the minimum and maximum limits obtained for the K_d distributions toward lower (weaker sorption) values, as documented in DOE (2009am, Enclosure 3, Table 1.1.2-1). DOE referenced studies both from within and outside the DOE program in SNL (2007bj, Section 6.1.3.1) in making these selections.

In selecting experimental data to inform the TSPA K_d distributions, DOE did not include data from experiments where the final radionuclide concentration may have exceeded a solubility limit. DOE addressed this uncertainty indirectly for transport modeling purposes by assigning K_d ranges that were based on low (i.e., weaker sorption) K_d values [SAR Section 2.3.8.3.1, SNL (2007bj, Appendix A), and SNL (2007ah)].

In addition to the batch sorption data that DOE obtained by assessing K_d variability as a function of time, radioelement concentration, atmospheric composition, water composition, particle size, and temperature, DOE addressed data uncertainty by performing confirmatory column tests on selected radionuclides that DOE had identified as important contributors to mean annual dose in previous performance assessment calculations, as addressed in SAR Section 2.3.8.3.1 and SNL (2007ba, Table 4-1).

In the TSPA model calculations, DOE sampled K_d values from the ranges that DOE had developed for specific radionuclides to account for experimental uncertainty and variability in geologic conditions, including water chemistry and rock type, as detailed in SAR Table 2.3.8-2; SNL (2007bj, Appendices A, B, I, and J and Addendum 1); and DOE (2009am, Enclosure 3). To further develop K_d values for the actinides americium, neptunium, plutonium, and uranium, DOE characterized the effects of variability in geochemistry and mineral surface area using a non-electrostatic surface complexation modeling approach described in Davis, et al. (1998aa) and in SNL (2007bj, Addendum 1 and Appendix A, Sections A7 and A8). Where data were otherwise incomplete, DOE also supplemented the experimental data and the surface complexation modeling data with published K_d values from peer-reviewed scientific literature and reports prepared by other agencies {e.g., SNL (2007bj, Section A1[a])}.

NRC Staff's Review

The NRC staff reviewed the information DOE provided about the development of sorption coefficient data. Based on the NRC staff's knowledge and experience (e.g., Bertetti et al., 2011aa; Turner et al., 2002aa), the NRC staff concludes that DOE adequately incorporated sorption modeling in performance assessment calculations for the following reasons:

- DOE used site-relevant sorption data to address the anticipated effects of pH, Eh, major ion water chemistry, rock composition, rock surface area, and radionuclide concentration on radionuclide sorption concentration.

- DOE based its empirical determinations of K_d values on a set of site-specific rock types that are representative of the unsaturated zone geology.

- DOE's use of the J-13 and UE-25 p#1 water chemistries in its empirical determinations of K_d values adequately bounded the ranges reported for unsaturated zone water chemistries for major ions such as sodium, calcium, and bicarbonate at Yucca Mountain.

- DOE addressed experiment uncertainty by performing multiple batch experiments for a range of particle sizes, and identified that the effects of particle size on sorption are typically small except for the very fine (e.g., clay-sized) fraction.

- DOE validated K_d values estimated from static (batch) sorption experiments by conducting column tests, using similar materials, that added a transport component to the results.

- DOE chose to bias the minimum and maximum limits obtained for the K_d distributions toward lower (weaker sorption) values so that the effectiveness of sorption as a retardation mechanism would tend to be underestimated for dissolved radionuclides.

- DOE considered a valid, large set of radionuclides in the sorption model, using a linear K_d approach to sorption that is consistent with the other transport components of the TSPA model.

The NRC staff also reviewed DOE's surface complexation model used to extend the range of sorption data for key actinides in transport models. Based on the NRC staff's knowledge and experience with other, independently developed surface complexation models that have been used for similar insights regarding radionuclide sorption behavior (e.g., Turner et al., 2002aa), the NRC staff concludes that DOE acceptably used surface complexation modeling to extend the limited chemical conditions in the batch crushed tuff experiments and to support DOE's technical basis for the upper and lower limits of sorption coefficients for the targeted actinides.

Sorption Modeling for Radionuclide Transport

A potential uncertainty that is widely recognized and generically associated with the use of K_d values in transport modeling (e.g., Chapman and McKinley, 1987aa) is that individual K_d values are lumped parameters that do not explicitly take into account spatial and temporal variabilities or the role of specific surface-related processes that may affect radionuclide sorption. DOE addressed model uncertainty in its TSPA calculations by sampling K_d values stochastically from uncertainty distributions in which the distribution ranges were developed from the spatial and temporal ranges of expected system conditions, as detailed in SAR Table 2.3.8-2; SNL (2007bj, Appendices A, B, I, and J and Addendum 1); and DOE (2009am, Enclosure 3). In addition, DOE reduced the upper bounds of the K_d distributions for some radionuclides (specifically, those of cesium, plutonium, and radium) relative to the range indicated by available data to account for the possible effects of slow sorption kinetics for these elements, as identified in SNL (2007bj, Appendix A, Sections A8.4 and A8.6). In SNL (2007bj, Appendix A, Section A6), DOE explained that its modeled uncertainty distributions for K_d values, which used low ranges of K_d values for sorption relative to the measured values, tended to underpredict the effectiveness of sorption compared to the experimental distributions, and resulted in faster radionuclide travel times through the unsaturated zone than would otherwise be expected.

Also, rather than sample the K_d distribution independently for each radionuclide in a TSPA simulation, DOE developed a correlation matrix for the 11 sorbing radioelements on the basis of their ranked sensitivities to six variables (pH, Eh, water chemistry, rock composition, rock surface area, and radionuclide concentration). DOE used this approach to approximate similarities in sorption behavior among radioelements and to ensure that transport behaviors were represented consistently within a single realization of the model, as detailed in SNL (2007bj, Appendix B, Section B1).

In addressing model uncertainty in radionuclide transport calculations, DOE implemented K_d uncertainty distributions for matrix sorption that in most cases predicted less sorption compared to measured distributions, and DOE did not take credit for sorption (i.e., $K_d = 0$) in fractures (fast flow paths) except in fault zones. In TSPA simulations, DOE modeled sorption in fault zones using the same K_d values and K_d uncertainty distributions as the matrix properties for either devitrified tuff or zeolitic tuff, depending on the location of the modeled fault zone (DOE, 2009am, Enclosure 2). Sensitivity analyses that DOE reported in SNL (2008ag, Section P21) in which no credit was taken for sorption (i.e., $K_d = 0$) in the fault zones resulted in only a slight increase, less than 5 percent, of the mean annual dose (SNL, 2008ag).

DOE developed information from natural analogs to provide qualitative comparisons for sorption model confidence building at the field scale (SAR Section 2.3.8.4.4), and DOE used general observations of sorption-related transport behavior to support the conceptual models (e.g., SAR Section 2.3.8.4.4.6). DOE also used observations from field sites at Busted Butte, south of Yucca Mountain, and alcove tracer tests with nonradioactive chemical homologues in the Exploratory Studies Facility to provide limited quantitative evaluations of sorption in the radionuclide transport model abstraction (SAR Section 2.3.8.3.3).

NRC Staff's Review

On the basis of its review and the NRC staff's knowledge and experience (e.g., Bertetti, et al., 2011aa; Turner, et al., 2002aa), the NRC staff concludes that DOE adequately incorporated sorption modeling in performance assessment calculations for the following reasons:

- DOE based its sorption modeling on an empirical K_d modeling approach that is well established (e.g., Freeze and Cherry, 1979aa; Till and Meyer, 1983aa) and has been broadly used to describe radionuclide transport (e.g., Sheppard and Thibault, 1990aa; Chapman and McKinley, 1987aa).

- DOE defined and documented the limitations of the K_d approach and used stochastically sampled K_d probability distributions and simplifying assumptions about the effectiveness of sorption to address model and data uncertainty.

- DOE considered the range of expected site geochemical and physical conditions in developing the K_d probability distributions, and addressed uncertainty by using either low

K_d values or bounding assumptions that reduce the credit given to radionuclide sorption in the TSPA model.

- DOE adequately described how it obtained, used, and interpreted experimental data with site-specific materials, alternative computer models, field tests, and natural analogs to provide a technical basis to support the TSPA model abstraction of radionuclide sorption.

- DOE considered alternative sorption modeling approaches and used them to support the technical basis for the K_d distributions.

- DOE adequately described the method used to assess the sensitivity of radioelement sorption behavior to variability in geochemical and physical conditions, and DOE acceptably used that method to correlate sorption characteristics among the radioelements, ensuring consistency among the sorption parameters for each TSPA model realization.

- DOE identified potential sources of uncertainty on the basis of site- and radionuclide-specific data and propagated the uncertainty through the unsaturated zone transport model abstraction by using lower, limited ranges of K_d values. With respect to the TSPA model abstraction, this underprediction means that DOE takes less credit for sorption in the unsaturated zone than experimental results would indicate.

- DOE used observations from natural analogs to support model abstraction and uncertainty by constraining sorption processes in unsaturated fractured rock.

- DOE reduced the significance of model uncertainty of radionuclide transport in unsaturated zone fractures by taking no performance assessment credit at all for sorption in the fractures.

Although DOE’s approach of modeling sorption in the fault zones in the TSPA model abstraction is not the same as DOE’s treatment of sorption in fractures elsewhere in the TSPA model abstraction, the NRC staff concludes that the fault zone K_d values are acceptable because (i) DOE’s conceptualization of the sorption is based on known fault zone characteristics, and (ii) the K_d values rely on the same technical basis that DOE used to develop K_d values for the unsaturated zone matrix. DOE also demonstrated that allowing sorption in the fault zones had an insignificant effect on radionuclide transport results.

In summary, in reviewing DOE’s representation of sorption processes in the unsaturated zone, the NRC staff concludes that DOE adequately described how geochemical data were obtained, used, and interpreted to derive the K_d parameter distributions used to represent data uncertainty. Further, the NRC staff finds that where there is model uncertainty, DOE acceptably used assumptions and selected parameter values that would likely reduce the credit given to radionuclide sorption in its TSPA analysis.

2.2.1.3.7.3.2.3 Matrix Diffusion

Diffusion is a physical process in which dissolved species or suspended particles move from a region of high concentration to a region of low concentration, in accordance with the concentration gradient. DOE described matrix diffusion as a fracture–matrix interaction that uses diffusion to transfer radionuclides between fractures and the rock matrix. In

DOE (2009an, Enclosure 6), DOE identified matrix diffusion as an important transport mechanism in the unsaturated zone transport abstraction, especially for strongly sorbing radionuclides, stating that it is the main process by which radionuclides can move from a fracture-dominated flow path into the matrix.

As described by DOE in SNL (2007bj, Section 6.1.2.4), radionuclide transport by matrix diffusion in the unsaturated zone depends on two properties: (i) the matrix diffusion rate (i.e., the rate that a radionuclide can diffuse from water in a fracture into water in the pore spaces of the rock matrix) and (ii) the effective fracture–matrix interface area, which is the area across which diffusion can occur. In turn, DOE described that the matrix diffusion rate itself depends on three values: (i) the radionuclide concentration gradient between fracture and matrix; (ii) the calculated water saturation of the rock; and (iii) the value of the effective matrix diffusion coefficient, which is a measure of how readily a particular radioelement diffuses through a tortuous pathway of interconnected pores in the rock matrix, compared to the diffusion rate of the same radioelement in free water.

Development of Matrix Diffusion Coefficients

In determining values of the effective matrix diffusion coefficient, DOE used empirical data to estimate the tortuosity of interconnected pore spaces in representative Yucca Mountain tuff samples under saturated conditions. DOE adjusted the tortuosity estimates to account for unsaturated conditions in the lower natural barrier at Yucca Mountain and subsequently calculated unsaturated zone effective matrix diffusion coefficients for each radioelement. To address uncertainty in the effective matrix diffusion coefficient, DOE developed standard normal cumulative probability distributions for the effective matrix diffusion coefficient that were sampled stochastically in the TSPA analysis for each radioelement with respect to the individual model units (SAR Section 2.3.8.3.2; Reimus, et al., 2007aa).

NRC Staff's Review

The NRC staff reviewed the supporting information provided by DOE in SAR Section 2.3.8.3.2; (Reimus, et al., 2007aa; BSC 2004bi, Section 5.2.1.1) about how DOE developed effective matrix diffusion coefficients and validated the experiment observations. The NRC staff concludes that the DOE data are sufficient for model justification for the following reasons:

- In developing radionuclide-specific effective matrix diffusion coefficients for the unsaturated zone transport abstraction, DOE adopted a standard theoretical approach (e.g., Freeze and Cherry, 1979aa, Section 3.4) to estimate parameter values from laboratory measurements of diffusion properties using site-specific Yucca Mountain tuff samples.

- DOE addressed the uncertainty of the effects of secondary mineral coatings (e.g., calcite, oxides, clay minerals) on matrix diffusion models by conducting diffusion experiments with paired core samples (i.e., samples with fracture coatings and without), and determined that in the tested sample pairs, mineral coatings on fracture surfaces did not impede diffusion rates.

- DOE supported the diffusion experiment data development methods by using site characterization data to identify that fracture coatings are not abundant on fracture surfaces in Yucca Mountain tuffs.

Implementation of the Active Fracture Model

DOE stated that not all connected fractures in unsaturated rocks actively conduct water (SAR Section 2.3.2.2.2.1) and that, instead of uniform flow, individual fractures may have gravity-driven fingering flow that wets only a portion of a fracture surface (SAR Section 2.3.8.2.2.1). To adjust the size of the effective fracture–matrix interface area to account for the general observations of flow in the unsaturated fractures, DOE adopted the active fracture model (Liu, et al., 1998aa) for fracture–matrix interactions in the unsaturated zone at Yucca Mountain. In particular, DOE used an active fracture model parameter, *gamma*, and the modeled effective water saturation (i.e., the average water saturation of the connected fractures, adjusted by the unsaturated zone flow model for residual fracture saturation) to increase the modeled distance between flowing fractures. DOE's active fracture model predicts that in unsaturated fractured rock, fewer fractures in a given volume serve as flow pathways than would be expected under fully saturated conditions in the same rock volume. Having fewer flow paths in a fractured rock volume also reduces the size of the effective (i.e., wetted) fracture–matrix interface area for unsaturated zone fracture–matrix interactions, thereby decreasing the capacity of matrix diffusion to retard radionuclide transport through the fractured rock.

DOE applied the active fracture model in developing two TSPA model abstractions: (i) modeling unsaturated zone groundwater flow, as documented in SAR Section 2.3.2.4.1 and (ii) modeling radionuclide transport in the unsaturated zone, as documented in SAR 2.3.8.5.2.4. The NRC staff has reviewed DOE's implementation of the active fracture model in the site-scale unsaturated zone flow model, as documented in SER Section 2.2.1.3.6. However, DOE applied and interpreted part of the active fracture model differently for the unsaturated zone radionuclide transport abstraction. Consequently, the NRC staff conducted an additional review of the active fracture model as DOE applied it to model radionuclide transport by matrix diffusion.

In applying the active fracture model for flow field calculations in the site-scale unsaturated zone flow model (SAR Section 2.3.2.2.2.1), DOE estimated the *gamma* parameter for individual model layers by flow model calibration, as detailed in SNL (2007ad, Section 6.3.2), and assigned a fixed value of *gamma* parameter for each model layer for the flow model. In contrast, sensitivity analyses reported in SNL (2008an, Section 6.6.4) indicated that the radionuclide transport model calculations were more sensitive to *gamma* uncertainty values than were the fluid flow model calculations. DOE therefore reasoned it would be inappropriate in the radionuclide transport calculations to assume that the *gamma* parameter was tightly constrained by the fixed values used in the fluid flow model. For radionuclide transport calculations, DOE instead sampled *gamma* values independently from an uncertainty distribution that was not limited to the calibrated fluid flow model values (SAR Section 2.3.8.5.2.4). Specifically, DOE's conceptual model of matrix diffusion in SNL (2008an, Section C5) assumes that matrix diffusion should be less effective in the unsaturated rocks than in the saturated rocks due to the reduced size of the wetted fracture-matrix interface area in unsaturated fractures.

In developing and supporting the matrix diffusion model, DOE acknowledged the impracticality of conducting large-scale transport tests to observe and measure the effects of fracture–matrix interactions in unsaturated rocks under natural conditions (SAR Section 2.3.8.3), and DOE cited uncertainties about the potential significance of scale-dependent transport processes in fractured rocks (BSC, 2006aa, Section 6.4.1; Liu, et al., 2004aa). SNL (2008an, Section 6.6.4) identified that the size of the effective fracture–matrix interface area was the most uncertain term affecting radionuclide diffusion rates in the matrix diffusion model. To address model uncertainties about quantifying the effective fracture–matrix interface area for unsaturated zone transport calculations, DOE sampled the value of the active fracture model *gamma* parameter from a broad, uniform distribution that covered an intermediate range of 40 percent of all possible *gamma* values. The wide range of sampled *gamma* values produced a correspondingly wide range of results for radionuclide transport by matrix diffusion, as described in SNL (2008an, Section 6.6.4).

DOE conducted no unsaturated zone field experiments for model support specifically to evaluate DOE's matrix diffusion model under expected repository conditions, but DOE observed in several large-scale experiments in the Exploratory Studies Facility that tracer transport in fractured rocks took significantly longer than predicted by matrix diffusion in DOE's process-level models (SAR Sections 2.3.8.3.3.2.1, 2.3.8.3.3.2.2, and 2.3.8.3.3.3). DOE's simulations of these tracer tests included numerical analyses based on the same model assumptions and on the same broad range of *gamma* parameter values as DOE used for matrix diffusion in the unsaturated transport abstraction (SAR Section 2.3.8.4.4.4.3).

DOE cited field observations and numerical simulations of tracer migration in several large-scale transport experiments in the Exploratory Studies Facility to support its conceptual model of matrix diffusion in fractured, unsaturated rocks and use of the active fracture model for matrix diffusion calculations in the TSPA model (SAR Sections 2.3.8.3.3.2.1 and 2.3.8.3.3.3). DOE provided empirical observations and sensitivity analyses from field-scale experiments and process-level model analyses to address matrix diffusion model uncertainty. These were addressed in SNL (2007ad, Section 6.3.2); BSC (2004ag, Sections 7.4.1 and 7.4.2); BSC (2004av, Section 6.12.2.4); SNL (2007bf, Section 7.5); and Liu, et al. (2003aa). DOE also provided results of TSPA simulations in SAR Section 2.4.2; SNL [2008ag, Section 7.7.1(a)] and in DOE (2009an, Enclosure 6) to demonstrate that the modeled effect of matrix diffusion in delaying releases of radionuclides from the unsaturated zone was relatively insignificant, even for moderately and strongly sorbing radionuclides in the northern part of the repository area where fracture-dominated transport prevails.

NRC Staff's Review

The NRC staff notes that in the scientific literature, there are considerable differences reported about the significance of fracture–matrix interactions (e.g., matrix diffusion) in unsaturated rocks. As part of its review, the NRC staff compared DOE's descriptions of large-scale tracer test results with published results of other modeling studies and unsaturated zone field studies at Yucca Mountain and elsewhere. Some transport studies identified fracture–matrix interactions as important (e.g., Zhou, et al., 2007aa; Liu, et al., 2004ab; Salve, et al., 2002aa; Hu, et al., 2001aa; Dahan, et al., 1999aa), and other studies did not (e.g., Pearcy, et al., 1995aa; Davidson, et al., 1998aa; Winterle and Murphy, 1999aa). Given the wide range of observations from various field studies in the scientific literature, the NRC staff concludes that DOE's use of a broad, uniform uncertainty distribution for the *gamma* parameter in the active fracture model is an acceptable treatment of model uncertainty for matrix diffusion in the unsaturated zone at Yucca Mountain.

The NRC staff reviewed and compared the transport data from large-scale field tests at Yucca Mountain and the DOE simulations of the test results. Based on its review, the NRC staff concludes that the DOE field observations and transport simulations support the DOE statements in Section 6.4.1 of BSC (2006aa) that DOE has not overestimated the effectiveness of matrix diffusion in delaying the migration of radionuclides through the unsaturated zone in TSPA calculations. Therefore, the NRC staff concludes that DOE's treatment of matrix diffusion model uncertainty and DOE's analyses to support the matrix diffusion model are adequate because DOE's modeling approach did not overestimate the effectiveness of matrix diffusion in retarding radionuclide transport.

In summary, the NRC staff evaluated the information DOE provided on (i) matrix diffusion in SAR Section 2.3.8, (ii) the hydrogeologic characteristics of the unsaturated zone at the Yucca Mountain site, and (iii) field and laboratory studies of fracture–matrix interactions in the unsaturated fractured rocks at Yucca Mountain and elsewhere, as detailed in NRC (2005aa, Section 5.1.3.7) and McMurry (2007aa). The NRC staff concludes that DOE has provided an adequate description and technical basis for matrix diffusion in the unsaturated zone transport abstraction for the following reasons:

- DOE coupled the two transport-related physical phenomena—advection in fractures and diffusion in the rock matrix—in an approach that was consistent with DOE's dual-permeability model framework for unsaturated zone flow and radionuclide transport.

- DOE's conceptual model for the unsaturated zone matrix diffusion acceptably addresses differences between matrix diffusion in the unsaturated zone compared to matrix diffusion in the saturated zone because DOE reasonably assumed that a comparatively smaller effective fracture–matrix interface area would be available for fracture–matrix interactions in unsaturated rocks.

- DOE calculated effective matrix diffusion coefficients using parameters measured using acceptable experiment procedures and site-specific rock samples.

- DOE adequately addressed data uncertainty in calculations of the effective matrix diffusion coefficients by including the natural variation of the rock properties in the range of parameter values.

- DOE addressed the effect of model uncertainty regarding matrix diffusion in the unsaturated zone transport abstraction by sampling a broad, uniform distribution of values for the active fracture model *gamma* parameter.

- DOE addressed model uncertainty about the extent and importance of fracture–matrix interactions by varying the size and extent of the fracture–matrix interface area available for matrix diffusion over a large range of potential values, as detailed in SNL (2008an, Section 6.6.4), and by simulating the uncertain effect of spatially and temporally variable flow conditions on transport rates in unsaturated fractures, as provided in SNL (2008ag, Section 6.3.9.2).

- DOE supported the assumptions and uncertainties of the matrix diffusion modeling approach by demonstrating, through sensitivity analyses and by comparison with large-scale transport tests, that radionuclide transport was less impeded by matrix diffusion in the unsaturated zone transport abstraction than would otherwise be expected in a natural system.

2.2.1.3.7.3.2.4 Colloid-Associated Transport

Colloids are minute solid particles of any origin or composition that are suspended in a liquid. Colloids can form by many processes in natural or engineered systems—for example, by physical or chemical degradation of preexisting solid materials or by precipitation from a solution—or they can be of biological or geological origin (e.g., microbes, clay minerals). Colloids influence radionuclide transport in the unsaturated zone because the transport path and transport rate of radionuclides associated with a colloid (e.g., radionuclides attached by sorption to the colloid surface) are determined by the transport behavior of the colloid instead of by processes that might otherwise affect the transport rate of the radionuclide as a dissolved species (e.g., matrix diffusion, or sorption in the rock matrix). Compared to dissolved radionuclides, colloids migrate preferentially in fractures, where travel times tend to be fast, because the small size of matrix pore openings inhibits the transfer of colloidal particles from fractures to matrix.

DOE's conceptual model in SNL (2008ag, Section 6.3.9.1) defined two modes of colloid-associated radionuclide transport: reversible colloids, in which radionuclides are temporarily (reversibly) attached to colloids by sorption, and irreversible colloids, in which radionuclides are assumed to be permanently attached to or embedded in the colloid. According to SNL (2007bi, Section 6.5.3), the effectiveness of radionuclide transport by colloids depends on (i) the transport characteristics of the colloids themselves; (ii) the concentration of colloids; and (iii) radionuclide sorption coefficients onto colloids and onto the immobile rock matrix; however, the overall effect of colloid-associated transport of reversible colloids is to facilitate the transport of radionuclides through the system. DOE addressed model uncertainty for colloid-associated transport in the unsaturated zone by applying a number of simplifying assumptions about colloid-associated transport processes. The assumptions resulted in fast and relatively unimpeded transport of radionuclides associated with colloids compared to slower modeled travel times for the same radionuclides transported as solutes.

Reversible Colloid Transport

DOE represented reversible colloid transport by modeling reversible sorption of dissolved radionuclides onto naturally occurring colloids in groundwater, using the same empirical K_d modeling approach that DOE used for reversible sorption in the rock matrix. DOE then applied an empirically determined colloid retardation factor, described in BSC (2004bc, Section 6.4.3), to account for colloid attachment and detachment processes in fractures that can hinder colloid movement in fractures. For simplicity, DOE assumed that all reversible colloids in the unsaturated zone are represented by the smectite clay mineral montmorillonite, a colloid-forming mineral in Yucca Mountain tuffs that has a high sorption capacity. DOE, in SNL (2007bi, Section 6.5.3), described how it modeled colloid-associated reversible sorption for six radioelements (americium, cesium, plutonium, protactinium, thorium, and tin) on the basis of their strong affinity for sorption onto montmorillonite under expected conditions in the unsaturated zone. DOE estimated the concentration of colloids in groundwater from data collected in saturated zone field studies from the Yucca Mountain area and from tabulated data for groundwater analyses elsewhere, as provided in SNL (2008an, Table 6-21). To address data uncertainty, DOE used the same estimated range of variability for groundwater colloid concentrations in the EBS, the unsaturated zone, and the saturated zone, but each transport abstraction sampled the range of values independently from the others in TSPA code simulations to account for the potential variability in groundwater colloid concentrations among the different environments, as identified in SNL (2008an, Section 6.5.12). DOE further addressed data uncertainty for reversible colloids by selecting ranges of montmorillonite sorption coefficients that emphasized large K_d values (i.e., strong sorption onto colloids), so as not to underestimate the effectiveness of radionuclide attachment to colloid surfaces, as described in DOE (2009am, Enclosure 14).

NRC Staff’s Review

The NRC staff reviewed DOE’s technical basis for the colloid-associated transport model in the context of the NRC staff’s independent understanding of colloid-associated transport modeling, colloid stability, and colloid transport properties in natural and engineered systems. As DOE noted in SAR Section 2.3.8.3, colloid transport mechanisms in unsaturated, fractured rocks are not well characterized by field studies. Accordingly, the NRC staff’s review of DOE’s technical basis for colloid-associated transport of radionuclides in the unsaturated zone focuses on how DOE addressed data and model uncertainty in developing parameter values and modeling colloid-associated transport processes. The NRC staff evaluated information DOE provided in SAR Section 2.3.8 and references therein, particularly SNL [2008ag, Section 7.7.1(a)], SNL (2007bi, Section 6.3.1), and SNL (2008an). The NRC staff reviewed the DOE integration of colloid-associated transport between the EBS, the unsaturated zone, and the saturated zone by examining SAR Section 2.3.7 (Waste Form Degradation and Mobilization and Engineered Barrier System Flow and Transport), SAR Section 2.3.9 (Saturated Zone Flow and Transport), and supporting references. The NRC staff also considered additional information that DOE provided to clarify details of the colloid-associated transport model in DOE (2009am, Enclosures 9 through 14). Based on its review, the NRC staff concludes that the data and methods DOE used to estimate unsaturated zone transport parameters for reversible colloids are acceptable for the following reasons:

- DOE compensated adequately for a scarcity of unsaturated zone colloid transport data by using data from saturated zone groundwater analyses and Yucca Mountain saturated zone colloid transport tests to estimate unsaturated zone colloid properties.

- By incorporating the available site-specific data to set initial and boundary conditions for colloid properties, DOE’s colloid-associated transport model adequately accounted for system variability and included sufficient data to describe colloids in the natural system.

- DOE addressed data uncertainty adequately by (i) sampling large ranges for colloid-associated parameter values to account for data uncertainty about natural colloid properties and (ii) sampling the ranges of parameter values separately for the unsaturated zone and saturated zone transport abstractions to account for data uncertainty and spatial heterogeneity in the natural system.

In evaluating DOE’s treatment of model uncertainty for reversible colloids, the NRC staff compared DOE’s selection of ranges of sorption coefficients for montmorillonite colloids, detailed in SNL (2008an, Table 6-22), with DOE’s selection of ranges of sorption coefficients for the unsaturated rock matrix, detailed in SNL (2007bj, Table 6-1[a]). The NRC staff’s comparison of values confirmed that for the radionuclides of interest, DOE’s sorption coefficients for the montmorillonite colloids promoted stronger sorption onto colloids than onto the rock matrix. Therefore, the NRC staff concludes that including reversible colloid-associated transport in DOE’s model is acceptable because it does not overestimate radionuclide travel times in the unsaturated zone.

Irreversible Colloid Transport

DOE’s colloid-associated transport model assumes that all irreversible colloids are generated within the EBS by the degradation of metals or wasteform materials, and the only radionuclides associated with irreversible colloids are isotopes of plutonium and americium (SAR Section 2.3.7.12.3.2; SNL 2007bi, Section 6.3.1). On the basis of field evidence for fast colloid transport in groundwater (e.g., Kersting, et al., 1999aa), DOE designated a small fraction (less than 0.2 percent) of the irreversible colloid flux as a “fast fraction” that is transported from the EBS to the accessible environment without any retardation. The rest of the irreversible colloid flux is subject to several potential colloid retardation processes, including (i) fracture-related colloid attachment and detachment processes, as DOE detailed in SNL (2008an, Section 6.5.13); (ii) the direct release of irreversible colloids from the EBS into the low permeability rock matrix beneath the repository drifts, as described in DOE (2009am, Enclosure 9); and (iii) the advective transfer of irreversible colloids laterally from fracture flow paths into the rock matrix, subject to flow field conditions (i.e., matrix permeability large enough to accommodate the advective flux) and subject to colloid size exclusions at the fracture–matrix interface, as described in SAR Section 2.3.8.4.5.4 and SNL (2008ag, Sections 6.3.9.1 and 6.3.9.2). In SNL (2008ag, Section 6.3.9.1), and in SNL (2008an, Section 6.5.9), DOE also described a fourth retardation process, the matrix filtration (straining) of irreversible colloids at the interface between the matrix of one rock unit and the matrix of the underlying rock unit, resulting in the permanent immobilization of irreversible colloids in the unsaturated zone. DOE compared unsaturated zone breakthrough curves for irreversible colloids with and without matrix filtration in SNL (2008an, ERD 02, Section III) and observed that including matrix filtration as a retardation process diminished the flux of irreversible colloids out of the unsaturated zone by as much as 80 percent in the southern half of the repository area, as illustrated in SNL [2008an, ERD 02, Figure 6.6.2-6(c)]. However, DOE’s final TSPA model did not implement matrix filtration in the TSPA simulations that DOE used to support compliance with 10 CFR 63.113, as explained in DOE (2009am, Enclosure 11).

NRC Staff’s Review

In evaluating DOE’s treatment of model uncertainty for irreversible colloids, the NRC staff examined results of DOE unsaturated zone TSPA calculations for plutonium and americium radionuclides transported as dissolved species and as irreversible colloids, as illustrated by SAR Figure 2.4-108 and DOE (2009am, Enclosure 10, Figures 3 and 4). In addition, the NRC staff examined DOE sensitivity analyses in SNL (2008an, ERD 02, Section III), where DOE examined how irreversible colloid travel times through the unsaturated zone differed if the calculations included the effect of colloid filtration in porous rock matrix. The NRC staff concludes that the data and methods DOE used to estimate unsaturated zone transport parameters for irreversible colloids is acceptable for the following reasons:

- DOE used simplifying assumptions that resulted in faster transport of irreversible colloids than solutes in the unsaturated zone.
- DOE took little or no credit in TSPA calculations for retardation processes such as the filtration of colloids at matrix–matrix interfaces that may significantly slow the transport of radionuclides associated with irreversible colloids compared to radionuclides transported as solutes.
- Although DOE specified that all irreversible colloids originated in the EBS, DOE approximated the generation and transport of additional irreversible colloids in the host rock by specifying that all naturally occurring, reversible colloids in the unsaturated zone were represented by montmorillonite, which has a strong sorption capacity for the six radioelements included in the colloid transport model.

In summary, on the basis of its review of colloid-associated radionuclide transport in the unsaturated zone, the NRC staff concludes that (i) DOE provided an adequate technical basis for the unsaturated zone colloid-associated transport model and (ii) DOE integrated the model with other components of the unsaturated zone transport abstraction for the following reasons:

- DOE incorporated important processes and features of colloid transport that were consistent with the physical setting at Yucca Mountain.

- DOE provided a conceptual model for colloid-associated transport of radionuclides that incorporated observable phenomena to distinguish colloids from solutes, such as colloid sizes, colloid sorption properties, and colloid transport behavior in fracture-dominated flow systems.

- DOE's conceptual treatment of reversible and irreversible colloid-associated transport in the unsaturated zone transport abstraction was consistent with DOE's conceptual model for reversible and irreversible colloids in the EBS and saturated zone transport abstractions.

- DOE adequately documented the conceptual and mathematical basis for the associated transport processes (e.g., retardation of colloids by attachment processes in fractures, reversible sorption of radionuclides onto colloids, colloid size exclusion processes at fracture–matrix interfaces, and unretarded colloidal transport), using an approach that was consistent with existing models for contaminant transport in fractured rocks in the literature (e.g., Sudicky and Frind, 1982aa).

- Because colloid transport properties in unsaturated, fractured rocks at Yucca Mountain and elsewhere were not well quantified by observations or experiments, DOE estimated unsaturated zone parameter values from site-specific saturated zone field and laboratory measurements. DOE addressed the large data uncertainty by sampling colloid parameter values probabilistically from large distribution ranges.

- To compensate for a scarcity of empirical observations of unsaturated zone colloidal transport in field experiments or natural analogs to support the model abstraction output, DOE's modeling approach addressed uncertainty by using a number of simplifying assumptions to take little or no credit for colloid retardation processes in the unsaturated zone (SAR Sections 2.3.8.3.4 and 2.3.8.2.2.3).

2.2.1.3.7.3.2.5 Radionuclide Decay and Ingrowth

Radioactive decay is a general term for the processes by which unstable radionuclides spontaneously disintegrate to form a different nuclide that may or may not also be radioactive. DOE's particle-tracking model in the unsaturated zone transport abstraction includes the loss of radionuclides over time due to radioactive decay and, where applicable, the model calculates the corresponding increase (ingrowth) of daughter radionuclides in decay chains, as described in SNL (2008an, Section 6.4.4). DOE assumed that upon radioactive decay of plutonium and americium in irreversible colloids, the decay chain daughters (e.g., uranium, neptunium) would be released from the irreversible colloid to migrate as dissolved species, with the exception of plutonium-239 produced by radioactive decay of americium-243, which DOE assumed would remain irreversibly attached to the colloid (SAR Section 2.3.8.2.2.3).

NRC Staff's Review

The NRC staff examined DOE's radionuclide transport analyses in SAR Sections 2.3.8 and 2.4.2 and confirmed that DOE's results corresponded with expected changes in the transported inventory due to radioactive decay and ingrowth. The NRC staff did not conduct a detailed technical review of DOE's model abstraction because DOE calculated radionuclide decay and ingrowth using a standard mathematic decay equation, with radioactive decay constants that are known with precision (International Union of Pure and Applied Chemistry, 1997aa). The NRC staff concludes that DOE's representation of radioactive decay and ingrowth is acceptable for the following reasons:

- DOE used a well-documented modeling approach with no significant uncertainties.

- DOE's model assumptions about the ingrowth-related transport behavior of decay chain radionuclides in irreversible colloids are consistent with DOE's model assumptions about the sorption behavior of the same radionuclides where they are associated with reversible colloids in DOE's model.

2.2.1.3.7.4 Evaluation Findings

The NRC staff reviewed the applicant's SAR and other information submitted in support of the license application, which includes information required by 10 CFR 63.21(c)(1), (9), (10), and (15), and finds, with reasonable expectation, that the relevant requirements of 10 CFR 63.114 and 63.342 are satisfied regarding the abstraction of radionuclide transport in the unsaturated zone. In particular, the NRC staff finds that DOE

- Included field data related to the geology, hydrology, and geochemistry of the surface and subsurface from the site and the region surrounding Yucca Mountain to define parameters and conceptual models used in the performance assessment calculation, in compliance with 10 CFR 63.114(a)(1)

- Accounted for uncertainty and variability in the parameter values used to model radionuclide transport in the unsaturated zone, in compliance with 10 CFR 63.114(a)(2)

- Considered and evaluated alternative conceptual models that are consistent with currently available data and scientific understanding, in compliance with 10 CFR 63.114(a)(3)

- Provided a technical basis for the inclusion of FEPs affecting radionuclide transport in the unsaturated zone, consistent with the limits on performance assessment in 10 CFR 63.342, and evaluated in sufficient detail those processes that would significantly affect repository performance, in compliance with 10 CFR 63.114(a)(4-6)

- Provided technical bases for the models of radionuclide transport in the unsaturated zone used in the performance assessment to represent the 10,000 years after disposal, in compliance with 10 CFR 63.114(a)(7)

- Used performance assessment methods for the period of geologic stability consistent with the methods used to demonstrate compliance for the initial 10,000 years following permanent closure, in compliance with 10 CFR 63.114(b)

- Included in the performance assessment for the period of geologic stability those FEPs used for the initial 10,000-year period, consistent with specific limits, in compliance with 10 CFR 63.342(c).

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CHAPTER 11

2.2.1.3.8 Flow Paths in the Saturated Zone

2.2.1.3.8.1 Introduction

Safety Evaluation Report (SER) Section 2.2.1.3.8 provides the U.S. Nuclear Regulatory Commission (NRC) staff's evaluation of the U.S. Department of Energy's ("DOE" or "applicant") representation of flow paths in the saturated zone within the context of the applicant's performance assessment evaluation. The NRC staff reviewed information provided in the DOE's Safety Analysis Report (SAR) included with the license application submitted on June 3, 2008 (DOE, 2008ab) and subsequent update of February 19, 2009 (DOE, 2009av), and information provided in response to requests for additional information (RAIs) (DOE, 2009an,bc).

Features and processes of groundwater flow in the saturated zone are included in DOE's performance assessment evaluation for the proposed geologic repository at Yucca Mountain, Nevada. The performance assessment analysis described in SAR Section 2.4.2.3.2.1 includes the flow of water (i) starting from precipitation falling on Yucca Mountain, (ii) in the unsaturated zone above and below the repository, and (iii) in the saturated zone through the controlled environment to the accessible environment. This groundwater is the principal means by which radionuclides released from the repository could be transported to the accessible environment (SAR Section 2.1). Exposure to extracted groundwater is one of the risk-significant pathways to the reasonably maximally exposed individual (RMEI).

DOE identified the saturated zone as a feature important to the capability of the lower natural barrier (SAR Section 2.1.1.3). Specifically, DOE indicated in DOE (2009an, Table 2.1-1 Expanded) that a combination of slow advective flow, long transport distance, and geochemical retardation of radionuclides in the saturated zone can substantially reduce the rate of radionuclide movement to the RMEI location. Saturated zone groundwater flow, as described in SAR Section 2.3.9.1, includes the features, events, and processes (FEPs) that affect the movement of groundwater in the saturated zone to the accessible environment and their implementation (or abstraction) in the Total System Performance Assessment (TSPA).

SER Section 2.2.1.1 provides the NRC staff's evaluation of DOE's identification and description of barriers and their capabilities as well as the consistency of these descriptions with the specific representations of these barriers in the TSPA and process-level models. The saturated zone flow abstraction receives information about the magnitude and patterns of groundwater flow downward through the unsaturated zone. The NRC staff evaluates unsaturated zone flow and transport in SER Sections 2.2.1.3.6 and 2.2.1.3.7. The saturated zone flow abstraction provides information about the direction, distance (flow paths), and amount (specific discharge) of groundwater flow to the saturated zone transport abstraction. The NRC staff's evaluation of those processes and characteristics most specific to radionuclide transport in the saturated zone is provided in SER Section 2.2.1.3.9. The applicant's analysis of the effect of future climate change on water flow in the saturated zone is evaluated in this section (SER Section 2.2.1.3.8) whereas the NRC staff's evaluation of the nature of future climate change is presented in SER Section 2.2.1.3.5.

2.2.1.3.8.2 Regulatory Requirements

The applicant is required to provide information specified in 10 CFR 63.21(c)(1), (9), (10), (15), and (19) that is related to the abstraction of flow paths in the saturated zone. The requirements in 10 CFR 63.114 (Requirements for Performance Assessment) and 10 CFR 63.342 (Limits on Performance Assessments) include postclosure performance assessment to demonstrate compliance with 10 CFR 63.113 (Performance Objectives for the Geologic Repository after Permanent Closure). Compliance with 10 CFR 63.113 is reviewed in SER Section 2.2.1.4.1.

The regulation at 10 CFR 63.114 requires, in part, that a performance assessment

- Include data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the surface and subsurface from the site and the region surrounding Yucca Mountain [10 CFR 63.114(a)(1)]

- Account for uncertainty and variability in the parameter values [10 CFR 63.114(a)(2)]

- Consider and evaluate alternative conceptual models [10 CFR 63.114(a)(3)]

- Provide technical bases for either the inclusion or exclusion of FEPs, including effects of degradation, deterioration, or alteration processes of engineered barriers that would adversely affect performance of the natural barriers, consistent with the limits on performance assessment in 10 CFR 63.342, and evaluate in sufficient detail those processes that would significantly affect repository performance [10 CFR 63.114(a)(4-6)]

- Provide technical basis for the models used in the performance assessment to represent the 10,000 years after disposal [10 CFR 63.114(a)(7)]

The NRC staff's evaluation of DOE's inclusion or exclusion of FEPs is presented in SER Section 2.2.1.2.1. Requirements for performance assessment for the initial 10,000 years following disposal are specified in 10 CFR 63.114(a). Requirements for the performance assessment methods for the time from 10,000 years through the period of geologic stability, defined in 10 CFR 63.302 as 1 million years following disposal, are specified in 10 CFR 63.114(b) and 10 CFR 63.342. These sections provide that through the period of geologic stability, with specific limitations, the applicant

- Use performance assessment methods consistent with the performance assessment methods used to demonstrate compliance for the initial 10,000 years after disposal [10 CFR 63.114(b)]

- Include in the performance assessment those FEPs used in the performance assessment for the initial 10,000-year period (10 CFR 63.342)

The requirements in 10 CFR 63.342(c)(2) pertain to the effects of climate change on performance for the period from 10,000 to 1 million years after disposal. In addition, the requirements in 10 CFR 63.342(c)(1) pertain to the effects of seismic and igneous activity on repository performance, subject to the probability limits in 10 CFR 63.342(a) and 10 CFR 63.342(b). Specific constraints on the analysis required for seismic and igneous activity are given in 10 CFR 63.342(c)(1)(i) and 10 CFR 63.342(c)(1)(ii).

The NRC staff's review of the SAR and supporting information follows the guidance in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa; Section 2.2.1.3.8, Flow Paths in the Saturated Zone), as supplemented by additional guidance for the period beginning 10,000 years after disposal (NRC, 2009ab). The YMRP acceptance criteria for model abstractions that provide guidance for the NRC staff's review of the applicant's abstraction of flow paths in the saturated zone are

1. System description and model integration are adequate
2. Data are sufficient for model justification
3. Data uncertainty is characterized and propagated through the abstraction
4. Model uncertainty is characterized and propagated through the abstraction
5. Model abstraction output is supported by objective comparisons

In its review of the SAR and supporting information, the NRC staff used a risk-informed approach and guidance provided by the YMRP, as supplemented by NRC (2009ab), for aspects of flow paths in the saturated zone that are important to repository performance. The NRC staff considered all five criteria provided in the YMRP in its review of information provided by DOE. In the context of these criteria, only those aspects of the model abstraction that substantively affect the performance assessment results, as determined by the NRC staff, are discussed in detail in this SER section. The NRC staff's determination is based both on risk information provided by DOE and on the NRC staff's independent analyses and knowledge gained through experience.

2.2.1.3.8.3 Technical Review

DOE analyzed the groundwater flow system in the vicinity of the proposed repository at Yucca Mountain to establish the direction, distance, and magnitude of water movement. DOE delineated the direction of and distance along water flow paths and computationally estimated the magnitude (specific discharge) of water flow using multiple groundwater flow models at different scales and degrees of simplification. Specific discharge, in turn, is used to determine the timing of radionuclide transport.

The objective of the NRC staff's technical evaluation in this section is to determine the acceptability of the applicant's delineated flow path directions and distances and the applicant's estimates of specific discharge for both present and future conditions. The information evaluated in this section is from SAR Section 2.3.9 and from relevant supporting documents that are cited when referred to in this section of the SER.

2.2.1.3.8.3.1 System Description and Integration of Models Relevant to Flow Paths in the Saturated Zone

DOE used multiple models at different scales to describe and quantify portions of the saturated zone groundwater flow system in the vicinity of the Yucca Mountain site. The site of the proposed repository at Yucca Mountain is within the Death Valley regional groundwater flow system located in the southern part of the Great Basin, which constitutes a subprovince of the larger Basin and Range physiographic province. DOE implemented models at the regional scale and Yucca Mountain site scale, and developed abstractions from those models for the performance assessment. In this subsection, the NRC staff reviews the system description, incorporation of features and processes into hydrogeologic and flow models, and integration of the saturated zone flow models at the regional and site scales. The NRC staff's review of the adequacy of saturated zone flow data, models, and abstraction is found in SER Sections 2.2.1.3.8.3.2 through 2.2.1.3.8.3.5.

DOE described included FEPs and the manner in which they are included in the saturated zone flow models in SAR Section 2.3.9. DOE described the capability of the saturated zone to function as a barrier to delay radionuclide migration by slow advective flow and/or long transport distance. The applicant identified a number of general characteristics and processes important to the function of the saturated zone barrier including stratigraphy, water-conducting features, faults, fractures, properties of host rock and other (alluvial) units, groundwater flow in the geosphere (magnitude and direction of groundwater flow), advection and dispersion, climate change, matrix diffusion, and sorption (SAR Sections 2.1.1.3 and 2.3.9.1). Additionally, the applicant stated that these types of characteristics and processes of the saturated zone have been included in saturated zone flow and transport models presented in SAR Section 2.3.9.

At the regional scale, the Death Valley groundwater system reflects the arid climatic conditions and the complex geology of Basin and Range flow systems (SAR Section 2.3.9.2.1). Groundwater in the regional system generally flows from recharge areas at high altitudes to the regional hydrologic sink in the bottom of Death Valley (SAR Section 2.3.9.2.1). The regional groundwater flow pattern is also conceptualized as a series of shallow and localized flow paths superposed on deeper regional flow paths (SAR Section 2.3.9.2.1).

In the Yucca Mountain region, groundwater flows generally from north to south, following these regional flow patterns. A relatively small amount of recharge occurs in the immediate vicinity of Yucca Mountain migrating downward through the unsaturated zone to the saturated zone. Once in the saturated zone, groundwater flows through a volcanic rock aquifer in the northern portions of the general flow system, transitioning into an alluvial aquifer system in the southern portions of the Yucca Mountain region. Beneath both the volcanic and alluvial aquifers is a carbonate rock aquifer (SAR Section 2.3.9.2.1).

DOE indicated that at the regional scale, a significant amount of groundwater flows through the relatively permeable, laterally continuous, and thick carbonate aquifer, below the volcanic and alluvial aquifers. On the basis of the regional geological framework and observations obtained from several drilled boreholes, the applicant concluded that vertical groundwater movement, to the extent it occurs, is upward rather than downward between the carbonate aquifer and the overlying volcanic and alluvial aquifers (SAR Section 2.3.9.2.2.4). Additionally, DOE modeling results support the direction of the gradient (SAR Section 2.3.9.2.3.2) at borehole locations where observations indicated a significant gradient was present. DOE conceptualized that the upward hydraulic gradient restricts groundwater flow paths originating from the proposed repository location to the shallower volcanic and alluvial aquifers, precluding radionuclides from entering the regional carbonate aquifer. The applicant also indicated that the upward gradient likely will be sustained during future climates and water uses (SAR Section 2.3.9.2.2.4). The NRC staff’s review of the gradient between the lower carbonate and the volcanic and alluvial aquifers is in SER Section 2.2.1.3.8.3.2.

At the site scale, DOE stated that groundwater flow occurs from the recharge areas in the north, through the Tertiary volcanic aquifers into the valley-fill aquifer, and continues south toward the compliance boundary. The applicant used various site-scale saturated zone flow and transport models to predict groundwater flow paths and calculate the transport of radionuclides from their introduction at the water table below the proposed repository to the accessible environment. The applicant summarized the interdependencies and information exchanges among these models (SAR Figure 2.3.9-1). The nominal case site-scale saturated zone flow model is conceptualized, and input parameters determined, on the basis of information derived from *in-situ* field tests; the U.S. Geological Survey Death Valley Regional Groundwater Flow System Model (DVRGFSM, which provides recharge and boundary conditions); the applicant's site-scale hydrogeologic framework model; the applicant's site-scale unsaturated zone flow model; and expert elicitation (SAR Section 2.3.9.1).

DOE's site-scale hydrogeologic framework model is a three-dimensional conceptual model of the spatial distribution of hydrogeologic units in the Yucca Mountain area. It covers an area of 1,350 km^2 [521 mi^2] and a thickness of about 6 km [3.7 mi] (SNL, 2007an). Direct input to the site-scale hydrogeologic framework model consists of (i) hydrogeologic information from both the DVRGFSM and the site-scale geologic framework model, which were generated from DOE's investigations; and (ii) lithostratigraphic interpretations and coordinates from the Nye County Early Warning Drilling Program (NC–EWDP) boreholes (SNL, 2007an). The NC–EWDP is a DOE-funded, Nye County-directed and implemented hydrogeologic investigation program. The applicant used information from this program to supplement its own investigations. Within the site-scale hydrogeologic framework model, DOE divided the Yucca Mountain geologic units into five basic saturated zone hydrogeologic units on the basis of similar hydrogeologic properties: upper volcanic aquifer, upper volcanic confining unit, lower volcanic aquifer, lower volcanic confining unit, and lower carbonate aquifer (SNL, 2007an). DOE stated that certain characteristics affecting flow, primarily the porosity and permeability of the hydrogeologic units, are highly variable. To represent discrete features and regions having distinct hydrological properties within the model domain, the applicant identified and incorporated 10 hydrogeologic features into the flow model to represent such features as fault zones, hydrologic flow barriers, and zones of enhanced permeability (SNL, 2007ax). The location at which groundwater flows from fractured volcanic rocks to alluvium is significant because of the differences in the hydrologic properties between these two rock units (SAR Section 2.3.9.2.1). The NRC staff evaluates DOE's approach to treat uncertainty associated with the tuff/alluvium contact in SER Section 2.2.1.3.8.3.3.

DOE's site-scale saturated zone flow model is a three-dimensional, finite-element numerical model that simulates groundwater flow in the area defined by the site-scale hydrogeologic framework model {i.e., 30 × 45 × 6 km [18.6 × 28.0 × 3.7 mi]}. DOE stated the flow model domain is sufficiently large to (i) assess groundwater flow and contaminant transport to the accessible environment, (ii) minimize boundary effects on flow magnitude and direction at Yucca Mountain, and (iii) include wells in the Amargosa Desert at the southern end of the modeled area (SAR Section 2.3.9.2.3.1). To predict flow magnitude and direction, the site-scale saturated zone flow model requires hydrogeologic information from the site-scale hydrogeologic framework model and boundary fluxes from the DVRGFSM. The NRC staff's evaluation of boundary conditions and physical attributes incorporated into the site-scale model is found in SER Section 2.2.1.3.8.3.2.

DOE stated that the sources of surface recharge in the immediate vicinity of Yucca Mountain are precipitation and flood flows from Fortymile Wash and its tributaries (SAR Section 2.3.9.2.1). The site-scale saturated zone flow model obtains surface recharge information from the applicant's site-scale unsaturated zone flow model over the area that lies directly below the site-scale unsaturated zone flow model domain. DOE used the 2004 version of the site-scale unsaturated zone flow model. However, DOE stated that an updated site-scale unsaturated zone flow model has been developed and is used in other parts of the TSPA (SAR Section 2.3.9.2.2.3). The NRC staff evaluates the impact of using an older version of the site-scale unsaturated zone flow model to estimate surface recharge over the area directly below the site-scale unsaturated zone flow model domain in SER Section 2.2.1.3.8.3.2.

DOE's three-dimensional site-scale saturated zone flow and transport abstraction model receives flow field information from the site-scale saturated zone flow model to generate 200 stochastic realizations of the flow field that reflect uncertainty in key parameters. The applicant prepared the input for each flow realization by scaling all permeability values in the site-scale saturated zone flow model using a scaling factor sampled stochastically from the probability distribution of a groundwater-specific discharge multiplier. For permeability values within the volcanic aquifer hydrogeologic units, DOE also sampled stochastically the horizontal anisotropy ratio (the ratio in the permeability in one horizontal principal direction relative to the permeability in a different principal direction, usually vertical). The NRC staff's review of data uncertainty for these and other parameters is found in SER Section 2.2.1.3.8.3.3. The steady-state groundwater flow solution for each realization was established by running the site-scale saturated zone flow model (SAR Section 2.3.9.3.4.1). After completing the 200 realizations using the site-scale saturated zone flow model, the resulting 200 flow fields were input to the site-scale saturated zone flow and transport abstraction model. These flow fields provided the TSPA model with 200 radionuclide unit mass breakthrough curves at the compliance boundary for 4 source subregions and 12 radionuclide groups, resulting in 9,600 breakthrough curves (SAR Figure 2.3.9-16). The NRC staff's review of uncertainty of flow fields is found in SER Section 2.2.1.3.8.3.4, and the breakthrough curves are evaluated in SER Section 2.2.1.3.9.

The one-dimensional saturated zone transport abstraction model, which provides the transport simulation capability for radionuclide daughter products resulting from decay and ingrowth, uses a simplistic one-dimensional representation of the three-dimensional saturated zone flows. The one-dimensional saturated zone transport abstraction model consists of three pipe segments. The first pipe segment is 5 km [3.1 mi] long. The lengths of the second and third pipe segments are estimated from particle tracking results of the three-dimensional saturated zone flow and transport abstraction model. The variable lengths account for uncertainty in the location of the volcanic/alluvial aquifer contact. Average, homogeneous material properties and specific discharges are specified within each pipe. The average specific discharge along each pipe segment is calculated by dividing the flow path length by the 50th percentile of particle travel times in BSC (2005ak, Section 6.5.1). Uncertainty in these parameters is reviewed by the NRC staff in SER Sections 2.2.1.3.8.3.3 and 2.2.1.3.8.3.4. DOE used specific discharge multipliers to reflect the four different climate states: present-day, monsoon, glacial-transition, and post-10,000-year. DOE used the same multiplier for the glacial-transition and post-10,000-year climate states (SNL, 2008ag, Table 6.3.10-3). DOE stated that the range of simulated average glacial-transition, and hence the post-10,000-year, infiltration from the infiltration model used to provide input to the saturated zone model is approximately consistent with the range of deep percolation values specified in 10 CFR 63.342(c)(2) for the million-year period (SNL, 2008ag, Section 6.3.10.2). The NRC staff finds that the small difference the DOE identified between the relevant recharge boundary condition in the saturated zone model and the distribution of deep percolation values specified in 10 CFR 63.342(c)(2) is acceptable because it would not affect the estimation of dose.

NRC Staff's Review

The NRC staff evaluated DOE's description of the saturated zone groundwater flow system in the vicinity of the proposed repository and the applicant's approach to integrate the multiple models used to quantify groundwater flow paths from the location of the proposed repository to the compliance boundary. The NRC staff finds the description, integration, and approach is acceptable because

- DOE provided sufficient information to describe the aspects of hydrology, geology, physical phenomena, and couplings that are relevant to the regional and site-scale saturated zone groundwater flow system. DOE incorporated the features, processes, and other information into appropriate models used for hydrogeology and saturated zone flow.

- DOE's treatment of FEPs during the initial 10,000 years following permanent closure in this abstraction continues unchanged through the period of geologic stability (defined as 1 million years in 10 CFR 63.302), and DOE incorporated the effect of climate change for the period of geologic stability estimates of saturated zone flow using recharge values consistent with the deep percolation values specified in 10 CFR 63.342(c)(2).

- The roles of models and the interdependencies between different models were clearly identified and illustrated (e.g., DVRGFSM, site-scale hydrogeologic framework model, site-scale saturated zone flow model, site-scale saturated zone flow and transport model abstraction, and one-dimensional saturated zone transport model abstraction).

- Conditions and assumptions in models of saturated zone flow were consistently identified through the abstraction process (i.e., consistent with other interrelated model abstractions) and were consistent with information provided in the SAR.

- Initial and boundary conditions used in the TSPA abstraction (i.e., site-scale saturated zone flow and transport model abstraction and one-dimensional saturated zone transport model abstraction) were consistent with the nominal case site-scale saturated zone flow model, which in turn was consistent with the DVRGFSM.

2.2.1.3.8.3.2 Sufficiency of Baseline Data to Justify Models of Flow Paths in the Saturated Zone

DOE used site-specific data to develop and corroborate the conceptual model of groundwater flow in the saturated zone and to calibrate the site-scale saturated zone flow model. The site-specific data used include water-level measurements, *in-situ* hydrologic and tracer testing conducted in the vicinity of Yucca Mountain, regional hydrogeologic model predictions, and parameters from expert elicitation. The NRC staff evaluated the sufficiency of DOE's baseline data used to develop predictions of flow paths and groundwater flow rates in the saturated zone in this section.

Hydraulic and Tracer Tests

The NRC staff reviewed the test methods and results of the hydraulic and tracer tests DOE conducted to corroborate its conceptualization of groundwater flow in the volcanic aquifers. These tests included several hydraulic and tracer tests (cross-hole tests) at the C-Wells Complex, consisting of boreholes UE-25 c#1, UE-25 c#2, and UE-25 c#3 (SAR Figure 2.3.9-7). DOE concluded that flow in the volcanic rock units mainly occurs through a well-connected fracture network and that large-scale horizontal anisotropy of aquifer permeability exists in the saturated zone, which is preferentially oriented in a north-northeast direction. The open-hole surveys done at the C-Wells Complex also yielded information on stratigraphy, lithology, matrix porosity, fracture density, and the major flowing intervals.

On the basis of the NRC staff's review of DOE's hydraulic and tracer tests, the NRC staff finds the applicant's hydraulic and tracer tests are appropriate to corroborate its conceptualization of groundwater flow in the volcanic aquifers because (i) the C-Wells Complex, on the flow path from the proposed repository to the accessible environment, represents an appropriate location for inferring *in-situ* volcanic aquifer properties; (ii) the applicant used widely accepted techniques in the scientific community for conducting these hydraulic and tracer tests and analyzing the observations; and (iii) the large-scale and cross-hole hydraulic tests yielded sufficient data to substantiate the applicability of the mathematical approach used with respect to representing site-scale groundwater flow in volcanic aquifers. DOE used the cross-hole well testing results to support its conclusions that (i) well-connected fracture networks exist in the volcanic aquifers and (ii) large-scale horizontal anisotropy of aquifer permeability exists in the volcanic aquifers, which is preferentially oriented in a north-northeast direction, consistent with the dominant fracture network observed at outcrops and in cores. The NRC staff finds that cross-hole testing is an appropriate approach to support the applicant's conclusions about well-connected fracture networks and anisotropy of aquifer permeability because cross-hole tests tend to sample a larger number of possible flow paths and, therefore, are appropriate for interpreting large-scale trends and effective permeability values. The NRC staff finds that the results the applicant obtained in these tests are consistent with its assumption of fracture-dominated flow in the volcanic aquifer system.

The NRC staff reviewed the results of the hydraulic and tracer tests DOE conducted to corroborate its conceptualization of groundwater flow in the alluvium. These tests included the hydraulic and tracer tests conducted at the Alluvial Testing Complex (centered at Nye County well NC–EWDP–19D), which is located along the simulated flow paths (SAR Section 2.3.9.2.4.2). DOE indicated that the saturated alluvium significantly reduces the movement of radionuclides to the accessible environment. The alluvial aquifer is generally conceptualized as a homogeneous hydrogeologic unit in the site-scale saturated zone flow model, except near the Fortymile Wash area. Testing results described in SNL (2007ax, Section 7.2.2.3) from the Alluvial Testing Complex and Nye County well 22S indicated that alluvium permeabilities vary over two orders of magnitude. DOE added a high-permeability zone—the Lower Fortymile Wash alluvial zone—to take into account "possible channelization" within the alluvium, as identified in SNL (2007ba, Section 6.4.3.7) and SAR Table 2.3.9-8. Similarly, DOE accounted for the effect of alluvium spatial heterogeneity on radionuclide transport by varying the effective porosity parameter in its performance assessment code. The NRC staff finds the data derived from alluvial hydraulic tests to be acceptable for the intended use because (i) the Alluvial Testing Complex is on the simulated flow path and underlying alluvial structure is representative of the Lower Fortymile Wash alluvium and (ii) techniques used to interpret hydraulic and tracer test data (for example, type curve fitting of pump test data) followed an approach that is widely used in the scientific community.

Water–Level Data

The NRC staff evaluated the sufficiency of water-level data DOE used to calibrate its site-scale saturated zone flow model. DOE used 161 time-averaged water-level measurements from 132 wells (multilevel measurements were obtained from some wells) within the model domain to (i) provide calibration targets for the site-scale saturated zone flow model, (ii) truncate the top of the flow model grid, and (iii) provide the boundary conditions around the perimeter of the model. DOE stated that water-level calibration targets represent steady-state values and reflect current water uses wherever pumping takes place (SNL, 2007ax). The NRC staff reviewed the coverage of water-level measurements that DOE collected and finds that the spatial and temporal coverage of water-level data is sufficient to calibrate the site-scale saturated zone flow model because (i) uncertainty was incorporated in the performance assessment, and (ii) additional locations and observations would not likely affect model predictions used as input for performance assessment.

Water Flow Between the Lower and Upper Aquifers

DOE used water-level data from 17 wells to determine whether an upward gradient exists from the lower volcanic aquifer to the upper volcanic aquifer within the modeled domain (SAR Table 2.3.9-6). DOE concluded that (i) a notable upward vertical gradient appears to exist between the lower and upper volcanic aquifer at locations nearest Yucca Mountain and (ii) the direction of the vertical hydraulic gradient varies from location to location away from Yucca Mountain (SNL, 2007ax).

The NRC staff evaluated the methodology and the sufficiency of data DOE used to establish the lower boundary conditions of the site-scale saturated zone flow model. DOE derived constant-head boundary conditions from water-level data. In SNL (2007ax, Section 6.3.1.5), DOE stated that coverage of water-level measurements was insufficient to specify depth-dependent head boundaries. In SAR Section 2.3.9.2.3.1, DOE indicated that vertical gradients develop internally in the model domain in response to hydrogeologic conditions and the calibrated model is capable of representing the upward vertical gradients observed between the deeper regional carbonate aquifer and overlying volcanic aquifers. DOE supplemented the SAR with a contour map of vertical hydraulic gradient distributions simulated by the site-scale saturated zone flow model, as shown in DOE (2009bc, Figure 1.1 of Enclosure 3). DOE stated that the directions of the vertical hydraulic gradient developed internally by the site-scale saturated zone flow model are consistent with the observations in the wells penetrating the lower carbonate aquifer. The NRC staff reviewed DOE (2009bc, Enclosure 3) and notes that DOE modeled an upward vertical gradient both at the locations of the wells that penetrate the carbonate aquifer (UE-25 p#1 and Nye County well NC–EWDP–2DB) and in the vicinity of Yucca Mountain.

The NRC staff finds that the DOE site-scale flow model adequately represents the observed upward vertical gradient because the simulated conditions are consistent with observations at the two wells. The upward vertical gradient refers to flow between the deeper regional carbonate aquifer and the overlying volcanic aquifers. Further, the NRC staff finds that the simulated distribution of vertical gradients, including the upward gradient in the flow path from the repository footprint to the accessible environment, are consistent with the principles of regional groundwater flow considering zones of recharge and configuration of hydrogeologic units.

During the NRC staff review, an inconsistency was noted in the assigned weights for two of the wells (UE-25p#1 and NC–EWDP–2DB) that DOE used during calibration to impose the lower boundary condition on the site-scale saturated zone flow model. In SAR Section 2.3.9.2.3.2, DOE stated that these wells, which show an upward vertical gradient, are assigned a weight factor of 10 during model calibration to ensure that the model will reflect the upward gradients. However, the NRC staff notes that these weight factors are inconsistent with weight factors listed in SNL (2007ax), the report referenced to support the information in the SAR. In Table 6.8 of SNL (2007ax), DOE indicated a weight factor of 20 assigned to UE-25 p#1 and a weight factor of 1 assigned to NC–EWDP–2DB. In response to an RAI issued by the NRC staff concerning this inconsistency, DOE confirmed that the actual weight factors used in the model calibration were 10 for both well locations (DOE, 2009bc, Enclosure 3). The NRC staff finds that the RAI response (DOE, 2009bc, Enclosure 3) acceptably addresses the inconsistency because it identified the appropriate weight factors used in the site scale flow model. DOE stated that it would clarify SAR Section 2.3.9.2.3.2 and make changes in the related supporting report (SNL, 2007ax) to reflect the actual weight factors used in model calibration.

Recharge Data

The NRC staff reviewed the sufficiency of recharge data used in DOE's site-scale saturated zone flow model. DOE used recharge derived from an older version of the site-scale unsaturated zone flow model. The NRC staff concludes that the impact of using an older version of the site-scale unsaturated zone flow model is small because the site-scale unsaturated zone component of recharge constitutes a small percentage (9 percent) of the total recharge within the domain of the site-scale saturated zone flow model. Surface recharge for other portions of the upper boundary within the domain of the site-scale saturated zone flow model was derived from the DVRGFSM model and measured stream losses along Fortymile Wash. The total surface recharge to the site-scale saturated zone flow model represents about 19 percent of the total inflow flux to the model domain. As a result, the recharge values from the unsaturated zone flow model are less than 2 percent of the total water budget for the model domain; therefore, the impact of uncertainty in the surface recharge from the site-scale unsaturated zone flow model to the site-scale saturated zone flow model is relatively small.

Vertical Anisotropy of Permeability

Vertical anisotropy of permeability is fixed at a horizontal-to-vertical ratio of 10:1 on the basis of information the expert elicitation panel provided (CRWMS M&O, 1998ac). The NRC staff finds that the horizontal-to-vertical anisotropy ratio of 10:1 for permeability used by DOE is in the range generally accepted by the scientific community for horizontally layered flow systems (e.g., Spitz and Moreno, 1996aa) like that in the Yucca Mountain region. The NRC staff's finding is also supported by experience gained from prelicensing interactions and activities (e.g., Sun, et al., 2008aa; Winterle and Farrell, 2002aa).

DOE considered the effect of vertical anisotropy on simulated flow paths in an alternative conceptual model. In SER Section 2.2.1.3.8.3.4, the NRC staff evaluates the model uncertainty associated with the uncertainty in vertical and horizontal anisotropy ratios.

Site-Scale Model Calibration

DOE calibrated the site-scale saturated zone flow model using an industry-standard parameter estimation program (PEST) followed by manual adjustments. During calibration, DOE appropriately assigned higher weights to observation wells located on potential flow paths to the compliance location. After calibrating the site-scale model using the parameter estimation program, manual adjustments were made to several zones to improve model match. The NRC staff finds the usage of PEST with subsequent manual adjustments acceptable because it is an approach widely used by the scientific community for model calibrations. The calibrated site-scale saturated zone flow model has a weighted root-mean-square residual of 0.82 m [2.7 ft] (calculated using differences between observed and simulated heads). SAR Figure 2.3.9-13 shows locations of all water-level measurements and calibration residuals (i.e., differences between simulated and observed water levels at the calibration target locations). The NRC staff finds acceptable the magnitude of the weighted parameter estimation program calibration residual for the scale of the model and the nature of the predictions made by the model (flow path direction and groundwater specific discharge) because the root-mean-square residual value of 0.82 m [2.7 ft] is small compared to the total water-level elevation variation of more than 300 m [986 ft].

The NRC staff notes that the purpose of model calibration is to provide parameter estimates for a given conceptual model and is not intended to resolve uncertainties in model conceptualization (e.g., uncertainty in stratigraphy). Thus, DOE considered different alternative conceptual models to address model uncertainties. The NRC staff evaluates DOE's treatment of model uncertainty and alternative conceptual models in SER Sections 2.2.1.3.8.3.4 and 2.2.1.3.8.3.5.

Summary of NRC Staff's Review

The NRC staff has evaluated the data the applicant used to develop and corroborate the conceptual model of groundwater flow in the saturated zone and to calibrate and validate the site-scale saturated zone flow model and finds the data are sufficient because

- DOE adequately summarized geological, hydrological, and geochemical data used to develop and implement models of saturated zone flow.

- DOE used appropriate techniques correctly to conduct relevant well testing.

- The description and justification of how the data were used, interpreted, and synthesized into model parameters were sufficient.

- Sufficient data were collected to establish initial and boundary conditions.

- Sufficient information was provided to substantiate that the site-scale saturated zone flow model is calibrated and applicable to site conditions.

2.2.1.3.8.3.3 Uncertainty in Data Used in Models of Flow Paths in the Saturated Zone

Uncertainties in model input parameters may directly affect the advective flow rate of groundwater and lengths of groundwater flow paths predicted by DOE's nominal case site-scale saturated zone flow model. In the performance assessment evaluation, DOE incorporated the uncertainty in model parameter inputs by stochastically sampling values from probability distributions of the groundwater-specific discharge multiplier, horizontal anisotropy in permeability, flowing interval spacing and fracture porosity in the volcanic units, effective porosity in the alluvium, and longitudinal dispersivity, as identified in SNL (2007ax, Section 6.3). This SER section focuses on reviewing DOE's methodologies for developing probability distributions of the specific discharge multiplier and horizontal anisotropy in permeability. The NRC staff evaluates the other uncertain model parameter inputs relevant to radionuclide transport calculations in SER Section 2.2.1.3.9.

Specific Discharge Values and Multiplier

To incorporate uncertainty in specific discharge in model abstractions, DOE generated multiple realizations of the three-dimensional saturated zone flow field (refer to SER Section 2.2.1.3.8.3.1 for additional discussion on the interdependencies of the different saturated zone abstraction models). For each realization, DOE scaled (i) the values of recharge and all values of permeability simultaneously using a stochastically sampled specific discharge multiplier and (ii) the values of north-south and east-west permeability within the zone of volcanic rocks using a stochastically sampled horizontal anisotropy ratio.

DOE established a probability distribution for the groundwater-specific discharge multiplier, whose function is to capture the range in variability and uncertainty in the parameters that generate the specific discharge calculations. In turn, the specific discharge calculations provide the basis from which groundwater travel times and radionuclide mass breakthrough curves are generated (SAR Section 2.3.9.2.3.3).

The uncertainty range and probability distribution for the groundwater-specific discharge multiplier was originally obtained through an expert elicitation process (CRWMS M&O, 1998ac). The expert elicitation panel suggested a truncated log-normal distribution ranging from 0.01 to 10. The median specific discharge derived from the expert elicitation process is 0.6 m/yr [2 ft/yr], as defined in CRWMS M&O (1998ac, Section 3.2). On the basis of tracer tests performed at the Alluvial Testing Complex and Nye County well cluster 22S, DOE reduced the range of uncertainty of the specific discharge multiplier using a Bayesian update procedure, where the range the expert elicitation panel supplied was assumed as a prior probability distribution and the estimated specific discharges from the Alluvial Testing Complex were used to estimate a log-normal likelihood function.

The NRC staff reviewed the Bayesian update procedure used by DOE. The NRC staff notes that the Bayesian updating method requires a dataset to be composed of independent and identically distributed random samples. In DOE (2009bc, Enclosure 1, Figure 1-1), the applicant provided additional information explaining the rationale for using the Bayesian statistical procedure, stating that (i) each combination of interpretation method and effective porosity value provides an independent and equally likely outcome and (ii) the 12 data values follow approximately a log-normal distribution. The NRC staff notes that DOE (2009bc, Enclosure 1) (i) did not present a goodness-of-fit statistical test to justify the log-normality of the 12 data values, although the small sample size might preclude the meaningfulness of such a statistical test and (ii) did not demonstrate the mutual independence of the estimation methods. Whereas log-normality and independence are assumptions in DOE’s implementation of the procedure that may not have been fully supported, the NRC staff determined that the uncertainty considered by DOE is adequate because any change would not significantly increase dose, noted as follows. The NRC staff finds that uncertainty in specific discharge is appropriately bounded and propagated in the applicant’s performance assessment because (i) for each estimation method, DOE obtained three specific discharge estimates by assuming the underlying unknown porosity equal to the maximum, median, and minimum porosity values of a porosity distribution and, therefore, likely bounded the estimation uncertainty related to the alluvium heterogeneity; (ii) the four estimation methods may have bounded the estimation uncertainty related to using each individual method alone; and most importantly, (iii) realizations in DOE’s performance assessment produce conservative transport times for nonsorbing solutes on the order of 10–100 years for the glacial-transition climate state, which does not underestimate the risk estimate (see also SER Section 2.2.1.3.8.3.4).

DOE used specific discharge estimates derived from alluvium testing to update the specific discharge multiplier that is subsequently applied to the entire flow model. DOE conceptualized fluid flow in volcanic tuff aquifers differently from that in the alluvium. The former is dominated by flow in well-connected fractures, while the latter is a porous medium. The input that DOE used as prior information in the Bayesian procedure was taken from the distribution the expert elicitation panel provided, which the panel derived from tests performed at the C-Wells Complex in volcanic aquifers. However, variability of specific discharge in volcanic aquifers is likely smaller than that in the alluvium because of the difference in flow paths and permeability in the two types of aquifers at the scale of testing and scale of numerical grid. Therefore, the NRC staff finds that applying the same specific discharge multiplier distribution to both volcanic and alluvial aquifers does not result in underestimation of the overall uncertainty.

As a final result, DOE obtained a truncated log-normal distribution for the specific discharge multiplier that ranges from 1/8.93 to 8.93 (BSC, 2005ak). DOE chose to update the specific discharge multiplier instead of the specific discharge itself so that the mean specific discharge would not change during the Bayesian updating procedure that incorporated data from Alluvial Testing Complex and Nye County well cluster 22S tracer testing (DOE, 2009bc, Enclosure 2). DOE (2009bc, Enclosure 2) stated that (i) variations in mean specific discharge along the flow paths continue to be captured in the baseline, three-dimensional site-scale flow model and (ii) updating the uncertainty distribution should be performed after normalization (i.e., use the specific discharge multiplier). The NRC staff notes that the normalization process alone would not change the fact that the specific discharge distribution in the alluvium is different from that in volcanic aquifers. Although DOE calibrated the three-dimensional site-scale flow model against water levels, and not specific discharges, the NRC staff finds that simulated specific discharges are consistent with *in-situ* estimates (see SER Section 2.2.1.3.8.3.5). In addition, using the updated specific discharge multiplier to address the glacial-transition climate state, DOE estimated conservative median transport times from the repository to the 18-km [11.2-mi] boundary for nonsorbing solutes on the order of 10–100 years (SAR Section 2.3.9.3.4.1; see also SER Section 2.2.1.3.8.3.4). The NRC staff finds that the predicted radionuclide travel times are not inappropriately reduced, because the approach used to update the specific discharge multiplier led to conservative median travel times. For travel times less than 100 years for nonsorbing radionuclides, the NRC staff notes that further reduction to median travel time will not significantly affect dose estimated in a performance assessment.

Horizontal Anisotropy

The NRC staff reviewed the probability distribution DOE established for the horizontal anisotropy of permeability. DOE stated in SNL (2007aw, Section 6.2.6) that hydraulic testing at the C-Wells Complex indicated significant flow anisotropy at larger scales in the fractured volcanic tuffs. In SAR Section 2.3.9.2.2.2.1, DOE indicated that the horizontal anisotropy ratio is estimated using different methods and the ratio ranges from 3.3 to 17, with directionality orienting flow paths more north-south than east-west. The cumulative distribution function for the horizontal anisotropy ratio, which has lower and upper bounds of 0.05 and 20, respectively, is specified through a tabulated form in the performance assessment model.

The maximum anisotropy ratio (i.e., 20) is greater than the highest value the NRC staff independently estimated on the basis of site-specific data (Ferrill, et al., 1999aa). On the basis of the information DOE presented and the NRC staff's independent estimate, the NRC staff finds acceptable the horizontal anisotropy ratio probability distribution because it reasonably represents the level of uncertainty associated with the permeability anisotropy at the site scale.

Potentially Undetected Fast Flow Paths

DOE assumed that the properties of all hydrogeologic units in the saturated zone site-scale flow model may be represented as homogeneous. Lithologic logs from the Nye County Early Warning Drilling Program (NC–EWDP) (Winterle and Farrell, 2002aa; Sun, et al., 2008aa) revealed significant spatial heterogeneity in the alluvium of Fortymile Wash. DOE stated (SAR Section 2.3.9.3.3.3.6) that the TSPA model and the range of uncertainty in the effective porosity encompass the behavior that would be obtained with an explicit model of a high-permeability channel in the alluvium. DOE also stated (BSC, 2005ak) that the range of uncertainty in effective porosity of the alluvium implicitly accounts for the potential existence of undetected stratigraphic and sedimentological features.

In an independent analysis, the NRC staff conceptualized the Fortymile Wash alluvium as a gravel-dominated deposit, having lower permeability zones interstratified with higher permeability deposits (Sun, et al., 2008aa). Without explicit modeling of the interstratified layers, the NRC staff approximated the behavior of the gravel-dominated deposit by deriving a dispersivity term that results in the arrival of fluid ahead of that traveling with the average velocity (Sun, et al., 2008aa).

On the basis of the NRC staff's review of (i) DOE's characterization and representation of parameter uncertainty in specific discharge and effective porosity through corresponding probability distribution functions, (ii) the travel times DOE's performance assessment code predicted, and (iii) the NRC staff's independent analysis, the NRC staff finds DOE's sampling from probability distributions of specific discharge multiplier and effective porosity adequately addressed the worst-case scenario resulting from potentially undetected fast-flow paths.

Volcanic and Alluvial Aquifer Contact Zone

DOE introduced an alluvium uncertainty zone in the TSPA model to treat uncertainty associated with the contact location between the volcanic aquifer and the alluvium. On the basis of drilling records, DOE conceptualized the uncertainty zone as a quadrilateral area in which the boundary between volcanic units and the alluvium is randomly varied among realizations. The boundaries of the alluvium uncertainty zone are determined for a particular realization by parameters representing the western boundary and northern boundary in DOE's one-dimensional saturated zone transport abstraction model. These parameters have uniform distributions from 0.0 to 1.0, where a value of 0.0 corresponds to the minimum extent of the uncertainty zone and 1.0 corresponds to the maximum extent of the uncertainty zone in a westerly direction and northerly direction, respectively (SNL, 2007ax). Therefore, in the one-dimensional transport abstraction model, the flow path length of each pipe segment varies as a function of the horizontal anisotropy, the western boundary of the alluvial uncertainty zone, and the region from which the radionuclide source originates beneath the repository. The NRC staff has previously reviewed well drilling records from various phases of the NC–EWDP (Winterle and Farrell, 2002aa; Sun, et al., 2008aa). On the basis of these reviews, the NRC staff finds that (i) DOE reasonably bounded the extents of the alluvium uncertainty zone and (ii) the uniform distributions defined for parameters representing western and northern extents of alluvium reasonably propagate uncertainties associated with the actual geometry of the volcanic and alluvium contact.

Use of Expert Elicitation

DOE used an expert elicitation process to obtain a probability distribution for the specific discharge multiplier (now considered prior distribution) and vertical anisotropy ratio. Based on the description provided in SER Section 2.3.9.2.2.6, the NRC staff found that DOE provided an adequate description of how an expert elicitation was used in the DOE development of data in saturated zone models. The NRC staff reviewed the information the expert elicitation panel provided. The NRC staff finds the panel's probability distributions for specific discharge multiplier and vertical anisotropy ratio acceptable on the basis of staff understanding of groundwater flow systems obtained from extensive pre-licensing experience and independent analyses (e.g., Ferrill, et al., 199aa; Sun, et al., 2008aa; Winterle and Farrell, 2002aa). An overall NRC staff evaluation of DOE's expert elicitation procedures is provided in SER Section 2.5.4.

Summary of NRC Staff's Review

The NRC staff finds the data uncertainty characterization and representation in DOE's site-scale saturated zone flow and abstraction models acceptable for the following reasons:

- DOE adopted parameter values, assumed ranges, probability distributions, and bounding assumptions that were technically defensible and reasonably account for uncertainties and variabilities.

- DOE reasonably incorporated the hydrologic effect (e.g., water table rise) of potential climate change, on the basis of a reasonably complete search of paleoclimate data, using scaling factors for different climate states.

- The uncertainties associated with included features and events pertaining to groundwater flow in the saturated zone were reasonably represented in the model abstractions.

- Results from the expert elicitation panel defining the specific discharge multiplier and vertical anisotropy ratio probability distributions are reasonable.

2.2.1.3.8.3.4 Uncertainty in Flow Paths in the Saturated Zone Models

In this section, the NRC staff evaluates the alternative conceptual models DOE used to assess model uncertainties for the saturated zone flow paths, as presented in SAR Section 2.3.9.2.3.4. DOE used five alternative conceptual models to assess the significance of model uncertainties of certain features and processes in the abstraction, including (i) vertical anisotropy, (ii) horizontal anisotropy, (iii) permeability in the northern high-gradient region of Yucca Mountain, (iv) increased vertical permeability of the Solitario Canyon Fault, and (v) climate-induced water table rise. NRC staff also reviewed additional model uncertainties considered by DOE that could affect specific discharge.

Vertical Anisotropy

The site-scale saturated zone flow model that DOE used to develop the performance assessment abstraction uses a 10:1 anisotropy ratio for horizontal-to-vertical permeability in volcanic and valley-fill alluvium units. This vertical anisotropy ratio was originally suggested from an expert elicitation panel (CRWMS M&O, 1998ac) because reduced vertical permeability is inherent in any layered groundwater flow system. To test whether this assumption leads to any systematic bias, DOE considered an alternative model with vertical permeability equal to the horizontal permeability. This alternative model resulted in a 28 percent increase in calculated specific discharge at a location 5 km [3.1 mi] downgradient from the proposed repository boundary and also resulted in a near doubling of the weighted root-mean-square calibration error. Because this alternative model results in a degraded calibration and the inclusion of vertical anisotropy is considered more representative of the layered system, the model with the 10:1 horizontal-to-vertical anisotropy ratio is the only one DOE used to develop the model abstraction. The NRC staff finds that DOE's analysis sufficiently demonstrated that it is not necessary to consider alternative models with different vertical anisotropy ratios, because the potential effect on specific discharge is not significant compared to the range of uncertainty already considered in DOE's performance assessment model and the deleterious effects on model calibration statistics.

Horizontal Anisotropy

The NRC staff reviewed the alternative model DOE used to demonstrate the sensitivity of the nominal case site-scale saturated zone model to horizontal anisotropy. DOE's analysis demonstrated that removal of horizontal anisotropy (i.e., assuming isotropic horizontal permeability) results in a 31 percent decrease in modeled specific discharge rates across the 5-km [3.1-mi] boundary and shifts the flow paths eastward. The effect on model calibration error is negligible. This analysis demonstrated that both isotropic and anisotropic cases are reasonably consistent with the observations in calibration wells. On the basis of this analysis, DOE included a range of horizontal anisotropy ratios for saturated zone flow and transport model abstraction. As discussed in SER Section 2.2.1.3.8.3.3, the NRC staff finds that the range of parameter uncertainty considered for horizontal anisotropy of permeability is appropriate.

Permeability in High-Gradient Region

The NRC staff reviewed an alternative modeling analysis in which DOE removed the large hydraulic gradient north of the proposed repository area by increasing permeability in this region. This alternative model results in a 15-fold increase in calculated specific discharge 5-km [3.1-mi] downgradient from the proposed repository and an 8-fold increase in root-mean-square calibration error. On the basis of this result, DOE concluded that, although the cause of the high gradient is not entirely certain, it is nevertheless important to represent this feature in the model for present-day conditions. The NRC staff notes that any modeling analysis that does not include low-permeability structural features north and west of the repository area would not be consistent with the measured high gradients nor with the water table position calibration points downgradient of the repository (as indicated by the large increase in the root-mean-square without the low-permeability feature). The NRC staff, therefore, finds that DOE appropriately included these features in the site-scale saturated zone flow model used to develop its performance assessment abstraction.

Permeability in Solitario Canyon Fault

The NRC staff evaluated DOE's alternative model used to examine the importance of the vertical permeability of the Solitario Canyon Fault. For the alternative model, DOE increased the vertical permeability of the Solitario Canyon Fault 100 fold, which demonstrated an insignificant effect on simulated water levels, flow paths, or specific discharge. As a result, DOE concluded that this alternative model would not be considered further for their site-scale saturated zone flow model and abstraction to the performance assessment model. Because the simulated water levels, flow paths, and specific discharge did not significantly change in this alternative model, the NRC staff finds that DOE appropriately excluded this alternative from consideration in the site-scale saturated zone flow model.

Climate-Induced Water Table Rise

The site-scale saturated zone flow model DOE used to develop the abstracted flow paths for the performance assessment does not consider explicitly the effect of an elevated water table under future, wetter climate conditions. Instead, DOE incorporated the effect of long-term climate change by applying a scaling factor to instantly increase the volumetric flow rate or specific discharge. The scaling factors for monsoonal and glacial-transition climatic conditions are 1.9 and 3.9, respectively. To demonstrate that this simplified approach for including effect of climate change does not bias results towards overestimates of barrier performance, DOE provided an alternative evaluation of the potential effects of water table rise on abstracted flow paths. On the basis of an estimated increase in specific discharge by a factor of 3.9 for the glacial-transition climate state, DOE estimated the increased hydraulic gradient necessary to drive this increased groundwater flow would result in a water table rise of approximately 20 m [66 ft] at the southern end of the model area that gradually increases to about 50 m [164 ft] in the area below the proposed repository location and as much as 100 m [328 ft] in areas north of the repository. Projecting this linear increase in water table elevation onto the hydrogeologic framework model indicates that elevated flow paths could travel a greater proportion of distance through the lower permeability Calico Hills formation in the volcanic tuffs. Because the travel time in low permeability units is slower than in higher permeability units, the NRC staff notes that neglecting water table rise in the saturated zone model would be conservative because it would result in shorter travel times compared to directly incorporating a water table rise. The NRC staff, therefore, finds that DOE's use of present-day water table elevations, combined with a scaling factor approach to increase specific discharge estimates for future climates, sufficiently approximates the performance-affecting aspects of future, wetter climate conditions. Further support for the NRC staff conclusion is provided by an independent analysis by Winterle (2005aa) that indicates an elevated water table would not significantly affect flow paths from beneath the proposed repository area.

Additional Model Uncertainties that May Affect Specific Discharge

In SAR Section 2.3.9.2.3.5, DOE provided qualitative consideration of several additional model uncertainties that could affect estimates of specific discharge. These considerations and the NRC staff's review are summarized in the following paragraphs.

The NRC staff evaluated DOE's treatment of uncertainty in the hydrogeologic contact surfaces (as in the hydrogeologic framework model) represented in the model. DOE explained that horizontal contact-surface uncertainty would have a lesser effect on specific discharge compared to uncertainty of contact surfaces in the vertical direction. DOE concluded that the potential effect of this model uncertainty is within the bounds of uncertainty considered for the specific discharge uncertainty multiplier parameter used in the performance assessment. The NRC staff finds this acceptable because having a flow path travel a longer horizontal distance in a particular unit will not significantly affect the flow rate, but if a permeable layer is vertically thicker or thinner than presumed in the model, the specific discharge rate would necessarily decrease or increase to accommodate the same volumetric flow.

The NRC staff evaluated DOE's consideration of model uncertainty related to the potential for a fault-dominated flow system with specific discharge focused in flow paths along major fault systems. This type of model conceptualization can produce rapid travel times by focusing high discharge rates into narrow zones (the fault systems) within the groundwater flow system. The NRC staff finds that this model uncertainty is appropriately addressed in the model abstraction by DOE's use of parameter ranges for effective porosity and specific discharge because the combination of very small values for effective porosity in tuff with high specific discharge rates introduces numerous realizations for the performance assessment that replicate the behavior of a fault-dominated flow and transport system. Numerous realizations in the performance assessment produce transport times for nonsorbing solutes on the order of 10 to 100 years for the glacial-transition climate state (SAR Section 2.3.9.3.4.1 and Figure 2.3.9-16). The NRC staff concludes that these realizations with rapid transport times reasonably represent the potential for focused high-permeability flow paths.

Summary of NRC Staff's Review

The NRC staff has evaluated the methods DOE used to characterize model uncertainty, and propagate the effects of this uncertainty, through the performance assessment abstraction and determines the methods are acceptable for the following reasons.

- DOE's performance assessment appropriately considered alternative conceptual models and modeling approaches to account for various uncertain FEPs.

- The modeling approach DOE used in the performance assessment is reasonably consistent with available data and current scientific understanding, and the results and limitations are appropriately considered in the abstraction.

- Conceptual model uncertainties were adequately defined and documented, and effects on conclusions regarding performance were properly assessed.

- Uncertainties in data interpretations for several aspects of the model were considered by analyzing reasonable alternative conceptual flow models that could not be ruled out by site data.

- Alternative modeling approaches DOE considered are consistent with available data and current scientific knowledge, and appropriately consider their results and limitations, using tests and analyses that are sensitive to the processes modeled.

2.2.1.3.8.3.5 Model Support Based on Comparison With Alternative Models or Other Information

In SAR Section 2.3.9.2.4, DOE presented its use of objective comparisons to build confidence in the saturated zone flow model abstraction. In this section, the NRC staff's review focuses on support for the range of flow paths and specific discharge estimates considered for the saturated zone flow and transport model abstraction on the basis of relative importance to overall system performance.

The NRC staff reviewed information DOE used to support model-simulated groundwater flow paths. This information includes a comparison of simulated water-level elevations to those observed in wells not used in the model calibration (e.g., NC–EWDP Phase V data) (SAR Table 2.3.9-9). This comparison shows that the largest differences between simulated and observed water levels generally occur in areas of steep hydraulic gradients near geologic features, such as the U.S. Highway 95 fault and the Solitario Canyon fault. Residual errors between observed and simulated water levels are generally smaller in the areas of simulated flow paths from the repository to the compliance boundary. The highest residual errors are generally in areas where water levels change by tens of meters (1 m = 3.28 ft) over short distances, and the overall range of residual errors is shown to be similar to the range of errors obtained during the site-scale saturated zone model calibration (SAR Figure 2.3.9-13). SNL (2007ax) described the calibration and confidence-building process. The NRC staff concludes DOE's comparison of modeled results to water-level measurements from wells not used in the model calibration demonstrates the model reasonably reproduces present-day water levels in the model areas important to determining flow and transport paths from the repository to the compliance boundary. Although an independent analysis by Winterle, et al. (2003aa) showed that such residual error could be significantly reduced by adjusting the shapes of modeled geologic features, doing so does not significantly change the modeled groundwater flow paths and specific discharge from the repository to the compliance boundary.

DOE discussed analyses by Freifeld, et al. (2006aa) of testing done in Nye County well 24PB that provides a range of estimates for specific discharge at the top of the Crater Flat tuff unit in the transition area from the volcanic aquifer to the valley-fill alluvial aquifer. On the basis of fluid electrical conductivity logging and distributed thermal perturbation sensor measurements, and assuming a porosity of 0.01, the estimated specific discharge in the flowing intervals ranged from 5–310 m/yr [16–1,017 ft/yr], as identified in Freifeld, et al. (2006aa, Section 3.3.3 and Table 4). The upper end of this range is significantly greater than the specific discharge rates considered in the performance assessment. DOE stated the high flow rate was observed in a relatively narrow interval of the borehole and that upscaling this estimate using an assumed median flow interval spacing of 25.8 m [84.6 ft] (from the parameter uncertainty distribution) reduces the estimated specific discharge to a range of 0.07–4.1 m/yr [0.2–13.5 ft/yr].

Freifeld, et al. (2006aa) proposed that "… additional data sets at other locations should be collected to examine whether the current data set is representative of the regional flow system near Yucca Mountain." The NRC staff concludes, however, the observation of high-flow zones spaced tens of meters (1 m = 3.28 ft) apart is consistent with the conceptual model of flow in the fractured tuffs being through a network of relatively widely spaced fracture zones. The identified zones of high transmissivity are relatively thin and, when averaged over the entire penetrated thickness of the Crater Flat tuffs, which is appropriate for comparison to the model grid scale, the specific discharge estimates are reasonably consistent with the upper end of the range of uncertainty considered for specific discharge in the abstraction. Additionally, because the high groundwater flows entered the well at a lower interval and exited through an interval more than 40 m [131 ft] higher, the wellbore itself could be the cause of the high flow rates by connecting two vertically distinct permeable zones with groundwater flow driven by an upward vertical hydraulic gradient. The calculated high rates of specific discharge for this well could be more a reflection of flow driven by a local upward gradient that is short-circuited by the borehole and not representative of horizontal flow rates along groundwater flow paths in the aquifer system. Therefore, the NRC staff finds that the results reported in Freifeld, et al. (2006aa) are not conclusive regarding horizontal specific discharge at the scale of interest to the model abstraction, but generally support the concept of widely spaced, flowing intervals in the volcanic tuff units.

NRC Staff's Conclusions on Model Support

The NRC staff has evaluated the approaches DOE used to compare performance assessment output to process-level model outputs and/or empirical studies and determines the approaches are acceptable for the following reasons:

- The models implemented in DOE's performance assessment abstraction for saturated zone flow provided results consistent with output from detailed process-level models and empirical observations from field tests.

- Outputs of flow paths in the saturated zone abstractions reasonably reproduced the results of corresponding process-level models and empirical observations.

- The procedures DOE used to construct and test the mathematical and numerical models used to simulate flow paths in the saturated zone were well documented in the SAR and in supporting references (SNL, 2007an,aw,ax, 2008ab).

- The site-scale saturated zone flow model was developed on the basis of an underlying geologic framework, calibrated to minimize error compared to observed water levels, and compared to results from other models and field testing data not used in the model development (procedures that reflect reasonable and generally accepted scientific practices).

- DOE provided several supporting analyses to demonstrate that the ranges of flow paths and specific discharge estimates used in the abstraction of flow paths in the saturated zone are reasonably consistent with site data and field tests.

- DOE models and results are consistent with alternative models developed by NRC staff that tested different aspects of saturated zone flow in the Yucca Mountain region, including flow in fractured volcanic tuffs, flow patterns from the repository to the accessible environment, and hydrologic characterization and flow in the alluvium (e.g., Ferrill, et al, 1999aa; Winterle, 2005aa; Winterle, et al., 2002aa; Sun, et al., 2008aa).

2.2.1.3.8.4 Evaluation Findings

The NRC staff has reviewed the SAR and other information submitted in support of the license application, which includes information required by 10 CFR 63.21(c)(1), (9), (10), (15), and (19), and finds, with reasonable expectation, that the relevant requirements of 10 CFR 63.114 and 10 CFR 63.342(c) are satisfied regarding the abstraction of flow paths in the saturated zone.

In particular, the NRC staff finds that DOE has adequately

- Included data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the surface and subsurface from the site and the region surrounding Yucca Mountain to define parameters and conceptual models used in the performance assessment calculation, in compliance with 10 CFR 63.114(a)(1)

- Accounted for uncertainty and variability in the parameter values used to model flow paths in the saturated zone, in compliance with 10 CFR 63.114(a)(2)

- Considered and evaluated alternative conceptual models that are consistent with currently available data and scientific understanding, in compliance with 10 CFR 63.114(a)(3)

- Provided a technical basis for the inclusion of FEPs affecting flow paths in the saturated zone, consistent with the limits on performance assessment in 10 CFR 63.342, and evaluated in sufficient detail those processes that would significantly affect repository performance, in compliance with 10 CFR 63.114(a)(4-6)

- Provided technical bases for the models of flow paths in the saturated zone used in the performance assessment to represent the 10,000 years after disposal, in compliance with 10 CFR 63.114(a)(7)

- Used performance assessment methods for the period of geologic stability consistent with the methods used to demonstrate compliance for the initial 10,000 years after disposal, in compliance with 10 CFR 63.114(b)

- Included in the performance assessment for the period of geologic stability those FEPs used for the initial 10,000-year period, consistent with specific limits, in compliance with 10 CFR 63.342(c)

2.2.1.3.8.5 References

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CHAPTER 12

2.2.1.3.9 Radionuclide Transport in the Saturated Zone

2.2.1.3.9.1 Introduction

Safety Evaluation Report (SER) Section 2.2.1.3.9 provides the U.S. Nuclear Regulatory Commission (NRC) staff's evaluation of the U.S. Department of Energy's ("DOE" or the "applicant") model abstraction for transport of radionuclides in the saturated zone. DOE presented its description of this abstraction in Safety Analysis Report (SAR) Section 2.3.9 of its license application (DOE, 2008ab). DOE's Total System Performance Assessment (TSPA) includes the flow of water from precipitation on Yucca Mountain, its migration as groundwater through the unsaturated zone above and below the repository, and the flow of groundwater in the saturated zone through the controlled area to the accessible environment boundary. This groundwater flow is the principal means by which radionuclides released from the repository are transported to the accessible environment. Because exposure to groundwater contaminated with radionuclides from the repository is one of the principal contributors to dose to the reasonably maximally exposed individual (RMEI), the performance assessment must include the components that affect significantly the timing and magnitude of transport for any radionuclides released from the repository. Radionuclide transport in the saturated zone, as described in SAR Section 2.3.9, includes the features, events, and processes (FEPs) that affect the movement of radionuclides from where they enter the saturated zone below the repository to the accessible environment boundary approximately 18 km [11.18 mi] south of the repository and their implementation (or abstraction) in the TSPA.

The NRC staff's evaluation focuses on the following processes, detailed in subsequent sections, that DOE identified in SAR Section 2.3.9 as important to radionuclide transport in the saturated zone: (i) advection, because most of the radionuclide mass is carried through the saturated zone by water flowing toward the accessible environment; (ii) sorption, because sorption in the saturated volcanic rocks and alluvium has the largest overall effect on slowing radionuclide transport in the saturated zone; (iii) matrix diffusion in volcanic rock, because matrix diffusion coupled with sorption slows radionuclide transport in the saturated zone near the repository area; (iv) colloid-associated transport, because radionuclides attached to colloids may travel faster through the saturated zone than would otherwise be expected; and (v) radioactive decay and ingrowth, because these processes affect the quantities of radionuclides released from the saturated zone over time. The NRC staff's review of DOE's technical basis for excluding other FEPs is addressed in SER Section 2.2.1.2.1 (Scenario Analysis).

2.2.1.3.9.2 Regulatory Requirements

The applicant is required to provide information specified in 10 CFR 63.21(c)(1), (9), and (15) that is related to the abstraction of radionuclide transport in the saturated zone. The requirements in 10 CFR 63.114 (Requirements for Performance Assessment) and 10 CFR 63.342 (Limits on Performance Assessments) include postclosure performance assessment to demonstrate compliance with 10 CFR 63.113 (Performance Objectives for the Geologic Repository after Permanent Closure). Compliance with 10 CFR 63.113 is reviewed in SER Section 2.2.1.4.1.

The regulation at 10 CFR 63.114 requires, in part, that a performance assessment

- Include data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the surface and subsurface from the site and the region surrounding Yucca Mountain [10 CFR 63.114(a)(1)]

- Account for uncertainty and variability in the parameter values [10 CFR 63.114(a)(2)]

- Consider and evaluate alternative conceptual models [10 CFR 63.114(a)(3)]

- Provide technical bases for either the inclusion or exclusion of FEPs, including effects of degradation, deterioration, or alteration processes of engineered barriers that would adversely affect performance of the natural barriers, consistent with the limits on performance assessment in 10 CFR 63.342, and evaluate in sufficient detail those processes that would significantly affect repository performance [10 CFR 63.114(a)(4-6)]

- Provide a technical basis for the models used in the performance assessment to represent the 10,000 years after disposal [10 CFR 63.114(a)(7)]

The NRC staff's evaluation of DOE's inclusion or exclusion of FEPs is in SER Section 2.2.1.2.1. Requirements for performance assessment for the initial 10,000 years following disposal are specified in 10 CFR 63.114(a). Requirements for the performance assessment methods for the time from 10,000 years through the period of geologic stability, defined in 10 CFR 63.302 as 1 million years following disposal, are in 10 CFR 63.114(b) and 10 CFR 63.342. These sections provide that through the period of geologic stability, with specific limitations, the applicant

- Use performance assessment methods consistent with the performance assessment methods used to demonstrate compliance for the initial 10,000 years after disposal [10 CFR 63.114(b)]

- Include in the performance assessment those FEPs used in the performance assessment for the initial 10,000-year period [10 CFR 63.342]

The NRC staff's review of the SAR and supporting information follows the guidance in the Yucca Mountain Review Plan (YMRP), NUREG–1804 (NRC, 2003aa), Section 2.2.1.3.9, Radionuclide Transport in the Saturated Zone, as supplemented by additional guidance for the period beyond 10,000 years after permanent closure (NRC, 2009ab). The YMRP acceptance criteria for model abstractions that provide guidance for the NRC staff's review of the applicant's abstraction of radionuclide transport in the saturated zone are

1. System description and model integration are adequate
2. Data are sufficient for model justification
3. Data uncertainty is characterized and propagated through the abstraction
4. Model uncertainty is characterized and propagated through the abstraction
5. Model abstraction output is supported by objective comparisons

In its review of the SAR and supporting information, the NRC staff used a risk-informed approach and guidance provided by the YMRP, as supplemented by NRC (2009ab), for aspects of radionuclide transport in the saturated zone important to repository performance. The NRC staff considered all five YMRP acceptance criteria in its review of information provided by DOE.

In the context of these criteria, only those aspects of the model abstraction that substantively affect the performance assessment results, as determined by the NRC staff, are discussed in detail in this SER Section. The NRC staff's determination is based both on risk information provided by DOE and on the NRC staff's knowledge gained through experience and independent analyses.

2.2.1.3.9.3 Technical Review

The NRC staff reviewed information in SAR Section 2.3.9 and references therein that described how DOE predicted the transport of radionuclides in the saturated zone from below the repository to the accessible environment. The NRC staff's technical review focuses on how DOE (i) developed a system description that incorporated site-specific geological, hydrological, and geochemical features of Yucca Mountain in the saturated zone radionuclide transport abstraction, including how the transport abstraction was integrated with other TSPA model components (SER Section 2.2.1.3.9.3.1), and (ii) established the technical bases for modeling the major risk-significant processes related to radionuclide transport in DOE's process-level models and in the saturated zone radionuclide transport abstraction (SER Section 2.2.1.3.9.3.2). The major processes considered by DOE were advection and dispersion, sorption, matrix diffusion, colloid-associated transport, and radionuclide decay and ingrowth.

In conducting the technical review, the NRC staff noted that DOE's saturated zone transport model potentially underestimates certain radionuclide releases from the saturated zone, particularly in the context of DOE's assumption that long-lived members of radioactive decay chains are in secular equilibrium in the saturated zone. The NRC staff reviews DOE's treatment of radioactive decay and ingrowth processes in SER Section 2.2.3.9.3.2.5, but some background information about secular equilibrium is provided here as context for the NRC staff's review of how this assumption affects other parts of the saturated zone transport model. In a closed system, radioactive decay chains reach secular equilibrium, a condition where all component radionuclides have equal activity. In this context, *activity* is a measurement of the radioactivity of a substance, in terms of the number of atoms in a sample that disintegrate (decay) per unit of time. Deviations from secular equilibrium, as either excess or deficient activity of a daughter relative to a parent, can develop in open or dynamic systems, particularly where parents and daughters have different chemical behavior. For example, disequilibria between parent and daughter may develop along a transport path if a long-lived parent is more strongly sorbed than its decay products. Given sufficient differences in chemical behavior, such disequilibria can manifest over time in groundwater transport systems even where fluxes of long-lived parent radionuclides are in steady state. For the Yucca Mountain saturated-zone transport system, where the transport path is long and geologically varied and the modeled release of contaminants is slow and limited, this effect would be most apparent over long performance periods, on the order of hundreds of thousands of years or more. Accordingly, the NRC staff's review includes a consideration of DOE's technical basis for assumptions of secular equilibrium, in the context of saturated zone radionuclide transport processes and other simplifying assumptions implemented by DOE in the saturated zone transport abstraction.

2.2.1.3.9.3.1 System Description and Model Framework

This section provides the NRC staff's review of DOE's overall system description as it relates to a conceptual model for radionuclide transport in the saturated zone. This section also provides the NRC staff's review of the model framework developed by DOE for the integration of radionuclide transport in the saturated zone as an abstraction in DOE's performance assessment model. In the TSPA, the saturated zone radionuclide transport abstraction receives information from the unsaturated zone transport abstraction about the time-dependent flux of radionuclides that are released from the unsaturated zone at the water table below the repository. The saturated zone radionuclide transport abstraction provides information to the biosphere model about the time-dependent flux of radionuclides and their decay chain daughters at the accessible environment boundary. The biosphere model then uses this information as input to calculate the annual dose to the reasonably maximally exposed individual (RMEI).

System Description

DOE related Yucca Mountain site characteristics to a conceptual model of the saturated zone extending from beneath the repository to the accessible environment boundary in which the flow of water would transport radionuclides through two primary geological units (fractured volcanic tuff and alluvium) and through major faults. In DOE's model, the disparate geological properties of the fractured volcanic rock and the alluvium are expected to have very different effects on water flow and radionuclide transport. Radionuclide transport through the fractured volcanic rock is generally fast because the rock has low porosity (void space), and the sparse distribution of flowing fractures channelizes the flow of water and limits opportunity for radionuclide interactions with the rock that could slow radionuclide transport (SAR Section 2.3.9.2.1). In contrast to DOE's unsaturated zone radionuclide transport model (SAR Section 2.3.8), DOE's saturated zone radionuclide transport model does not include flow (advection) in the pore spaces between mineral grains in the volcanic rock matrix because flow conditions in the rock matrix are essentially stagnant compared to the much higher flow velocities in the fractures. The conceptual model does assume, however, that the water in pore spaces in the rock matrix is connected with the water in the fractures, so diffusion of solutes between matrix and fractures (i.e., matrix diffusion) is included in the transport modeling.

DOE modeled the alluvium as a porous medium that has significantly higher porosity than the volcanic rock. Consequently, flow velocities in the alluvium are lower than in the fractured rock. DOE recognized that because of the way in which the alluvium was deposited, some preferential flow paths could exist in parts of the alluvium such as in buried gravel deposits. DOE accounted for these potential higher velocity flow paths by including effective porosity as a sampled parameter, with a range of values to accommodate uncertainty about these features in the model abstraction (BSC, 2005ak). Consistent with results from field-based transport testing (SNL, 2007aw), DOE did not take credit for matrix diffusion in the alluvium as a potential retardation process that might result from the dual porosity aspect of the alluvial system.

On the basis of field and modeling studies, DOE determined that for modeling purposes, groundwater flow and migration of radionuclides in the saturated zone would begin in fractured volcanic rock beneath the repository and would extend southeasterly toward Fortymile Wash before turning in a southerly direction beneath the wash, continuing from there towards the accessible environment boundary located approximately 18 km [11.18 mi] south of the repository. The subsurface contact between volcanic rock and the alluvium along this path occurs approximately 10 km [6.21 mi] south of the repository, at which point the water and radionuclides are expected to pass out of the volcanic rock and into the porous alluvium. The specific location of the contact between volcanic rock and alluvium along the flow path is a key geologic data uncertainty in DOE's transport abstraction (BSC, 2005ak). DOE addressed this uncertainty by designating the transition as the alluvium uncertainty zone, a sampled value which DOE constrained with subsurface geologic data from Nye County Early Warning Drilling Program (EWDP) wells.

DOE simulated the transport of radionuclides in the saturated zone as dissolved species and as species sorbed to mobile, colloid-sized particles. These two modes of transport are subject to various physical and chemical processes that affect their transport in groundwater. DOE identified advection and dispersion, sorption, matrix diffusion, colloid-associated transport, and radionuclide decay and ingrowth as important transport-affecting processes and incorporated these processes in the numerical models of radionuclide transport in the saturated zone (SAR Table 2.3.9-1). DOE's conceptual model described how each of the transport-affecting processes influences the rate at which radionuclides travel through the saturated zone model relative to the rate that water travels (SAR Section 2.3.9.2). DOE used sensitivity analyses and single-realization analyses of TSPA simulations to demonstrate how the saturated zone transport abstraction integrated the specific transport-related processes with the natural features of the saturated zone to slow the migration of radionuclides through the saturated zone, as detailed in SAR Section 2.1.2.3.6 and SNL (2008ag, Section 6.3.10).

DOE recognized that changes in future climate could affect radionuclide transport at Yucca Mountain by raising or lowering the elevation of the water table and by either increasing or decreasing the percolation rate. DOE noted, for example, that the saturated flow paths associated with an elevated water table could pass through volcanic rocks that contain a greater percentage of zeolites, which would tend to slow water movement and radionuclide transport (DOE, 2009de). DOE considered three climate states in modeling the initial 10,000-year performance period: (i) current, (ii) monsoonal, and (iii) glacial transition conditions. For evaluating the longer term repository performance after 10,000 years following repository closure, DOE used a “constant-in-time” climate with a prescribed deep percolation rate, as defined in 10 CFR 63.342(c)(2).

In the unsaturated zone transport model abstraction, DOE simulated the effects of climate change on the elevation of the water table by (i) shortening the unsaturated zone flow path length for transport calculations during wetter climates and (ii) instantaneously releasing to the saturated zone any radionuclides that were present in the portion of the unsaturated zone that became inundated at the time of the water table rise. In the saturated zone model abstraction, DOE addressed the effect of wetter climates on radionuclide transport by using specific discharge multipliers to simply increase the radionuclide flux for each future climate state, so that a larger mass reaches the accessible environment boundary. DOE compared the simplified modeling approach with a more detailed consideration of the effect of changes in water table elevations and flow rates in the saturated zone by using a three-dimensional site-scale transport model to generate particle tracks for the wetter climate states (SNL, 2007ba, Appendix E; DOE, 2009de). DOE stated the particle tracking results demonstrated that exclusion of these effects in the saturated zone transport model for TSPA did not result in an underestimation of dose, because the path lengths and travel times of radionuclides increased relative to the simplified use of specific discharge multipliers for radionuclide flux that DOE used in the performance assessment (SNL, 2007ba, Appendix E). TSPA results (SAR Figure 2.4-20) indicated that radionuclide release to the environment in the 10,000-year performance period will be dominated by unretarded radionuclides. The TSPA results also indicated that other radionuclides, with transport characteristics that are more sensitive to changes in chemical conditions, become more important at longer times.

NRC Staff's Review

The NRC staff compared DOE's conceptual model and system description of radionuclide transport in the saturated zone in SAR Section 2.3.9 and references therein with the NRC staff's understanding of the Yucca Mountain natural system, obtained from prelicensing field observations and independent analyses of saturated zone transport processes, as identified in NRC (2005aa) and Leslie, et al. (2007aa). The NRC staff concludes that DOE has provided an acceptable system description for radionuclide transport in the saturated zone for the following reasons:

- DOE's conceptual model includes FEPs that are reasonably expected to affect radionuclide transport in the saturated zone over the period of geological stability as defined in 10 CFR 63.302. DOE provided an adequate technical basis for the inclusion of these FEPs in the conceptual model because DOE used appropriately designed laboratory tests, field tests, and natural analog data to demonstrate how the FEPs would affect transport at Yucca Mountain.

- DOE identified and included important Yucca Mountain site characteristics that are expected to affect radionuclide transport, such as flow in the fractures of the volcanic tuff, the porous nature and uncertain extent of the alluvium, uncertainties in groundwater flow rates, and the expected range of site groundwater chemistry.

- DOE used Yucca Mountain site characterization data to assign geologic, hydrologic, and radionuclide transport parameter values to specific rock units or to define ranges of values for these properties to address uncertainty about the natural variability of the system.

- DOE incorporated results of laboratory tests, field tests, and natural analog studies, using site-specific materials and data, in developing the system description and process-level models.

- DOE provided reasonable technical bases for the inclusion of processes affecting radionuclide transport in the saturated zone and conducted sensitivity tests and specific modeling evaluations to verify appropriate incorporation of these processes into their system models.

Model Framework

For the TSPA calculations, DOE's saturated zone radionuclide transport abstraction simulated the transport of dissolved radionuclides and colloid-associated radionuclides through the saturated zone from beneath the repository, generating breakthrough curves at the accessible environment boundary for radionuclide species that DOE determined were risk significant. DOE integrated the saturated zone transport abstraction with three other model components in the TSPA: the site-scale saturated zone flow model (SAR Section 2.3.9), the unsaturated zone transport abstraction (SAR Section 2.3.8), and the biosphere model (SAR Section 2.3.10).

In DOE's unsaturated zone transport abstraction (SAR Section 2.3.8), radionuclides released from waste packages migrated through the fractures and rock matrix at rates affected by flow fields generated from the site-scale unsaturated zone flow model. The modeled boundary through which radionuclides from the unsaturated zone passed to the saturated zone was divided into four regions (or subareas). Depending on the conditions modeled, radionuclides could be released from as few as one or as many as all four of the subareas (SNL, 2008ag, Figure 6.3.10-6). For modeling purposes, DOE assumed that all radionuclides released from a subarea would enter the saturated zone at a single point (point source) in the subarea. During the transport simulations, DOE obtained the locations of the point sources by randomly selecting a point within each of the four subareas that represented a preferential flow pathway in the unsaturated zone flow model. The model collected, at the point source within the subarea, all of the unsaturated zone releases of radionuclides that occurred within any portion of that subarea and conveyed them to a single point in the saturated zone in the same subarea. The saturated zone transport model assumed that all radionuclide mass released from the unsaturated zone was transferred at the saturated zone point source into flowing fractures.

DOE implemented two model abstractions of saturated zone transport in the TSPA. One was a three-dimensional transport model, which was integrated with the site-scale saturated zone flow model by sharing the same three-dimensional, single (effective)-continuum, dual-porosity, particle-tracking transport model grid (SAR Section 2.3.9.2, SER Section 2.2.1.3.8). The dual-porosity aspect of the model refers to the porosities of the fractures and rock matrix; this allowed the model to consider matrix diffusion in fractured volcanic tuff. The effective continuum aspect of the modeling approach allowed DOE to assign average values to flow and transport parameters applied to cells of the numerical model representing the system. The saturated zone radionuclide transport model abstraction used the flow fields and other hydrologic characteristics defined by the three-dimensional saturated zone flow model and calculated unit breakthrough curves for each of the 12 radionuclide groups transported in the saturated zone. The three-dimensional model was run, and the unit breakthrough curves were developed and stored external to the TSPA model, as outlined in SNL (2008ag, Section 6.3.10).

DOE's second saturated zone radionuclide transport model abstraction was a one-dimensional transport model. The main purpose of the one-dimensional model was to calculate the radioactive decay, ingrowth, and transport for second-generation daughter radionuclides for four decay chains—the actinium, uranium, thorium, and neptunium series (SAR Figure 2.4-20). The one-dimensional transport model was implemented as four groups of GoldSim® (GoldSim Technology Group, 2006aa) pipe elements. One group was used for each of the four repository source regions. Each group of pipe elements consisted of three segments, representing the volcanic tuffs (the first two segments) and the alluvium (the last segment). DOE considered the lengths of the last two segments uncertain, consistent with the uncertain transition zone from saturated volcanic tuff to alluvium used in the three-dimensional model. DOE derived the total lengths of the pipe elements from particle-tracking results of the three-dimensional saturated zone model. Groundwater-specific discharge values in each pipe segment were also estimated from the three-dimensional site-scale flow model, as described in SNL (2008ag, Section 6.3.10). Where possible, the one-dimensional model used the same transport parameters, such as sorption coefficients, as the three-dimensional model. DOE adjusted other features of the one-dimensional model, such as dispersivity and flow tube diameters, to improve consistency between the breakthrough curve results from the two abstractions.

Each point source mass of radionuclides collected from the unsaturated zone radionuclide transport model abstraction was transferred to the three-dimensional and the one-dimensional saturated zone transport model abstractions. DOE generated three-dimensional model breakthrough curves by randomly sampling different locations within each of the four subareas to create a starting point for the saturated zone flow path(s). In contrast, the one-dimensional model used a fixed, centroid location within each subarea as a starting point for each of its four flow tubes. As a result of the different starting points for the three-dimensional and one-dimensional simulations, the path lengths were not necessarily the same for both methods. DOE described the comparison of the three-dimensional and one-dimensional simulations as not consistently overestimating or underestimating the travel time to the accessible environment.

In the TSPA calculations, DOE's saturated zone radionuclide transport abstraction was coupled to input from the unsaturated zone and output to the biosphere using the convolution integral method. In this method, a unit saturated-zone radionuclide mass breakthrough curve was computed (by the three-dimensional model) for a step-function mass flux source. This breakthrough curve was then combined with the radionuclide mass flux history from the unsaturated zone to produce a radionuclide mass flux history that was output to the biosphere. Within the TSPA, the convolution integral technique was implemented by a module called *SZ_Convolute*, which is based on assumptions of linear behavior and steady-state saturated zone flow conditions (SNL, 2008ag, Section 6.3.10.3). The *SZ_Convolute* module was also used to apply changes in specific discharge due to climate change and to correct radionuclide releases from the three-dimensional model for the effects of radioactive decay. In the TSPA calculations, DOE assumed that the saturated zone output mass that crosses the accessible boundary in a year was dissolved in 3,000 acre-ft [3.7×10^9 L] of water.

DOE's saturated zone transport model explicitly simulates the transport of the same 27 radionuclides that were passed to it by the unsaturated zone transport abstraction. Parameters affecting transport in the saturated zone were assigned to these 27 radionuclides, and these radionuclides were modeled in the saturated zone as being carried in the groundwater in the dissolved state and as temporarily and/or permanently associated with colloids. In addition to the 27 radionuclides for which transport is modeled explicitly in the saturated zone transport abstraction, DOE assumed that 4 additional radionuclides with half-lives of less than 29 years—Ac-227, Ra-228, Th-228, and Pb-210—are in secular equilibrium in the saturated zone with their long-lived parents—Pa-231, Th-232, U-232, and Ra-226, respectively, and that they are released from the saturated zone to the biosphere at the same activity as their parents.

DOE's saturated zone transport model passes to the biosphere model the time-varying mass of radionuclides that cross the accessible environment compliance boundary. The DOE biosphere model then calculates biosphere dose conversion factors for the ground water exposure scenario equivalent to the annual dose from all potential exposure pathways that the RMEI would experience as a result of the release of a unit concentration 1 Bq/m^3 [0.227 dpm/gal] of the primary radionuclide in groundwater at the accessible environment boundary. In addition to the 31 primary radionuclides provided as input to the biosphere model from the saturated zone transport abstraction, the biosphere model also accounts for the radiological effects of an additional 44 short-lived radionuclides that are the decay chain progeny of Ac-227, Ra-228, Th-228, and Pb-210 (SAR Table 2.3.10-5).

NRC Staff's Review

The NRC staff reviewed the information DOE provided in SAR Section 2.3.9 and in SNL (2007ba, Sections 6.3 and 6.4) and references therein about the saturated zone transport abstraction and its integration with related model abstractions in the TSPA calculations. The NRC staff reviewed the technical basis and model properties for DOE's saturated zone flow model abstraction in SER Section 2.2.1.3.8. The NRC staff's review of the unsaturated zone transport abstraction is documented in SER Section 2.2.1.3.7. The NRC staff's review of the biosphere abstraction is documented in SER Section 2.2.1.3.10.

The NRC staff reviewed DOE's model framework for transport of radionuclides in the saturated zone and conducted independent analyses using both TPA (Leslie, et al., 2007aa) and GoldSim (Bradbury, 2010aa, ab) to compare and confirm that the DOE modeling approaches and assumptions were consistent with its site description and conceptual model. The NRC staff finds that one-dimensional transport simulations using the pipe elements of the GoldSim (GoldSim Technology Group, 2006aa) computer code are acceptable because they adequately represent the important processes controlling radionuclide transport and concentrations. The NRC staff reviewed in detail the implementation of the convolution integral method (BSC, 2005ak) employed by DOE to quantify the mass transport of some radionuclides through the saturated zone. The NRC staff compared DOE's validation results of their three-dimensional and one-dimensional models (BSC, 2005ak) and concludes that the similar results confirm the acceptability of DOE's models of radionuclide transport in the saturated zone. The NRC staff concludes that DOE's model framework is adequate to represent site conditions and processes because DOE applied consistent conceptual models and estimates of data and model uncertainty for site conditions relevant to the saturated zone at Yucca Mountain.

On the basis of its review of information DOE provided about saturated zone radionuclide transport by advection in SAR Section 2.3.9.3 and references therein, the NRC staff confirmed that advective radionuclide transport in DOE's transport abstraction is integrated with the flow field information DOE's site-scale saturated zone flow model supplies to the transport abstraction. The NRC staff concludes that DOE's implementation of advective radionuclide transport is adequately integrated with the site-scale saturated zone flow model because both models use the same three-dimensional model grid, hydrologic properties, modeling approach, and flow fields to represent advection fluxes.

The NRC staff concludes that in DOE's integration of the unsaturated and saturated zone transport abstractions, DOE adequately transferred radionuclide mass between the two abstractions, based on the following reasons:

- DOE's transfer of mass adequately characterizes the uncertainty associated with the modeling methods.

- DOE sensitivity analyses of the effects of releasing radionuclide mass from the unsaturated zone as point sources indicated that the point source releases generally produced faster breakthroughs, as described in SNL (2007ba, Section 6.8.4).

- DOE's release of radionuclides as a point source generally produced a plume with less dispersion.

- Each radionuclide species and its type of transport remained consistent in the transfer of mass from the unsaturated zone to the saturated zone. For example, radionuclides associated with irreversible colloids in the unsaturated zone were passed to the saturated zone as irreversible colloids, while those associated with reversible colloid transport in the unsaturated zone were repartitioned as reversible colloids, according to the different colloid concentrations encountered in the saturated zone.

The NRC staff concludes that DOE's method of adjusting parameters in the one-dimensional transport model is acceptable because the parameters that were adjusted were shown to be unimportant to performance, and those parameters that were important to performance were held the same for the three-dimensional and one-dimensional models. The NRC staff finds that one-dimensional transport simulations using the pipe elements of the GoldSim (GoldSim Technology Group, 2006aa) computer code are acceptable because they adequately represent the important processes controlling radionuclide transport and concentrations. In addition, DOE's use of two alternative modeling approaches to simulate transport produced similar results, consistent with available data, which also provides support for the model abstractions.

2.2.1.3.9.3.2 Saturated Zone Transport Processes

In DOE's saturated zone transport abstraction, the migration of radionuclides through the saturated zone is influenced by the transport-affecting processes of advection and dispersion, sorption, matrix diffusion, and colloid-facilitated transport, as well as radioactive decay and ingrowth (SAR Section 2.3.9). The NRC staff's review focuses on DOE's process descriptions and integration and how DOE addressed data support, data uncertainty, model uncertainty, and model support.

2.2.1.3.9.3.2.1 Advection and Dispersion

Advection is the process by which radionuclides, both dissolved and associated with colloids, are carried in flowing water. Overall, DOE considered advection to be the most important transport process in the saturated zone (BSC, 2005ak, Section 6.3.1). Accordingly, the uncertainty of specific discharge, or the measure of flow in the saturated zone, has the greatest effect on travel times, as described in SNL (2007ba, Section 6.8.4).

Unlike DOE's model framework for unsaturated zone radionuclide transport, which allows radionuclides to travel advectively in fractures and in the matrix, radionuclides in the fractured volcanic tuff in the saturated zone move advectively only in the fractures and fault zones (i.e., no advection in the rock pore spaces between mineral grains). Hydrologic testing conducted by DOE in boreholes in the volcanic aquifer revealed that flow through fractures was generally spaced at intervals significantly greater than the spacing of the fractures themselves, as determined in drill core logging, indicating that not all fractures in the system contribute equally to flow. DOE included this site characteristic in the model abstraction as the uncertain parameter *flowing interval spacing in volcanic units*. DOE coupled this with another uncertain parameter, termed *Fracture porosity in volcanic units* (SAR Table 2.3.9-4), which DOE described as the flowing interval porosity, based on fracture spacing in the fractured tuffs (SAR Section 2.3.9.3.2.1). Values for this parameter were estimated using various conservative tracers and reactive tracers in C-Wells Complex testing (SNL, 2007aw). DOE used a combination of the fracture spacing and the porosity parameters to describe the characteristic of preferential pathways through the fracture volcanic aquifer. DOE also provided supporting information about advective transport processes from saturated zone tracer experiments in densely welded, fractured tuffs (fracture-flow dominated systems) at the C-Wells Complex and in the porous alluvium (SAR Section 2.3.9.3.2.1; SNL, 2007aw).

Advection in the alluvium of Fortymile Wash, like advection in the volcanic aquifer, involves preferential pathways. In the alluvium, DOE used an effective porosity parameter to compensate for the potentially reduced volume of the alluvium through which flow might occur. A smaller effective porosity results in higher average linear velocity (i.e., the distance water moves through porous material per unit time). DOE modeled the alluvium as a single-continuum medium. Consequently, there is no exchange of water or radionuclides between the effective porosity (where flow occurs) and the rest of the porosity (where no flow occurs) modeled in the alluvium. Field evidence of tracer exchange between effective porosity, where water is flowing, and porosity where water is stagnant, was inconclusive (SNL, 2007aw). DOE opted to not include matrix diffusion processes in the alluvium segment of their radionuclide transport model.

Dispersion describes the transverse spreading, perpendicular to flow, both horizontal and vertical, and longitudinal spreading, parallel to flow, of dissolved radionuclides in response to localized differences in flow conditions. At the large scale of the saturated zone transport model grid framework, DOE considered the effect of dispersion to be minimal (SAR Section 2.3.9.3.2.1). However, to allow transport calculations to provide an analysis of radionuclide travel time distributions, DOE included a longitudinal dispersion term in the transport model to capture the arrival of a dispersed solute front at the accessible environment boundary (SAR Section 2.3.9.3.2.1). Inclusion of longitudinal dispersion is supported by the field evidence of preferential pathways in the saturated zone in the vicinity of Yucca Mountain (SAR Section 2.3.9.3.2.1).

NRC Staff's Review

The NRC staff's review of DOE transport modeling results from TSPA and from process-level simulations (e.g., SAR Section 2.3.9; SNL, 2007bj, Section 6.1) confirms that advection and the specific discharge parameter used to represent advection are important to radionuclide transport in the saturated zone. The NRC staff finds that the DOE's approach to simulating advection in the alluvium is adequate because the approach includes processes and conditions shown to be present in the alluvium and excludes processes, such as matrix diffusion in alluvium, that have not been shown to occur. The NRC staff concludes that DOE provided an adequate technical basis for implementing advective radionuclide transport in the saturated zone radionuclide transport abstraction because the implementation is directly supported by the field testing in both the volcanic and alluvial aquifers.

The NRC staff finds that the methods to constrain the values of flow-related transport parameters are acceptable because the methods are supported by site characteristic data and field testing. The NRC staff finds that although a limited number of locations have been characterized to determine porosity and spacing, the DOE's estimation and implementation of the distribution range (large spacing and small porosity) compensates for the uncertainty in these parameters.

Although DOE's conceptual model does not explicitly account for all possible flow paths within the alluvium, the NRC staff considers the effective porosity approach to be an acceptable method for incorporating preferential pathways and minimizing the travel time through the alluvium because it is compatible with observed site characteristics, such as the occurrence of gravel paleochannels and lenses of clay (Bertetti and Prikryl, 2003aa; Nye County NWRPO, 2003aa; Ressler, et al., 2000aa).

The NRC staff reviewed DOE's assumptions and technical approach for including dispersion as a transport process in the saturated zone transport abstraction. On the basis of the NRC staff's recognition that dispersion does not significantly affect the results of transport calculations in a slowly evolving system where radionuclide concentrations gradually increase with time, the NRC staff concludes that the approach is acceptable.

2.2.1.3.9.3.2.2 Sorption

Sorption, as stated in SER Section 2.2.1.3.7.3.2.2, is a general term for chemical and physical processes that transfer a fraction of a dissolved species to the surface of a solid phase. Depending on specific properties of the dissolved species, the solid phase, and the liquid phase, the extent of sorption varies: some radionuclides will sorb strongly to the solid, some will sorb weakly onto the solid, and some will not sorb at all. As modeled by DOE for the transport of radionuclides through the saturated zone, sorption onto the fractured volcanic tuff matrix or onto alluvium results in retardation, or slowing, of radionuclides relative to rates of water flow through the saturated zone. In contrast, radionuclide sorption onto mobile colloids may enhance the transport rate of radionuclides relative to their sorption onto a stationary solid.

DOE identified sorption as an important process contributing to the barrier capability of the saturated zone (SAR Section 2.3.9). In particular, DOE model results indicate that sorption within the alluvium effectively delays the transport of moderately and strongly sorbing radionuclides for thousands of years or longer (SAR Sections 2.3.9 and 2.1.2.3.6). DOE estimated that sorption of dissolved thorium, americium, and protactinium is so effective in the saturated zone that, upon entering the saturated zone, these radionuclides cannot traverse it to reach the accessible environment within the regulatory period of 1 million years. For these radionuclides to be present at the accessible environment boundary within the million-year timeframe, DOE determined that they must either be transported through the saturated zone as colloids or be ingrown as the decay products of mobile parents.

DOE represented sorption in the saturated zone with a sorption coefficient (K_d), an empirically determined or modeled value that represents the ratio of the radionuclide concentration on the solid-phase to the radionuclide concentration in the groundwater. Low values of K_d indicate that little or no sorption (i.e., $K_d = 0$) occurs; higher values indicate moderate or strong sorption, which results in retardation. Retardation by sorption is expressed in transport calculations by a retardation factor (R_f) that depends on the K_d value and the physical properties (porosity and density) of the solid medium through which the radionuclide is transported. Thus, an R_f equal to a value of 1 indicates the solute is transported at the velocity of groundwater, while an R_f greater than 1 indicates the solute's transport is delayed relative to the groundwater. Retardation calculations assume that (i) K_d does not vary with changes in radionuclide concentration, (ii) sorption and desorption reactions are fast relative to the flow rate, and (iii) the bulk chemical composition of the groundwater is constant (Davis and Curtis, 2003aa; Langmuir, 1997aa; Davis and Kent, 1990aa).

DOE noted that the primary controls on sorption are (i) the characteristics of the mineral surfaces onto which sorption occurs, (ii) the chemistry of groundwater in the saturated zone, and (iii) the sorption characteristics of each element (SAR Section 2.3.9.3.2.2). DOE assumed sorption of dissolved radionuclides would occur only in the matrix of the volcanic tuff or in the alluvium. Citing uncertainties about the nature of the fracture coatings, DOE excluded sorption onto fracture surfaces in the volcanic rock (SAR Section 2.3.9.3.2.2; SNL, 2007ba). However, DOE did assume that solutes transported through designated fault or fault-related fracture zones could undergo sorption depending on the characteristics of the zone (BSC, 2005ak). In fault-related fracture zones, a small portion of the rock matrix within the fracture zone was conceptualized as allowing rapid diffusion, and a retardation factor was calculated accordingly (BSC, 2005ak). DOE also assumed that mobile colloids could be retarded within fractures of the volcanic tuff (SAR Section 2.3.9.3.3). DOE cited laboratory and field-scale transport experiments to support its conceptual model of colloid retardation in fractures (BSC, 2005ak; SNL, 2007aw).

Development of Sorption Values

DOE provides the range of K_ds used in the TSPA in SNL (2007ba, Appendix A). For sorption modeling, DOE grouped the various stratigraphic units in the saturated zone into two geologic media that have different sorption characteristics: fractured volcanic tuff and alluvium (SAR Section 2.3.9.3.2.2). DOE measured sorption data from batch and column experiments that used site-specific samples of crushed tuff and alluvium, and the experiments used water chemistries based on water samples from wells in the saturated volcanic tuff (UE-25 and J-13), carbonate aquifer (UE-25 p#1), and alluvium (various EWDP wells). DOE used water chemistries from wells UE-25, J-13, and UE-25 p#1 for batch sorption experiments with crushed volcanic tuff samples, and experiments with alluvium used a water chemistry representative of the alluvial aquifer (SNL, 2007ba, Appendix G). DOE stated that these water chemistries bracket the major ion chemistry observed in the saturated zone, as described in SNL (2007ba, Appendix A).

In SNL (2007ba, Appendix J and Appendix G), DOE described several long-term batch and column sorption experiments involving the sorption and desorption of uranium and neptunium in which the effective K_ds for these radionuclides were up to two orders of magnitude greater than those used in the TSPA calculations. DOE stated that the greater values demonstrated that there are a variety of sorption sites in the alluvium that have varying sorption affinities. DOE also stated the K_ds of the batch sorption experiments from which ranges of K_d values were selected for the TSPA tend to be biased to lower values because their short duration preferentially measured sorption on weak sorption sites, and the longer-duration experiments measured slower desorption rates that were associated with higher affinity sorption sites. DOE chose not to incorporate the higher effective K_ds for uranium and neptunium in the K_d distributions for TSPA calculations because they were obtained inconsistently from the methods used to obtain the other K_d values, but DOE identified that consideration of the strong sorption sites and slow desorption behavior would ultimately result in higher K_d values (more sorption) if extrapolated to longer time and distance scales.

DOE identified that mineral surface area and particle size were potential sources of data uncertainty related to the use of crushed tuff and alluvium in experiments. DOE referenced studies both from within and outside the DOE program indicating that the effects of particle size on sorption are typically small except for the very fine (e.g., clay-sized) fraction (SNL, 2007ba). The smallest particle size results in higher K_ds. The general DOE approach to addressing this uncertainty was to use batch experiments for a range of particle sizes and to bias the minimum and maximum limits for the K_d distributions toward lower (weaker sorption) values, as shown in DOE (2009am, Table 1.1.2-1). DOE also performed a limited number of confirmatory column tests on selected radionuclides that DOE had identified as important contributors to mean annual dose in previous performance assessments, as outlined in SAR Section 2.3.9.3.2.2 and SNL (2007ba, Table 4-1).

For the long-lived actinides (americium, neptunium, plutonium, and uranium), DOE further characterized the effects of variability in geochemistry and mineral surface area using a nonelectrostatic surface complexation modeling approach, supported by Davis, et al. (1998aa) and SNL (2007ba, Appendix A). In some cases, DOE also supplemented the experimental and modeling sorption data with data from the open literature (SNL, 2007ba). In TSPA calculations, DOE sampled K_d values from the specified ranges to account for experimental uncertainty and variability in geologic conditions, including water chemistry and rock type, as shown in SAR Table 2.3.9-4; BSC (2005ak); SNL (2007ba, Appendices A, C, G, and J); and DOE (2009am, Enclosure 3).

In the saturated zone transport model and abstraction, DOE assumed that four radioelements (carbon, chlorine, iodine, and technetium) were nonsorbing, and DOE assigned a fixed value of K_d = 0 (corresponding to R_f = 1) to each. Although results of field-based testing conducted by DOE indicated that the transport of the risk-significant radioelements technetium and iodine may be somewhat retarded in the alluvium, DOE concluded that the laboratory-based sorption tests conservatively supported an assumed value of K_d = 0 (SNL, 2007ba, Appendix G). For the remaining radioelements modeled in saturated zone transport calculations (americium, cesium, neptunium, plutonium, protactinium, radium, selenium, strontium, thorium, tin, and uranium), DOE developed ranges and statistical distributions of K_d values for each radioelement and for each modeled rock unit from a combination of empirical data, process modeling, statistical analyses, and expert judgment (SAR Table 2.3.9-4; SNL, 2007ba, Appendices A, C, G, and J). In order to examine the effects of broader chemistry ranges on several radionuclides, DOE also used surface complexation modeling, which involves reactions that form bound species at the mineral–water interface, to extend the limited conditions covered by the batch crushed tuff experiments.

NRC Staff's Review

The NRC staff finds that the methods applied to determine distributions of K_ds are adequate. DOE provided adequate information to support the assumptions associated with use of the K_d approach and methods used to determine K_d values (DOE, 2008ab; BSC, 2005ak; SNL, 2007aw, 2007ba). For example, DOE varied the concentration of radionuclides in sorption experiments to determine the effect on K_d. DOE varied the duration of sorption/desorption experiments to determine the rate of these reactions. By using groundwaters of different compositions, DOE demonstrated the effect of bulk chemistry on K_ds. Based on DOE and independent analyses of surface areas of site-specific materials and surface area effects on sorption (Bertetti, et al., 2004aa; Bertetti, et al., 2011aa), the NRC staff finds that the DOE's approach to surface area impacts is acceptable.

The NRC staff reviewed the geochemical controls on radionuclide sorption and the experimental data DOE used to develop the TSPA K_d distributions (Bertetti, et al., 2011aa). The NRC staff finds that the major ion chemistry (e.g., calcium, sodium, bicarbonate) of the waters used in DOE sorption experiments is comparable to that of saturated zone waters, as described in SNL (2007ba, Appendix A). The UE-25, J-13, and UE-25 p#1 water chemistries bound the ranges reported for saturated zone water chemistries for major ions such as sodium, calcium, and bicarbonate, and other parameters such as pH and redox state. Based on the NRC staff's knowledge and experience, these chemical characteristics are likely to be the most important for radionuclide sorption (e.g., Turner and Pabalan, 1999aa). The NRC staff further concludes that the chemistries of alluvial aquifer waters used in alluvium sorption experiments are representative of conditions in the alluvium (McMurry and Bertetti, 2005aa).

The NRC staff concludes that the experimental approaches used by DOE to develop K_d values are adequate. In selecting experimental data to inform the TSPA K_d distributions, DOE appropriately excluded data from experiments where the final radionuclide concentration indicated that the solubility limit of the radionuclide may have been exceeded, as described in SAR Section 2.3.9.3.2, SNL (2007ba, Appendix A), and SNL (2007ah), and DOE did not include a particular set of experiments that may have been conducted at initial concentrations that were above the solubility limits for some radionuclides [SNL (2007ba), Appendix A].

The NRC staff reviewed DOE's surface complexation model used to extend the range of sorption data for several key actinides. Based on the NRC staff's knowledge and experience with other, independently developed models that have been used for similar insights about radionuclide sorption (e.g., Turner, et al., 2002aa), the NRC staff concludes that DOE acceptably used surface complexation modeling to extend the limited chemical conditions in the batch crushed tuff experiments and to support DOE's technical basis for the upper and lower limits of sorption coefficients for the targeted actinides.

Sorption Model Uncertainty and Support

DOE addressed model uncertainty in the TSPA calculations by sampling K_d values stochastically from uncertainty distributions in which the distribution ranges were developed from expected system conditions. DOE conducted additional analyses to evaluate the effects of model scale and heterogeneity, as outlined in SNL (2007ba, Appendices C and D). Rather than sampling the K_d distribution independently for each radionuclide, DOE developed a correlation matrix for the 11 sorbing radioelements on the basis of their ranked sensitivities to six variables (pH, Eh, water chemistry, rock composition, rock surface area, and radionuclide concentration). DOE used this approach to approximate similarities in sorption behavior among radioelements and to ensure that transport behaviors were represented consistently within a single realization of the model, as outlined in SNL (2007ba, Appendix A). In addressing model uncertainty, DOE neglected sorption (i.e., K_d = 0) in fractures (fast flow paths), except for fault zones, and implemented K_d uncertainty distributions for matrix sorption. DOE explained that in most cases, DOE's modeling approach underpredicted the effectiveness of sorption as compared with measured distributions, and consequently, as modeled by DOE, radionuclide transport was less impeded by sorption than would otherwise be expected.

DOE used information from natural analogs and field tests to provide qualitative comparisons for sorption model confidence building at the field scale (SAR Section 2.3.9.3.4.1.3), and DOE used general observations of sorption-related transport behavior to support the conceptual models (e.g., SAR Section 2.3.9.3.4.1.4). DOE also used observations from field tests at Busted Butte south of Yucca Mountain, from the C-Wells location, and from two alluvium tracer tests to provide qualitative and limited quantitative evaluations of sorption in the radionuclide transport model abstraction.

NRC Staff's Review

The NRC staff notes that the empirical K_d modeling approach implemented by DOE is simplistic but well established (e.g., Freeze and Cherry, 1979aa; Till and Meyer, 1983aa), and it has been broadly accepted in modeling studies for decades as a method to describe radionuclide transport (e.g., Sheppard and Thibault, 1990aa). A potential model uncertainty associated with the K_d approach in general is that individual K_d values are lumped parameters that do not explicitly take into account spatial and temporal variability or the role of specific surface-related processes that may affect radionuclide sorption. This uncertainty is commonly addressed by using a range or distribution of K_d values that incorporates temporal and spatial variability. DOE also addressed an assumption inherent in the K_d modeling approach for sorption and the convolution integral approach that DOE used to calculate mass released from the saturated zone that sorption and system behavior are linear (BSC, 2005ak). DOE adjusted the ranges of K_d distributions to focus on the expected linear range of sorption, as described in SNL (2007ba, Appendix A).

The NRC staff's review of DOE's sorption data and models identified that most of the K_d uncertainty distributions specified by DOE underpredicted the effectiveness of sorption compared to the experimental distributions. In some cases, DOE reduced the upper bounds of the K_d distributions (specifically, those of cesium, plutonium, and radium) relative to the range indicated by available data to account for the possible effects of slow sorption kinetics for these elements, as shown in SNL (2007ba, Appendix A). The NRC staff recognizes that by using low ranges of K_d values for sorption, DOE's transport model underpredicts retardation, resulting in relatively faster modeled travel times for radionuclides through the saturated zone, and thus earlier breakthrough of transported contaminants and potentially larger modeled releases of radionuclides to the accessible environment. The NRC staff notes that this approach can be conservative relative to performance, especially for the 10,000-year period, during which DOE models predict that few radionuclides reach the accessible environment. However, the approach may not be as conservative for the one-million-year performance period because during the longer period available for radionuclides to reach the accessible environment, the difference in breakthrough time becomes less significant. For this longer performance period, one effect of relatively larger K_d values is that, later in time, the increased sorption potential can result in accumulation of certain radionuclides in alluvium along the transport path and in the vicinity of the receptor location. In such a case, the NRC staff notes two processes that could affect the barrier properties of the saturated zone.

First, a significant perturbation in groundwater chemistry in part of the saturated zone might cause the accumulated radionuclide to desorb and be released unexpectedly to the surface environment. With respect to this possibility, however, the NRC staff has reviewed DOE's technical basis for excluding geochemical interactions and evolution in the saturated zone, including temporal changes in groundwater composition, and found that DOE's exclusion of this FEP was acceptable (SER Section 2.2.1.2.1). The NRC staff's review concluded that DOE's basis for exclusion was adequately supported by site characterization data, field tests, laboratory experiments, and natural analogs. The NRC staff also notes that the range of groundwater compositions used by DOE to establish radionuclide transport parameters encompasses the temporal variances that have been documented during Yucca Mountain site characterization activities (Turner and Pabalan, 1999aa).

Second, the ingrowth of a daughter radionuclide with different sorption properties than its parent could lead to secular disequilibrium and transport of the daughter that was not otherwise considered by the assessment of performance. In such a case, the use of low-biased K_d values for the decay chain parent increases the uncertainty in predicting the timing and magnitude of the peak dose during the one-million-year period because the assumptions do not account for the accumulation of the parent in the saturated zone as a source to generate the more mobile daughter. Given that a conservative representation of risk for one aspect of repository behavior may not lead to an overall conservative representation of risk for the system as a whole, the NRC staff performed independent transport simulations involving radioactive decay and ingrowth, with a particular focus on the sensitivity of TSPA results to the K_d distributions to examine DOE's assumption that using low ranges of sorption values for K_d distributions would not lead to risk dilution under some circumstances for radionuclide transport (Bradbury, 2010aa). Using the information from DOE's sorption/desorption experiments, as outlined by DOE in SNL (2007ba, Appendix J), the NRC staff reviewed the potential effects of higher K_ds on concentrations of parent-daughter radionuclides during saturated zone transport. The sensitivity analyses conducted by the NRC staff used pipe elements of the GoldSim (GoldSim Technology Group, 2006aa) modeling program to identify that over long time periods (e.g., the one-million-year performance evaluation), the total concentrations of the radionuclides of some decay series increase as a function of K_d of the parent. The NRC staff's transport simulations indicated that biasing the K_ds of certain radionuclides to lower values may lead to an underestimation of total radionuclide concentrations when considering the sum of the parents plus their ingrown daughters. For example, the distribution used by DOE's TSPA for the K_d of uranium has a mean of ~5 ml/g. Increasing the K_d of uranium to 200 ml/g, similar to values of the effective K_d obtained by DOE in other sorption experiments (SNL, 2007ba, Appendix J), increases the total flux of parent plus daughters by as much as a factor of 12, compared to using the lower mean K_d for uranium. In this example, the dose that the NRC staff calculated from the increase of these radionuclides is on the order of 10 mrem [100 μSv], as estimated by scaling the calculated dose by the relative increase in fluxes of daughter radionuclides, compared to a dose of approximately 2 mrem [20 μSv] that DOE obtained using the lower K_d value for uranium sorption.

To arrive at this estimate, the NRC staff followed a method for determining the effect of increased flux on dose similar to that used by DOE (2009de). The method begins with DOE's TSPA results for the contributions to dose by individual radionuclides. As shown in SAR Figure 2.4-20b, DOE calculated the total mean annual dose at 1 million years after repository closure to be approximately 2 mrem [20 μSv]. The principal radionuclides contributing to this dose are Pu-242 {~0.6 mrem/yr [6 μSv]}, Np-237 {~0.4 mrem/yr [4 μSv]}, and I-129 {~0.2 mrem/yr [2 μSv]}. Daughter radionuclides of the uranium decay series account for most of the remaining calculated annual dose. Increasing this remaining dose component by the increased flux of uranium daughter products (in this example, by a factor of ~12) gives an estimated total mean annual dose on the order of 10 mrem [100 μSv], which is still significantly below the 100 mrem [1 mSv] regulatory standard. The NRC staff notes that this dose estimate is based on a single value for uranium sorption and is not weighted by the probability of occurrence, and independent analyses conducted by the NRC staff (Bertetti et al. 2011aa) indicate uranium K_d values for the saturated alluvium would likely be less than 200 mL/g. Similar analyses by the NRC staff involving neptunium as the parent instead of uranium determined that the impact for possibly higher sorption of Np-237 using an increased K_d could result in relatively smaller increases in total concentrations of parent plus daughters.

Another potential source of model uncertainty is the assumption that other transport-related factors are consistent over large scales spatially. The NRC staff notes that in DOE's analysis, pipe elements that DOE used to represent the alluvium ranged from 6.5 to 8.5 km [4.04 to 5.28 mi] in length. For each realization, a single K_d for each radionuclide is assigned to the full extent of the alluvium pipe element. The NRC staff used pipe elements of the GoldSim (GoldSim Technology Group, 2006aa) modeling program to examine the effects of pipe length on radionuclide release of parents and daughters to determine the sensitivity of DOE's approach to these parameters (Bradbury, 2010aa). The NRC staff analyses identified that spatial heterogeneity did affect radionuclide concentrations for the post-10,000-year regulatory period, but the magnitude of this impact is not significant to performance unless specific local geochemical conditions exist where sorption behavior is well outside the nominal ranges.

Based on its review of DOE's implementation of sorption processes in the saturated zone transport abstraction and on the basis of the NRC staff's knowledge and experience (e.g., Bertetti, et al., 2011aa; Turner, et al., 2002aa; Bradbury, 2010aa), the NRC staff concludes that DOE has adequately incorporated sorption modeling in performance assessment calculations for the following reasons:

- DOE used appropriate experimental techniques with site-specific materials, alternative computer models, field tests, and natural analogs to provide a technical basis to support the TSPA model abstraction of radionuclide sorption.

- DOE used site-relevant data to address the anticipated effects of pH, Eh, major ion water chemistry, rock composition, rock surface area, and radionuclide concentration on radionuclide sorption.

- DOE appropriately acknowledged the limitations of the K_d approach and used stochastically sampled K_d probability distributions and simplifying assumptions about the effectiveness of sorption to address model and data uncertainty, and DOE considered appropriate geochemical and physical conditions in developing the K_d distributions. In supporting the sorption model for performance assessment calculations, DOE also used alternate modeling approaches such as ion exchange reactions, in which ions of one element replace ions of another element within a mineral structure, and surface complexation, which involves reactions that form bound species at the mineral-water interface.

- DOE adequately described the method used to assess the sensitivity of radioelement sorption behavior to variability in geochemical and physical conditions, and DOE used these rankings to correlate sorption characteristics among the radioelements, ensuring consistency among the sorption parameters for each model realization.

- In comparison with results from DOE desorption experiments, DOE has biased the distributions of K_d values in TSPA to lower values that may underestimate the field-scale sorption of radionuclides. Although this method of biasing K_d distributions can underestimate concentrations of some daughters for disequilibrium in certain decay chains (DOE, 2009df), leading to greater uncertainty in the calculated concentrations of the daughter radionuclides in ground water, the NRC staff's independent analyses confirmed DOE's assumption that the impact of these potential greater uncertainties is not significant for performance (Bradbury, 2010aa).

2.2.1.3.9.3.2.3 Matrix Diffusion

Diffusion is a physical process in which dissolved radionuclides move from a region of high concentration to a region of low concentration without advection. DOE described matrix diffusion as a fracture–matrix interaction that uses diffusion to transfer radionuclides between the water in fractures and the water in the rock matrix. DOE identified matrix diffusion as a moderately important transport mechanism in the saturated zone transport abstraction, especially for strongly sorbing radionuclides, because it is the main process by which radionuclides can move from a fracture-dominated flow path into the matrix. DOE's modeled effectiveness of matrix diffusion depends on (i) the matrix diffusion rate (i.e., the rate that a radionuclide can diffuse from the water in a fracture into water in the pore spaces of the rock matrix) and (ii) the area of the fracture–matrix interface across which diffusion occurs, as outlined in SNL (2007bj, Section 6.1.2.4). The matrix diffusion rate depends on the concentration gradient of the radionuclide between fracture and matrix and the value of the effective matrix diffusion coefficient, which is a measure of how readily a particular radioelement diffuses through a tortuous pathway of interconnected pores in the rock matrix. DOE estimated tortuosities from empirical data for representative Yucca Mountain tuff samples and developed standard normal cumulative probability distributions for effective matrix diffusion coefficients that were sampled stochastically in TSPA for each radioelement with respect to the individual model units (SAR Section 2.3.9.3.2; Reimus, et al., 2007aa).

In contrast to fractures in the unsaturated zone, where not all connected fractures in unsaturated rocks are water-bearing at the same time, all fractures in the saturated zone are, by definition, water-bearing. However, as described in SER Section 2.2.1.3.9.3.2.1, not all fractures of the saturated zone aquifer participate in the flowing system. The flowing interval spacing reduces the number of fractures contributing to flow. DOE measured this site characteristic in field tests and conducted drill core logging to evaluate the spacing between fractures. Downhole spinner tests conducted by DOE at isolated intervals in boreholes indicated the spacing of flowing intervals. In the Yucca Mountain vicinity, the flowing interval spacing is greater than that of the fractures. DOE accounted for saturated zone transport-related model uncertainties by sampling values for the flowing interval spacing, the fracture porosity, and the effective diffusion coefficient in volcanic units. DOE's field experiments at the C-Wells Complex included cross hole tests where tracers with distinct diffusion coefficients were simultaneously injected into one well and the breakthrough curves of the different tracers were measured in a pumped well 30 m [32.8 yd] from the first. The differences in the breakthrough curves for the various tracers were used to demonstrate that matrix diffusion was affecting tracer migration.

NRC Staff's Review

The NRC staff evaluated the information DOE provided about matrix diffusion in SAR Section 2.3.9.3.2.1 and in DOE (2009an) and incorporated the NRC staff's independent understanding of matrix diffusion models, the hydrogeologic characteristics of the saturated zone at the Yucca Mountain site, and field and laboratory studies of fracture–matrix interactions in saturated fractured rocks at Yucca Mountain and elsewhere. The NRC staff finds that DOE adopted an established theoretical approach to estimate radionuclide-specific effective matrix diffusion coefficients and, in developing the parameter values for matrix diffusion coefficients, DOE appropriately (i) synthesized Yucca Mountain geological and hydrological data, (ii) adapted the estimated values for saturated conditions, and (iii) accounted for uncertainty and natural variability in diffusion characteristics of different rock units. The NRC staff concludes that DOE included sufficient and appropriate data and adequately addressed data uncertainty in developing effective matrix diffusion coefficients for the saturated zone transport abstraction.

The NRC staff finds DOE's field experiments provide adequate confirmation that matrix diffusion is a process of attenuation in the volcanic rocks of the saturated zone because the experiments were designed to identify the process, if present. The NRC staff compared DOE's large-scale tracer tests with published results from other saturated zone field experiments, modeling studies, and natural analogs at Yucca Mountain and elsewhere. In the context of data and model uncertainty about matrix diffusion as a saturated zone transport process, the NRC staff notes that DOE's performance assessment results illustrate that the retardation effect of matrix diffusion in the performance assessment is limited primarily to moderately and strongly sorbing radionuclides. The NRC staff's review concludes that DOE's TSPA results for radionuclide transport in the saturated zone are comparable either with or without matrix diffusion included as a retardation process. Consequently, the NRC staff concludes that DOE has provided an adequate technical basis for DOE's representation of matrix diffusion in the saturated zone transport abstraction.

2.2.1.3.9.3.2.4 Colloid-Associated Transport

As described in SER Section 2.2.1.3.7.3.2.4, colloids are minute solid particles of any origin or composition that become suspended in a liquid. Because colloids are mobile in water, a radionuclide that is attached to a colloid (e.g., by sorption to the colloid surface) will be transported by processes that move the colloid instead of by processes that would otherwise delay transport of the radionuclide as a dissolved species. Moreover, radionuclides attached to colloids tend to be transported preferentially in fast flow zones such as fractures or large pore throats because the size of colloids, compared to dissolved species, inhibits the transfer of colloids into fine-grained matrix, as described in SNL (2008an, Section 6.8.2).

DOE modeled colloidal transport in the saturated zone consistent with modeling used elsewhere in the TSPA, with two types of radionuclide attachment: reversible and irreversible (BSC, 2005ak). Colloids with irreversibly attached radionuclides were modeled as separate transported entities, with a retardation factor applied specifically to the fractured volcanic tuff and alluvial aquifers to simulate the effects of nonpermanent filtration. DOE assumed that the size of irreversible colloids could exceed that of the pores of the volcanic matrix. Consequently, DOE did not incorporate matrix diffusion of irreversible colloids in the saturated zone transport abstraction. Plutonium and americium were modeled as associated with both irreversible colloids and reversible colloids and as dissolved species in the saturated zone transport model, consistent with DOE's unsaturated zone transport model. Reversible colloidal transport was modeled using the K_c factor, which represented equilibrium sorption of aqueous radionuclides onto natural system colloids. Radionuclides associated with reversible colloid transport comprised 4 of the 12 radionuclide groups modeled in the saturated zone flow and transport abstraction. These four groups included (i) plutonium, (ii) cesium, (iii) tin, and (iv) americium, protactinium, and thorium. Application of the K_c factor and inclusion of reversible sorption to colloids lowered the effective diffusion coefficient and the sorption coefficient, K_d, for the radionuclides, enhancing advective transport.

DOE included colloid-associated transport in both the three-dimensional saturated zone flow and transport abstraction and the one-dimensional saturated zone transport model (BSC, 2005ak). In the TSPA model abstraction, radionuclide mass exiting the unsaturated zone was partitioned into solution and onto colloids for transport in the saturated zone, as outlined in SNL (2008ag, Section 6.1.4.9). Irreversible colloids leaving the unsaturated zone were passed to the saturated zone transport abstraction as a single, irreversible colloid flux. On the basis of data from field experiments that some colloids travel with little or no retardation (Kersting, et al., 1999aa; SNL, 2007aw), DOE designated a small fraction (less than 0.2 percent) of the irreversible colloid flux as a completely unretarded "fast fraction." For saturated zone transport calculations, DOE divided the irreversible colloid flux into a "slow" irreversible colloid fraction that is subject to modeled retardation processes during transport and the much smaller "fast" fraction. As noted by DOE, colloid-associated transport of radionuclides is affected by filtration, the rate of desorption from the colloid, and the colloid concentrations in the groundwater (SAR Section 2.3.9.3). Each of these factors was included in the saturated zone colloid-associated transport model.

DOE's colloid-associated transport model treats radioactive decay in irreversible colloids by assuming that if a decay product was also one of the two radioelements associated with an irreversible colloid in the model (i.e., isotopes of plutonium and americium), then the decay product remained irreversibly associated with the colloid (SAR Section 2.3.9.3.2.3). Otherwise, the decay product enters the aqueous phase as a dissolved species (SAR Section 2.3.9.3.2.3). In the model abstraction, there was no permanent filtration of irreversible colloids due to size exclusion in the tuff matrix, at the transition from tuff to alluvium, or in the alluvium, so no colloid size parameter was required in the saturated zone transport models (SAR Section 2.3.9.3.2.3). The nonpermanent filtration of irreversible colloids was implicitly included as part of the basis and development of the irreversible colloid retardation factor for both the tuff and the alluvium (SAR Section 2.3.9.3.2.3; BSC, 2005ak, 2004bc).

DOE's conceptual model assumed that reversible colloids could be represented by particles with the composition and characteristics of the clay mineral montmorillonite. DOE developed the uncertainty distributions for the concentration of groundwater colloids from data collected in saturated zone field studies from the Yucca Mountain region and from groundwater analyses elsewhere (BSC, 2005ak; SNL, 2007aw,bi). The colloid concentrations represented in the model covered a broad range of values that account for higher colloid concentrations measured in some groundwaters, with these higher concentrations given a low probability of occurrence. DOE assumed colloids to be stable for all water chemistry conditions in the saturated zone.

NRC Staff's Review

In reviewing DOE's technical basis for colloid-associated transport in the saturated zone, the NRC staff evaluated information DOE provided in SAR Section 2.3.9 and references therein (BSC, 2004bc; SNL, 2008ag,an, 2007bi). The NRC staff also considered additional information in DOE (2009am, Enclosures 9–14) and the NRC staff's independent experience with colloid-associated transport processes and models in heterogeneous natural systems such as the saturated zone at Yucca Mountain.

On the basis of the information DOE provided in SAR Section 2.3.9 and supporting references, the NRC staff concludes that DOE's representation of colloid-associated transport in the saturated zone is acceptable for performance assessment calculations for the following reasons:

- DOE developed an adequate conceptual and mathematical basis for colloid-associated transport processes in the saturated zone (e.g., retardation of colloids by attachment processes in fractured volcanic tuff and alluvium, reversible sorption of radionuclides onto colloids, colloid exclusion processes, and unretarded colloidal transport) that is consistent with existing models for contaminant transport in fractured rocks and porous media in the literature (e.g., Sudicky and Frind, 1982aa).

- DOE provided model results that are consistent with cross-hole field tests using microspheres showing decreased retardation of colloid-associated radionuclides relative to dissolved constituents. The modeling results and field-test results are consistent with the K_c factor approach used to represent colloid-associated transport.

- DOE selected a set of radioelements to model colloidal-facilitated transport that are the most strongly sorbed, and the saturated zone approach is consistent with that used in DOE's unsaturated zone model. The radioelements that are the most strongly sorbed to the colloids are those that contribute the most to dose.

- DOE's treatment of colloid-associated transport is consistent with DOE's model for partitioning of the radioelements among the three transport entities (dissolved species, reversibly associated with colloid, and irreversibly associated with colloid), which is evaluated in SER Section 2.2.1.3.4. The NRC staff considers the inclusion of similar colloid-associated modeling approaches and assumptions used for parent and daughter radionuclide attachment in both saturated zone transport models to be adequate and consistent.

- DOE's assumptions for colloid concentrations and stability in the saturated zone are consistent with groundwater analyses observations for the Yucca Mountain region.

Although naturally occurring colloids in Yucca Mountain groundwaters consist of montmorillonite, zeolite, and silica, the use of montmorillonite alone is adequate, as the specific mineral is less significant than the sorption coefficients assigned to it. DOE broadly addressed data uncertainty for sorption onto reversible colloids by selecting a reasonable range of montmorillonite sorption coefficients, which captures the sorption behavior of other potential colloid minerals.

- DOE adequately accounted for radioactive decay and ingrowth processes for radionuclides in the form of dissolved species, reversible colloid species, and irreversible colloid species included in the saturated zone transport abstraction. DOE's treatment of decay chain daughter nuclides for irreversible colloids is adequate because it is consistent with DOE's model assumptions about which radionuclides are associated with reversible and irreversible colloids.

- DOE's modeling approach adequately compensated for the high uncertainty in empirical observations for saturated zone colloidal transport in field studies or natural analogs by using reasonable probability distributions for most colloid-related parameters.

With respect to DOE's representation of radionuclide transport by reversible colloids, the NRC staff concludes that DOE provided an adequate technical basis by accounting for system variability in developing parameter values, where feasible, from site-specific data from saturated zone field tests in the Yucca Mountain area and sampling colloid-associated parameter values from large uncertainty distributions. The NRC staff finds that DOE adequately addressed model uncertainty because the results are consistent with the NRC staff's understanding of colloid-associated transport processes and the uncertainties involved in characterizing colloidal transport processes in natural systems.

With respect to DOE's representation of radionuclide transport by irreversible colloids, the NRC staff concludes that DOE's model is adequate because it includes processes that have been demonstrated to be present in field tests and lab experiments. DOE's approach used reasonable distributions of parameter values, simple model abstractions supported by field and lab tests, and analyses of natural analogs and underground nuclear tests. For example, the only radioelements irreversibly associated with colloids in DOE's model are plutonium and americium; this assumption is integrated with DOE's near-field model assumptions stating that after the failure of waste containers due to general corrosion in TSPA simulations, up to 30 percent of the Pu-242 flux transported to the accessible environment is by irreversible colloids (e.g., SAR Section 2.4.2.2.3.2.2 and Figure 2.4-108). The NRC staff concludes that DOE's election to not consider permanent filtration of irreversible colloids is acceptable because it allows for larger releases of colloid-associated radionuclides.

2.2.1.3.9.3.2.5 Radionuclide Decay and Ingrowth

Radioactive decay is a general term for the processes by which unstable radionuclides spontaneously disintegrate to form a different nuclide that may or may not be radioactive. Various heavy radionuclides are parents to decay chains of multiple radioactive daughters (e.g., SAR Figure 2.4-21). In the absence of chemical differences in transport behavior, the radionuclides in a decay chain reach secular equilibrium, where parents and daughters have equal radioactivity. Disequilibrium of naturally occurring decay chains is observed in many groundwater systems, due to geochemical processes in the aquifers that affect the accumulation or transport of decay chain members differently (e.g., Faure, 1986aa).

Loss of radionuclides over time due to radioactive decay and, where applicable, the potential ingrowth (increase) of radionuclide daughters were included in the DOE saturated zone transport abstraction. However, DOE's three-dimensional site-scale saturated zone transport model did not directly account for radioactive decay and ingrowth processes. Instead of explicitly including the ingrowth of decay progeny in the model, DOE adjusted the values for radionuclides that were received from the unsaturated zone and were transported in the saturated zone by including radioactive decay in the convolution integral model during the calculation of mass breakthrough. DOE used the three-dimensional site-scale saturated zone transport model to determine the saturated zone mass releases of nine radionuclides that are not members of decay chains (C-14, Cl-36, Cs-135, Cs-137, I-129, Se-79, Sn-126, Sr-90, Tc-99), and of eleven long-lived decay chain radionuclides (Am-243, Pu-239, Am-241, Pu-240, Pu-242, Pu-238, Np-237, U-234, U-232, U-236, and U-238).

DOE used two mechanisms to account for the ingrowth and transport of daughter radionuclides in the saturated zone. First, before initiating transport calculations in the three-dimensional site-scale saturated zone transport model, DOE took the as-received masses (i.e., the output from the unsaturated zone transport model) of five "first-generation" decay chain daughters (Np-237, U-234, U-236, U-238, and Pu-239), and boosted their masses by the amount their parents (Am-241, U-238, Pu-240, and Am-243, respectively) were expected to decay during their simulated performance time period in the saturated zone (SNL, 2008ag, Section 6.3.10.3). Second, DOE used the one-dimensional transport model to represent the transport of the radionuclides U-235, U-233, Th-230, Pa-231, Th-229, Th-232, and Ra-226 through the saturated zone to the biosphere and to account for their ingrowth during transport. DOE also used its one-dimensional saturated zone transport model to calculate the ingrowth of "second-generation" daughter radionuclides in selected decay chains (SAR Section 2.3.9.3.4.2; BSC, 2005ak).

To determine the total mass of radionuclides transported through the saturated zone, DOE added the mass of the secondary daughters transported by the one-dimensional model to the mass of radionuclides transported by the three-dimensional model. Lastly, in determining the saturated zone output to the biosphere model, DOE assumed that four additional decay chain members (Ac-227, Ra-228, Th-228, and Pb-210) were in secular equilibrium with their dissolved parents (SAR 2.3.9, Section 2.3.9.3.4.2.1). Overall, the saturated zone transport abstraction determined the release to the biosphere, including ingrowth by the decay of a parent, for 31 radionuclides. The biosphere model then used the releases to calculate radiological effects in the biosphere, as described in SAR Section 2.3.10. The biosphere model also implicitly accounted for the radiological effects of an additional 44 short-lived radionuclides that are the decay chain progeny of Ac-227, Ra-228, Th-228, and Pb-210 (SAR Table 2.3.10-5).

DOE supported the assumption of secular equilibrium in the saturated zone transport modeling approach, in which the released activity of daughters was set equal to that of their released parents, by conducting additional analyses to examine the sensitivity of this assumption to differences in sorption characteristics between parent and daughter radionuclides (DOE, 2009df). DOE's analyses identified that the ratio of activities for the daughters of the Th-232/Ra-228, Pa-231/Ac-227, Ra-226/Rn-222, and Ra-226/Pb-210 parent-daughter pairs could range from a value of 1 (e.g., Ra-226/Pb-210) to more than 1,400 (e.g., Ra-226/Rn-222) times that of the respective activities of the parents. DOE defined a sorption enhancement factor (SEF) as the ratio of the daughter activity relative to the activity of its parent in groundwater, resulting from the differences in sorption coefficients of the parent and daughter, and DOE stated that explicitly including the increased daughter activities in the dose calculation of the four parent/daughter pairs would cause the calculated maximum total mean annual dose to the RMEI to increase from a value of 2 mrem [0.02 mSv] to a value of 2.4 mrem [0.024 mSv]. DOE identified this as a small change relative to the 100 mrem [1 mSv] individual protection standard for the 10,000- to 1-million-year postclosure performance period.

NRC Staff's Review

The NRC staff has reviewed DOE's basis for the selection of nuclides included in the groundwater transport abstractions in SER Section 2.2.1.3.4.3.2 and found it to be acceptable. The NRC staff reviewed DOE's radionuclide transport analyses in SAR Sections 2.3.9 and 2.4.2 and confirmed that DOE's results corresponded with expected changes in the transported radionuclide inventory based on DOE's implementation of radioactive decay and ingrowth. DOE's calculations of radionuclide decay and ingrowth used standard mathematical equations and accepted decay constants and half-lives (International Union of Pure and Applied Chemistry, 1997aa). To confirm DOE's conclusion (DOE, 2009df) that deviations from secular equilibrium would not significantly affect TSPA results, the NRC staff performed independent analyses (Bradbury, 2010aa) using the pipe elements of the GoldSim (GoldSim Technology Group, 2006aa) code to simulate transport of decay chain radionuclides, including those short-lived radionuclides not included in DOE's model. Based on the results of these analyses and the information provided by DOE, the NRC staff concludes that although the concentrations and activities of some radionuclides may be somewhat greater than described in the SAR (as discussed in the following paragraphs), the resultant calculated doses for all radionuclides are well below the regulatory limit.

Regarding DOE's assumptions of secular equilibrium in groundwater for parents and daughters that have substantially different sorption properties, the NRC staff has reviewed in detail the supplemental information that DOE provided (DOE, 2009df) to support DOE's assumption that the ingrowth of a daughter with substantially different sorption properties in the saturated zone would not significantly change the dose calculated using DOE's assumption of secular equilibrium. Given the examples DOE presented, and additional analyses conducted by the NRC staff (SER Section 2.2.1.3.9.3.2.2), the NRC staff concludes that DOE's use of lower K_ds is acceptable in the context of decay and ingrowth processes because biasing K_ds of the parents in the saturated zone transport modeling has only a limited impact on the calculated total annual flux in the modeled examples.

The NRC staff identified several additional considerations that are related to DOE's assumption of secular equilibrium in groundwater. First, in addition to the parent-daughter radionuclide pairs that DOE considered in the supplemental model calculations (DOE, 2009df), the NRC staff notes that other radionuclide pairs could have significantly different activities than those assumed in the TSPA model, depending on the differences in effective K_ds between parent and daughter. For example, the NRC staff notes that the mean activity ratio of thorium and radium, assuming the use of selected, deterministic K_ds, would be on the order of 14 for all thorium/radium pairs (including Th-230/Ra-226, Th-228/Ra-224, Th-227/Ra-223, and Th-229/Ra-225) and not just for the Th-232/Ra-228 pair analyzed by DOE (2009de). However, many such pairs have nuclides that are extremely short-lived or have low inventories and have been reasonably excluded by DOE in their selection of radionuclides to include in transport modeling. Other nuclides such as Th-230 and Ra-226 are explicitly transported in the TSPA saturated zone abstraction, so their activity ratio correctly reflects the modeled sorption enhancement factor. Thus, the NRC staff concludes that the potential added contribution to dose from these other parent-daughter radionuclide pairs is not significant.

Second, the NRC staff identified that the inventory boosting method can underestimate or overestimate the amount of daughter in the saturated zone, depending on the transport characteristics of the parent and daughter. At early times in the saturated zone transport model, the daughter concentrations could be overestimated because DOE's inventory-boosting methodology assumes all of the ingrowth occurs when the parent radionuclide enters the saturated zone. At later times, the daughter concentrations may be underestimated because a quickly transported (low K_d) daughter may exit the model system before the parent would have actually generated the daughter via decay. To examine the significance of this consideration, the NRC staff conducted an independent analysis and compared one-dimensional transport simulations of the parents and first-generation daughters with radioactive decay and ingrowth to one-dimensional transport simulations of the parents and first-generation daughters with decay and inventory boosting instead of ingrowth, using deterministic values for K_ds that were selected to accentuate potential differences in transport behavior (Bradbury, 2010ab). The NRC staff finds that DOE's inventory-boosting method is acceptable because the results from the two types of simulation were comparable at later times when steady state was established.

Third, DOE's assumption of secular equilibrium is relevant to a source of uncertainty about the potential contribution of some short-lived radionuclides, including Po-210, an alpha-emitter with a half-life of about 138 days and a daughter of Pb-210 in the U-238 decay chain. There are some observations to indicate that Po-210 may not be removed by sorption from natural waters as readily as its immediate parent, Pb-210 (Serne, 2007aa; Hameed, et al., 1997aa; Outola, et al., 2008aa; Seiler, 2011aa; Seiler, et al., 2011aa). At several locations in the United States, Po-210 has been measured in groundwater at concentrations greater than would be expected based on measured Pb-210 concentrations (Seiler, 2011aa; Seiler, et al., 2009aa; 2011aa; Harada, et al., 1989aa). A reasonable explanation of these Po-210 excesses is that the specific geochemical conditions present in these locations leads to differential sorption of parent and daughter radionuclides. Although the behavior of polonium in groundwater is not fully understood, there are several geochemical parameters that appear common to locations where significant excess Po-210 has been recognized. Based on the NRC staff's review of scientific literature, conditions associated with excess polonium in groundwater include very low oxygen concentrations; microbial-mediated cycling of sulfur, manganese, or phosphorous, moderate to high organic carbon content; elevated measured concentrations of dissolved iron and manganese; and measured concentrations of iron oxide or manganese oxide colloids (Harada, et al., 1989aa; Burnett, et al., 1991aa; Upchurch, et al., 1991aa; Outola, et al., 2008aa; Seiler, et al., 2009aa, 2011aa; Seiler, 2011aa). In the Yucca Mountain saturated zone system, groundwater along the transport pathway and near the compliance boundary is characterized by oxidizing waters (elevated levels of dissolved oxygen) with low organic content; no evidence of microbial cycling of sulfur, manganese, or phosphorous; very low or non-measurable dissolved iron and manganese; and few or no iron oxide or manganese oxide colloids (SNL, 2007ax; SNL, 2007ba; BSC, 2004bc; Bertetti, et al., 2004aa). Thus, the NRC staff finds that the cited conditions, which would appear to favor excess Po-210, have not been observed along the potential flow path in the saturated-zone alluvium south of the controlled boundary. Moreover, the low-oxygen reducing conditions that may favor disequilibrium of Po-210 with respect to its precursors (Pb-210 and other U-238 decay-chain nuclides) would significantly limit the transport of the parent nuclides away from the repository (SNL, 2007ba). Consequently, the NRC staff concludes that excess Po-210, beyond the secular equilibrium values assumed by the applicant, is unlikely to occur in the saturated zone.

The NRC staff concludes that the DOE approach to radionuclide decay and ingrowth is adequate because DOE supporting analyses have identified that the uncertainties in radionuclide concentrations from the decay and ingrowth of parent-daughter pairs with differing transport properties are not significant, and that DOE has adequately incorporated potential temporal variations and uncertainties in groundwater chemistry in DOE's saturated zone transport model abstraction, as the NRC staff identified in SER Section 2.2.1.3.9.3.1. Additionally, the NRC staff's independent analyses confirm that the uncertainties in radionuclide concentrations from the potential accumulation of decay-chain parent radionuclides along the transport path are not significant for performance of the saturated zone.

2.2.1.3.9.4 Evaluation Findings

The NRC staff reviewed DOE's SAR and other information submitted in support of the license application, which includes information required by 10 CFR 63.21(c)(1), (9), and (15), and finds, with reasonable expectation, that the relevant requirements of 10 CFR 63.114 and 63.342 are satisfied regarding the abstraction of radionuclide transport in the saturated zone. In particular, the NRC staff finds that DOE

- Included field data related to the geology, hydrology, and geochemistry of the surface and subsurface from the site and the region surrounding Yucca Mountain to define parameters and conceptual models used in the performance assessment calculation, in compliance with 10 CFR 63.114(a)(1)

- Accounted for uncertainty and variability in the parameter values used to model radionuclide transport in the saturated zone, in compliance with 10 CFR 63.114(a)(2)

- Considered and evaluated alternative conceptual models that are consistent with currently available data and scientific understanding, in compliance with 10 CFR 63.114(a)(3)

- Provided a technical basis for the inclusion of FEPs affecting radionuclide transport in the saturated zone, consistent with the limits on performance assessment in 10 CFR 63.342, and evaluated in sufficient detail those processes that would significantly affect repository performance, in compliance with 10 CFR 63.114(a)(4-6)

- Provided technical bases for the models of radionuclide transport in the saturated zone used in the performance assessment to represent the 10,000 years after disposal, in compliance with 10 CFR 63.114(a)(7)

- Used performance assessment methods for the period of geologic stability consistent with the methods used to demonstrate compliance for the initial 10,000 years following permanent closure, in compliance with 10 CFR 63.114(b)

- Included in the performance assessment for the period of geologic stability those FEPs used for the initial 10,000-year period, consistent with specific limits, in compliance with 10 CFR 63.342(c).

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CHAPTER 13

2.2.1.3.10 Igneous Disruption of Waste Packages

2.2.1.3.10.1 Introduction

Safety Evaluation Report (SER) Section 2.2.1.3.10 evaluates the U.S. Department of Energy's ("DOE" or "applicant") models for the potential consequences of disruptive igneous activity at Yucca Mountain if basaltic magma rising through the Earth's crust enters repository drifts [DOE's igneous intrusion modeling case, Safety Analysis Report (SAR) Section 2.3.11.3 (DOE, 2009av)] or enters a drift and later erupts to the surface through one or more conduits (DOE's volcanic eruption modeling case, SAR Section 2.3.11.4). The proposed Yucca Mountain repository site lies in a region that has experienced sporadic volcanic events in the past few million years. DOE previously determined the probability of future igneous activity at the site to exceed 1×10^{-8} per year (SAR Section 2.2.2.2; CRWMS M&O, 1996aa; evaluated in SER Section 2.2.1.2.2). DOE therefore included igneous activity as one of three scenario classes in its performance assessment. Because basalt is the only type of magma that has been erupted in the past 8 million years in the Yucca Mountain region, DOE's performance assessment considers only basaltic igneous activity. As discussed in SER Section 2.2.1.2.2, the probability of more silicic (or rhyolitic, and therefore more explosive) igneous activity, of the type that produced extensive pyroclastic deposits in the area more than 10 million years ago, is well below 1 in 10,000 over 10,000 years. DOE screened such an event out as a potential disruptive event.

This SER Section evaluates the subsurface igneous processes DOE described (i.e., intrusion of magma into repository drifts, damage to waste packages and other engineered barriers, and formation of conduits to the surface, which potentially involves entrainment of waste into the conduit and toward the surface). DOE's models for volcanic ejection and dispersal of waste material into the surface environment are reviewed in SER Section 2.2.1.3.13. Together, SER Sections 2.2.1.3.10 and 2.2.1.3.13 evaluate DOE information and output used in the Total System Performance Assessment (TSPA) under the Igneous Scenario Class (see SER Sections 2.2.1.2.1, 2.2.1.2.2, and 2.2.1.4.1).

DOE examined the consequences of igneous disruption of the repository (Igneous Scenario Class) using the results of TSPA calculations through the two linked modeling cases: igneous intrusion and volcanic eruption (intrusion always precedes eruption). DOE's igneous intrusion modeling case provides TSPA parameter values for the number of waste packages failed (mass of waste) during an intrusive event, the temperature in the invaded drifts in the period after intrusion, and chemical changes to groundwater that may react with the basalt filling the drifts. The igneous disruption of waste packages abstraction integrates with other TSPA model components, such as the unsaturated zone radionuclide transport abstraction, and provides information about the flux of radionuclides released from the waste form into water entering the unsaturated zone after an intrusive event (SER Section 2.2.1.3.7). Exposure to radionuclides in groundwater extracted by pumping is one of the principal pathways for radiological exposure to the reasonably maximally exposed individual (RMEI).

In the DOE volcanic eruption modeling case, a key parameter affecting the overall radiological dose calculation is the number of directly affected waste packages and thus the amount of waste entrained in a volcanic eruption. This is also evaluated in this SER Section. DOE's model of the airborne transport and redistribution of radionuclides into soil includes the amount of waste erupted into the atmosphere, the amount deposited on the ground, and the redistribution of the waste-contaminated volcanic ash. This airborne transport and redistribution model is evaluated in SER Section 2.2.1.3.13 and provides information for the Volcanic Ash Exposure Scenario described in DOE's Biosphere Model (SAR Section 2.3.10). DOE's estimate of the annual probability of igneous events intersecting the repository (1.7×10^{-8} per year; SAR Table 2.3.11-4) is reviewed in SER Section 2.2.1.2.2 and discussed later in this SER Section. For these abstractions and the TSPA, DOE calculates probability-weighted results for both an intrusive-only dose and a total dose (intrusive plus volcanic) to the RMEI, which are evaluated in SER Section 2.2.1.4.1 and in the Risk Perspectives subsection in this SER, Section 2.2.1.3.10.3.1.

Igneous disruption models evaluated in this SER Section are the first in a sequence of models that track radionuclides released from the repository to the RMEI as a result of possible future igneous activity. Accordingly, the model abstractions evaluated in this SER Section serve as input to those reviewed in other SER sections, including those that examine the effects of potential igneous disruption of natural and engineered barriers in the subsurface repository (SER Section 2.2.1.3.2). DOE recognized that igneous events potentially have large consequences but a low likelihood (probability) of occurring in the future (SAR Section 2.3.11.1). Thus, DOE provided only a qualitative description of igneous effects on engineered system barrier capabilities in its demonstration of multiple barriers (SAR Section 2.1.1). Nevertheless, basaltic igneous activity represents a disruptive event that significantly degrades most of the capabilities of the engineered barrier system (SAR Section 2.1.2.2.5). To represent igneous events in the performance assessment, DOE removes the barrier capabilities of the waste package and drip shield and degrades the waste form, consistent with information provided in SAR Section 2.3.11. DOE further concluded in SAR Section 2.1.1 that igneous events will have limited effects on the upper and lower natural barrier systems because the possible igneous intrusive rock bodies have very small dimensions compared with the large volume of rock through which groundwater is flowing and, further, the zone of influence around the intrusions is limited (SAR Section 2.1.2.3.5; evaluated in SER Section 2.2.1.3.6 and in SER Section 2.2.1.3.10.3.2).

2.2.1.3.10.2 Regulatory Requirements

The applicant is required to provide information specified in 10 CFR 63.21(c)(1), (9), and (15) that is related to abstraction of igneous disruption of waste packages. The requirements in 10 CFR 63.114 (Requirements for Performance Assessment) and 63.342 (Limits on Performance Assessments) include postclosure performance assessment to demonstrate compliance with 63.113 (Performance Objectives for the Geologic Repository after Permanent Closure). Specific compliance with 63.113 is reviewed in SER Section 2.2.1.4.1.

The regulations for performance assessments in 10 CFR 63.114 require, in part, that a performance assessment

- Include data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the surface and subsurface from the site and the region surrounding Yucca Mountain [10 CFR 63.114(a)(1)]

- Account for uncertainty and variability in the parameter values [10 CFR 63.114(a)(2)]

- Consider and evaluate alternative conceptual models [10 CFR 63.114(a)(3)]

- Provide technical bases for either the inclusion or exclusion of features, events, and processes (FEPs) affecting igneous disruption of waste packages, including effects of degradation, deterioration, or alteration processes of engineered barriers that would adversely affect performance of the natural barriers, consistent with the limits on performance assessment in 10 CFR 63.342, and evaluate in sufficient detail those processes that would significantly affect repository performance [10 CFR 63.114(a)(4–6)]

- Provide a technical basis for the igneous disruption of waste packages model which, in turn, provides input or otherwise affects other models and abstractions used in the performance assessment to represent the 10,000 years after disposal [10 CFR 63.21(c)(9); 10 CFR 63.114(a)(7)]

The NRC staff's evaluation of the inclusion or exclusion of FEPs is given in SER Section 2.2.1.2.1. Requirements for performance assessments for the initial 10,000 years following disposal are set forth in 10 CFR 63.114(a). Regulations at 10 CFR 63.114(b) and 10 CFR 63.342 set forth requirements for the performance assessment methods for the time from 10,000 years through the period of geologic stability, defined in 10 CFR 63.302 as 1 million years following disposal. These sections provide that, through the period of geologic stability, with specific limitations, the applicant

- Use performance assessment methods consistent with the performance assessment methods used to demonstrate compliance for the initial 10,000 years following permanent closure [10 CFR 63.114(b)]

- Include in the performance assessment those FEPs used in the performance assessment for the initial 10,000-year period (10 CFR 63.342)

This model abstraction involves igneous activity, i.e. intrusive or volcanic disruption of waste packages. The requirements in 10 CFR 63.342(c)(1) pertain to the effects of seismic and igneous activity on the repository performance, subject to the probability limits in 10 CFR 63.342(a) and 63.342(b). Specific constraints on the seismic and igneous activity analyses are in 10 CFR 63.342(c)(1)(i) and (ii), respectively.

The NRC staff's review of the SAR and supporting information follows the guidance in the YMRP Section 2.2.1.3.10, Volcanic Disruption of Waste Packages, (NRC, 2003aa), as supplemented by additional guidance for the period beyond 10,000 years after permanent closure (NRC, 2009ab). The YMRP acceptance criteria for model abstractions provide guidance for the NRC staff's review of the applicant's abstraction of igneous disruption of waste packages are

1. System description and model integration are adequate
2. Data are sufficient for model justification
3. Data uncertainty is characterized and propagated through the abstraction
4. Model uncertainty is characterized and propagated through the abstraction
5. Model abstraction output is supported by objective comparisons

In its review of SAR and supporting information, the NRC staff used a risk-informed approach and the guidance in the YMRP, as supplemented by NRC (2009ab), for aspects of climate and infiltration important to repository performance. The NRC staff considered all five YMRP criteria in its review of information provided by DOE. In the context of these criteria, only those aspects of the model abstraction that substantively affect the performance assessment results, as determined by the NRC staff, are discussed in detail. The NRC staff's determination is based both on risk information provided by DOE and on the NRC staff's knowledge gained through experience and independent analyses.

2.2.1.3.10.3 Technical Review

DOE's analysis of FEPs considered ways that igneous activity could affect the proposed repository site. The NRC staff's evaluation of the applicant's FEP screening is in SER Section 2.2.1.2.1. DOE included the following FEPs and defined the igneous scenarios for the performance assessment (SAR Table 2.3.11-1): 1.2.04.03.0A, Igneous Intrusion into Repository; 1.2.04.04.0A, Igneous Intrusion Interacts with Engineered Barrier System Components; 1.2.04.06.0A, Eruptive Conduit to Surface Intersects Repository; and 1.2.04.04.0B, Chemical Effects of Magma and Magmatic Volatiles (SAR Table 2.2-5). This SER Section evaluates repository performance as affected by these FEPs.

The NRC staff's review is based on information presented in SAR Section 2.3.11 and relevant analysis and model reports (AMRs), material in other publicly available DOE and NRC reports, and relevant information published in peer-reviewed literature. The applicant also described and evaluated background information used to assess the likelihood and type of future igneous activity in the Yucca Mountain region in SAR Volume 1, General Information, and Volume 2, Section 1.1.2. That material is reviewed in SER Section 2.1.1.1.3.6 as part of the NRC staff's evaluation of the applicant's site characterization.

2.2.1.3.10.3.1 General Approach by DOE

Igneous activity can be solely intrusive (i.e., magma intruded into rocks below the Earth's surface) or can be both intrusive and extrusive (i.e., volcanic, in which, following intrusion, magma breaks through to the surface and erupts). The NRC staff notes that the terms "volcanic" and "intrusive" have sometimes been used interchangeably in the DOE license application and supporting documents. To avoid confusion, the NRC staff will refer to igneous activity that occurs beneath the Earth's surface as "intrusive" and activity above the surface as "extrusive" or "volcanic." All subsurface igneous processes that could disrupt the repository are considered intrusive and are reviewed in this SER Section, whereas the above-surface volcanic processes are evaluated in SER Section 2.2.1.3.13.

To evaluate the potential effect of future igneous activity on dose to the RMEI, DOE adopted a conceptual model in which rising basalt magma entering a repository drift (or drifts) could cause release of radionuclides via two pathways (SAR Section 2.3.11.1). The first pathway is intrusive igneous events, where magma rising toward the surface as a dike, or set of dikes, enters the drifts but stays beneath the surface. DOE also considered the other type of shallow-depth igneous intrusion, sills (relatively small subhorizontal igneous intrusions), but did not treat them separately in its analysis. One reason is that dikes must be present to feed magma into sills, and DOE showed that the consequences of intruding the repository by a sill would be similar to and more limited than by a dike. DOE also pointed to the relatively small size of sills in the Yucca Mountain region (Valentine and Krogh, 2006aa; Keating, et al., 2008aa) and thus did not include sills in its igneous disruption scenario for the repository (SAR Section 2.3.11.2.1.1). In the igneous intrusive scenario, DOE assumed that all drifts in the repository are intersected by the dike(s), magma fills all drifts, and all waste packages in the repository are damaged but remain in the drifts. No waste is released directly into the accessible environment in an intrusive-only igneous event, but radionuclides are released to the accessible environment through subsequent groundwater transport. DOE models this transport to occur through the same pathways represented in the nominal, seismic, and early failure scenario classes, which are evaluated in SER sections on unsaturated zone flow and transport (SER Sections 2.2.1.3.6, 2.2.1.3.7, 2.2.1.3.8, and 2.2.1.3.9).

The second pathway is an extrusive, or volcanic, igneous event. DOE considered a scenario where magma continues to rise to the surface as a dike after intersecting repository drifts and, on the basis of the behavior of basaltic eruptions in general, that surface activity along the resulting initial fissure (the surface expression of a dike) would rapidly localize, or focus, to a single, or few, points of magma effusion (SNL, 2007ae; SAR Section 2.3.11.4.1). A volcanic conduit wider than the dike would be expected to develop at that focus somewhere along the dike by excavation from the surface (vent) downwards. This conduit can potentially intersect a drift(s) or develop in the area between the drifts. Magma flow up a drift-intersecting conduit would entrain waste from disrupted packages, thereby providing a direct pathway for waste material to be released to the accessible surface environment during a volcanic eruption.

DOE explained that the volcanic (extrusive) part of the igneous scenario is an extension of the intrusive part (SAR Section 2.3.11.1) and concluded that every intrusive event that might intersect the repository is likely to have a conduit develop somewhere along one of the dikes, as described in SAR Section 2.3.11.2.1.2 and SNL Table 7-1 (2007ae). The conduit (or conduits) may, however, form outside the repository footprint or may not intersect a drift, and in that case, no waste material would be entrained into the magma that rises to the surface in an eruption. In effect, this would be equivalent to the intrusive-only case. In addition, DOE determined that conduits that might feed surface volcanoes may only develop along specific parts of dikes (SAR Section 2.3.11.4.2.1) and thus concluded that the probability of a volcanic event occurring at the repository is expected to be lower than the probability of an intrusive event. DOE also concluded that if an eruption that entrained waste material and transported it into the surface environment did occur at the repository, the potential doses to the RMEI location from radionuclides released through the intrusive and extrusive pathways would be additive. Further details of conduit development are evaluated in the NRC staff's review of the volcanic eruption modeling case (SER Section 2.2.1.3.10.3.3).

2.2.1.3.10.3.2 NRC Staff's Review of DOE Igneous Intrusion Modeling Case

After review of the applicant's material in SAR Chapter 2.3.11, as outlined above, NRC staff has determined that the DOE model for igneous intrusion and its effect on repository performance rely on four key conclusions:

1. Magma rising as a dike beneath repository drifts will intersect and flow into the drifts.

2. Any dike intersection into the repository footprint floods all drifts with magma, causing engineered barrier system components, including all waste packages and drip shields, to fail while magma and waste remain in the drifts.

3. Igneous intrusion does not alter the ambient hydrologic flow and transport regime significantly (i.e., the natural barriers above and below the drifts are not affected).

4. Subsurface conduits that develop beneath volcanoes can be represented by cylinders and only entrain waste within the part of the cylinder that intersects the drift.

Conclusions 1–3 solely concern the intrusive model case, while Conclusion 4 is also applicable to the volcanic model case. The NRC staff's review focuses on the risk-significant aspects of these conclusions. The NRC staff's overall risk perspective for these abstractions is given in the next subsection, and the specific technical aspects are evaluated under the subsequent subsections.

NRC Staff's Perspective on Risk

NRC staff has assessed the risks caused by an igneous event at the proposed repository on the basis of the applicant's information and the NRC's independent analysis. As stated in the Introduction (SER Section 2.2.1.3.10.1) of this section, while the probability of an igneous event is low, the consequences could be high. The igneous intrusion modeling case would constitute most of the calculated dose to the RMEI for the first 1,000 years following permanent closure of the repository, as shown in SAR Figure 2.4-18(a), and is approximately half the DOE calculated dose for the seismic ground motion modeling case in the ensuing 9,000 years.

In SAR Section 2.4.2.2.1.2.3, DOE provided its estimate of probability-weighted consequences of igneous activity (intrusive and extrusive) using the probability distribution from its expert elicitation for a Probabilistic Volcanic Hazard Assessment (PVHA). DOE identified that the probability-weighted igneous mean intrusive dose is estimated to be less than 0.001 mSv/yr [0.1 mrem/yr] for the 10,000-year period and the median dose less than 0.005 mSv/yr [0.5 mrem/yr] for the post-10,000-year time period (SAR Section 2.4.2.2.1.2.3.1). DOE estimates for the probability-weighted igneous extrusive (volcanic eruptive) mean dose alone are on the order of 10^{-6} mSv/yr [0.0001 mrem/yr] for the 10,000-year period and the median dose is less than 6×10^{-7} mSv/yr [6×10^{-5} mrem/yr] for the post-10,000-year time period (SAR Section 2.4.2.2.1.2.3.2). The NRC staff notes that the difference in magnitude for the dose consequences between the two igneous scenarios (intrusive and extrusive) predominantly results from the different number of waste package failures estimated to occur for each scenario, which causes the dose from the extrusive case to be several orders of magnitude below the intrusive case (SAR Section 2.2.1.4.1; evaluated in SER Section 2.2.1.3.10.3.3.

Effects of Igneous Intrusion on Performance of Natural Barriers

Because DOE did not rely on an evaluation of igneous events in its demonstration of multiple barriers, the NRC staff does not include a discussion of igneous events in SER Section 2.2.1.1. DOE screened out of its performance assessment the effect of igneous dikes and sills on groundwater flow and transport pathways surrounding drifts in the upper and lower natural barriers, as described in SNL (2008ac) (FEP 1.2.04.02.0A). At the drift wall, however, DOE included the effect of igneous intrusions, by assuming the drifts become degraded and the groundwater seepage barrier is eliminated. For this evaluation of the performance assessment, seepage is set equal to percolation. Igneous activity near repository drifts may alter hydrologic properties of the host rock or cause perching of water in the unsaturated zone. DOE's sensitivity analyses indicate that these effects on unsaturated zone flow in repository performance are small (FEP 1.2.04.02.0A; SNL, 2008ac). In particular, the potential effect of increased fracturing in and around a dike providing preferred water pathways has relatively little impact, given the predominance of fracture flow in the existing, undisturbed unsaturated zone beneath much of the repository footprint (see the NRC staff's review of unsaturated zone flow in SER Section 2.2.1.3.6). Farther away from the repository (in the far field, as defined in SER Section 2.2.1.3.6), igneous dikes and sills may modify saturated zone flow and plume pathways, but again, DOE found these effects were minor for performance (FEP 1.2.04.02.0A; SNL, 2008ac).

The NRC staff reviewed relevant information in the SAR (Section 2.3.11.2.1.1), in SNL (2008ac), and in SNL (2007ag) and concludes that the DOE treatment of igneous activity in the form of dikes and sills is adequate for performance assessment with respect to groundwater flow in the natural barriers on the basis of the following rationale. For dikes and sills that

- Intersect drifts, the seepage barrier is eliminated from DOE's performance assessment model

- Occur above the repository, any potential increases in focusing of groundwater flow that might be caused by the presence of intrusions (dikes and sills) are not important because NRC concluded in SER Section 2.2.1.3.6.3.2 that uncertainty in spatial variation in groundwater percolation (flow) is unimportant to performance

- Occur below the repository in the unsaturated zone, sensitivity analyses for groundwater transport that included potential changes to hydrologic properties of fractures and matrix lead to a smaller effect than that already considered for the uncertainty in net infiltration (FEP 1.2.04.02.0A; SNL, 2008ac)

- Occur in the saturated zone, the area the igneous activity and resulting rock bodies can potentially modify is small compared with the size of the saturated flow zone, with typically only a few decimeters [4–12 in] of disrupted rock around the ~1-m [3 ft]-wide dikes (SAR Section 2.3.11.3.2; SNL, 2007ag; Detournay, et al., 2003aa; Keating, et al., 2008aa).

In the igneous intrusion abstraction, DOE ignored sills for the reasons given in SER Section 2.2.1.3.10.3.1, and also because DOE's performance assessment assumes that a dike intruding the repository would result in failure of all waste packages and drip shields. The NRC staff concludes, on the basis of SER Section 2.2.1.3.10.3.1, that a potential sill would intersect fewer drifts than a dike swarm, and, therefore, DOE's dike model encompasses the potential consequences of sills. Thus, the NRC staff finds that DOE's approach to sills is acceptable because its igneous intrusive scenario adequately captures the potential impacts on repository performance.

Behavior of Intruding Magma in Drifts and Effects on the Engineered Barrier System

In developing the model for subsurface igneous processes, DOE concluded that basaltic magmas in the Yucca Mountain region would contain appreciable amounts of dissolved volatiles, primarily water (SAR Section 2.3.11.2.1.2; Nicholis and Rutherford, 2004). This dissolved water would become gaseous (i.e., degas from the magma) as pressure on the magma becomes lower due either to normal ascent toward the surface or by intersection with a repository drift. Significant amounts of gas expansion in the upper 300 m of rise [above ~1,000-ft depth] would cause magma in potential igneous events to flow more rapidly, and perhaps more extensively, than would be expected for magmas with little gas-driven expansion. In part because of the relatively high dissolved water contents expected for Yucca Mountain basaltic magmas, DOE concluded that all repository drifts would be rapidly filled by magma flow if an intrusive igneous event occurred within the repository footprint (SAR Section 2.3.11.2.1.2).

The NRC staff reviewed the information DOE presented to support the conclusion that basaltic magmas are expected to have relatively high dissolved water contents. The NRC staff finds that the information DOE cited supports this conclusion (discussed in SAR Section 2.3.11.3.2.3), and, in addition, the presence of hydrous minerals in some Yucca Mountain region basaltic rocks supports the DOE conclusion that dissolved magmatic water contents in these magmas are relatively high (Nicholis and Rutherford, 2004aa; SNL, 2007ae). The NRC staff concludes that uncertainties in estimates of the dissolved volatile content of these basalts do not affect performance significantly, given that DOE assumes that magma behavior at repository depths is driven by an exsolved (degassed) gas phase, and that, upon reaching a repository drift, the magma could release gas vigorously.

In the intrusive igneous case, DOE assumes that if a single rising dike intersects any part of the repository footprint where drifts containing waste packages are located, then all drifts in the repository are rapidly filled with magma. DOE developed this approach to account for the uncertainties in determining the physical characteristics of dikes at repository depths, and for uncertainties in magma flow processes in drifts intersected by dikes (SAR Section 2.3.11.3.1). For the ascending magma entering the drifts, DOE recognized that there are two possibilities for flow behavior, considering how rapidly and violently magma could enter a drift. The less rapid possibility is effusive and has a lava-like flow, while the other is more explosive, resulting in a fragmental, or pyroclastic flow (SAR Section 2.3.11.2.1.2; SNL, 2007ag; Woods, et al., 2002aa; Dartevelle and Valentine, 2005aa, 2009aa). The NRC staff also conducted independent confirmatory analyses (Woods, et al., 2002aa; Lejeune, et al., 2009aa) verifying that potential magma flow into drifts could occur quickly enough so that only minor cooling of the magma would occur. On the basis of the results of these independent studies and its own evaluation, the NRC staff concludes that DOE has developed an acceptable technical basis to propose that all drifts will be filled with basaltic magma if an intrusive igneous event occurs at the repository site, as further discussed in this section. This approach involves the disruption of all waste packages stored in the proposed repository (SAR Section 2.3.11.3). The NRC staff further concludes that this does not underestimate risk and that there are no technical uncertainties in this conclusion that could reasonably increase the DOE risk estimates.

According to the applicant's calculations, after intersection and intrusion by magma, drift temperatures are modeled at or near magmatic temperatures of 1,046–1,169 °C [1,915–2,136 °F], at which point plastic deformation of the waste packages begins. Additional DOE analysis showed waste packages could also be damaged by magmatic pressures as low as 4 MPa [580 lb in^{-2}]. DOE concluded waste package failure could result in waste forms that are exposed to high temperatures and that undergo chemical reactions with magma and its constituents. DOE assumed that the packages would encounter additional mechanical loads from the cooling and solidification of enveloping magma. Already weakened by the thermal effects of the magma, the mechanical loads associated with the magma would result in deformation of the waste package. DOE proposed that similar effects would occur for drip shields exposed to magmatic conditions. Thus, DOE concluded that uncertainties associated with the potential effects of magma on waste package and drip shield performance supports the assumption that all waste package and drip shield barrier capabilities are removed in models for igneous intrusive events (SAR Section 2.3.11.3.2.4). NRC staff notes that such processes are unprecedented in nature, but as the DOE model does not underestimate risk, the NRC staff finds it is adequate for modeling in the TSPA. DOE also concluded that exposure to magmatic conditions will result in unprotected waste forms that are, effectively, instantaneously degraded, such that radionuclides are assumed to be immediately available for hydrologic transport, as soon as the intrusive basalt rock is cool enough to allow water to contact waste (SAR Section 2.3.11.3.2.4), which the NRC staff also finds acceptable because it does not reduce potential dose. As discussed in the next subsection, the cooling time of the intrusion is short, relative to the time scale of groundwater percolation and flow, and relative to the regulatory period for postclosure repository performance. DOE further concluded that although the waste packages no longer serve as a barrier to water flow after an igneous event, corrosion products from degradation of waste package materials will be present and will strongly retard release of certain radionuclides into the unsaturated zone in the same manner as in the nominal scenario. The NRC staff evaluates the role of corrosion products in radionuclide release after an igneous intrusion scenario in SER Section 2.2.1.3.4 and concludes that it is not significant to dose.

The NRC staff reviewed the information DOE provided in SAR Section 2.3.11.3.2 to support the representation of engineered barrier and waste form responses to potential igneous intrusive events. The NRC staff concludes that the DOE approach for modeling waste package and drip shield response to potential intrusive events is acceptable because this approach is consistent with available information on the heat of the magmatic intrusion and possible behavior of the proposed type of waste packages, drip shields, and waste form. Further, given that DOE's analysis involves failure of the engineered barriers for all waste packages, NRC staff has identified no reasonable alternatives to this approach that would result in higher calculated doses. The NRC staff concludes that the DOE representation of waste form response as instantaneously degraded with all radionuclides available for subsequent hydrologic transport is acceptable, as this approach is consistent with available information, and there are no reasonable alternatives that would result in higher calculated doses.

Because DOE assumes that all waste package and drip shield capabilities cease during an igneous intrusive event, there are few uncertainties that are significant to the evaluation of potential intrusive igneous event dose consequences. Those that DOE proposed to have potential significance to dose are discussed in the next two sections. The NRC staff has determined, by reviewing DOE-provided information, that the physical conditions following magma intrusion into a drift could affect subsequent hydrologic (groundwater) flow and transport processes. Thus, the remainder of this SER section evaluates the DOE basis for calculating the effects of magma cooling on the drift environment and subsequent hydrologic flow and transport. While not important to dose, these evaluations are included because they are outputs to the DOE TSPA.

Magma Cooling and Heat Flow to Host Rock

The temperature in the drifts after magma intrusion is an output parameter to TSPA (SAR Section 2.3.11.6.7). This subsection evaluates the DOE estimates of centerline and wall temperature in the invaded drifts in the period after intrusion and the timing of the intrusive event with respect to the repository life cycle, reflecting the temperature of the host rocks during the period of heating by radionuclide decay. DOE included these temperatures for consideration of the post-intrusion environment in the damaged drifts and the period of time after which groundwater seepage through the drifts could return (i.e., after the temperatures fall below the boiling point of water).

The temperature in the drifts after magma intrusion provides an estimate of the cooling time of the basalt inside the drift. The cooling of the basalt inside the drift and the drift centerline temperature, as well as drift-wall temperature, also influence the spent fuel dissolution model and the calculation of diffusion coefficients. Diffusion coefficients are used to calculate near-field contaminant transport in the unsaturated zone rock.

DOE concluded that a short-lived (hours to days) intrusive event, as in SNL (2007ab, Figure F–1), would fill every drift in the proposed repository with basaltic magma at a temperature of approximately 1,100 °C [2,012 °F], with an upper bound of 1,200 °C [2,192 °F] (SAR Section 2.3.11.3.2.4). Following the intrusive event, the magma in the drifts begins to cool. DOE performed numerical simulations to model the magma cooling and heat flow in the rock between drifts, via non-steady-state heat conduction, with radial flow of heat from the magma-filled drifts into the host rock. DOE's model considers a single basalt-filled drift, recognizing that the heat from one 5-m [16-ft]-diameter magma-filled drift will not influence the next drift approximately 80 m [262 ft] away (SNL, 2007ag). The calculated temperature decreases with time and distance from the centerline of the drift. DOE calculated the thermal diffusivity of the rocks using the rock volumetric heat capacity and the thermal conductivity of the welded tuff at the repository horizon; DOE assumed that the thermal diffusivity of the welded tuff and the basaltic magma or rock would be the same (SNL, 2007ag, p. 5-3), which NRC staff finds acceptable because the thermal diffusivity of all rocks falls into a narrow range. DOE also tested a range of host rock states in the heat-flow calculation represented by either completely dry or completely wet. The drift wall temperature prior to magma intrusion in the DOE model runs was between 25 and 200 °C [77 and 392 °F]. DOE concluded that this range suitably represents temperatures at different times for the intrusive event (reflecting elevated repository temperatures for several thousand years after closure [e.g., see SAR Figure 2.3.5-33 for calculated repository drift-temperature decay curves]. These temperature distributions provided the DOE estimate of the cooling rate and thermal history of the repository drifts following an intrusive event.

DOE explained that the model does not include the effects from the latent heat of magma crystallization or the property contrasts between the magma and the tuff. Without latent heat effects, the one-dimensional model results underestimated peak temperatures and time needed for cooling. Therefore, DOE considered alternative models, including an analytical solution that approximated the effects of latent heat and numerical solutions in two dimensions that included both latent and radioactive heat. Noting that latent heat would be liberated during magma crystallization and that its effects would be most pronounced at very early times while the magma is still partially liquid, the applicant accounted for the effect of latent heat by increasing the initial temperature of the magma.

DOE considered the main uncertainty when modeling magma cooling and solidification to be the initial magma temperature. For dry magma, 1,150 °C [2,102 °F] was used, but the NRC staff notes that magma with a high water content could have a temperature as low as 1,046 °C [1,915 °F] (Nicholis and Rutherford, 2004aa). Although the difference is small, this lower starting temperature could slightly reduce the time required for the magma to cool to a solidified rock, which the NRC staff finds reasonable. However, as noted previously, DOE assumed a higher initial magma temperature instead of explicitly including the latent heat of magma crystallization. Other uncertainties the applicant considered included thermal conductivity, grain density, specific heat capacity, matrix porosity, saturation, and the lithophysal porosity of the host tuff, but DOE found the effect of variations of these properties to be small. Heat loss was modeled as purely conductive, as DOE did not expect convection to occur in stagnant magma within drifts. For an igneous intrusion event occurring after about 1,000 years into the postclosure period, DOE concluded that the repository drift walls would attain a temperature of 100 °C [212 °F] about 100 years after the intrusive event occurs, as in SNL Figure 2.3.5-33 (2008ag).

DOE showed that drift temperatures in the 100-year post-intrusion period abstracted to the total system performance assessment have little influence on the dose estimated from the intrusive scenario. The NRC staff finds this acceptable because DOE's adopted scenario involves the disruption of all waste in the repository and potential changes to dose from temperature increases due to magmatic heat are short lived (less than or equal to 100 years) compared with the time scale of groundwater percolation and flow, and relative to the regulatory period for postclosure repository performance; it is also acceptable because, as explained in the next subsection, some radionuclides that are important to dose have an inverse solubility relationship with temperature. The NRC staff also finds that DOE's conclusion that cooling basaltic magma would not significantly affect drift characteristics that are relevant to drift degradation is acceptable, again because DOE's adopted scenario involves the disruption of all waste in the repository and the calculated dose is significantly lower than the stipulated limit (SER Section 2.2.1.4.1). DOE used the final temperature of the drift and the cooled basalt temperature as an input to calculate the spent fuel dissolution model and diffusion coefficients. While the latent heat of crystallization would result in a slightly longer cooling time for the basaltic magma while it was still partially liquid, the NRC staff finds that this would be offset by the DOE assumption of a higher {by up to 50 °C [122 °F] } initial magma temperature. However, DOE concluded that an extended magma cooling time would have little influence on the dose estimated from the intrusive scenario, and on the hydrologic flow and transport after an igneous event, which the NRC staff also finds acceptable because of the short-lived cooling time for an intrusion (on the order of 100 years) compared with the much greater time scale for groundwater flow (as discussed in SER Section 2.2.1.3.6).

Percolation Flux Through Cooled Basalt

Chemical changes, expressed as the pH and ionic strength in groundwater that may react with the new basalt rock filling repository drifts after an intrusive magmatic event, comprise an output parameter to the TSPA model (SAR Section 2.3.11.6.7; SNL, 2005ae). This subsection evaluates DOE estimates of possible chemical changes that might occur to groundwater as it begins to seep through and possibly react with cooling and cooled basalt filling the drifts (SER Section 2.2.1.3.3).

In considering percolation of groundwater through the drift after an igneous intrusion into the repository, DOE assumed that solidified basalt rock in the drift has the same fracture, porosity, and permeability characteristics as the surrounding tuff. DOE also concluded that the newly introduced basalt rock could affect the chemistry of water that seeps into the drift; in particular, pH and ionic strength. To examine possible changes in these two chemical parameters of the seepage water, DOE selected for numerical analysis three groundwater samples from large, fractured basalt-hosted reservoirs and conducted an extensive literature review of the chemistry of basalt-hosted waters to provide a range of pH and ionic strength values, as described in SAR Sections 2.3.7.5.3.1 and 2.3.11.3.2 and SNL Section 4.1.2 (2007ae). Temperature can affect the pH of incoming fluids, so to avoid underestimating radionuclide solubilities, DOE calculated the parameter values at 25 °C [77 °F], rather than at higher temperatures that would have resulted in lower solubility limits (radionuclides of concern show retrograde or inverse solubility in this pH range) and therefore smaller mass releases.

As discussed in the previous subsection, for an igneous intrusion event occurring approximately 1,000 years into the postclosure period, water seepage and flow through the host rock mass is estimated to resume about 100 years after an intrusion occurs. This is equivalent to the time when the basalt in the drifts would reach ~100 °C [212 °F] along the drift centerline (SAR Section 2.3.11.3.3.8; SNL, 2008ag). This time also corresponds to when the repository drifts' walls are assumed to cool below the local boiling temperature, as shown in SNL

Figure 2.3.5-33 (2008ag). DOE modeled resumption of groundwater percolation through the invaded repository drifts and failed engineered barriers. DOE did not model release of radionuclides in gaseous form from the waste packages, because DOE's analyses indicate that this does not influence the final dose at the receptor (SNL, 2008ag). This is why groundwater percolation was the only pathway DOE considered for release of radionuclides. The NRC staff finds these conclusions (that the gaseous releases from a potential intrusive event would have very limited impact on overall calculated dose and that the groundwater pathway would be dominant) to be acceptable because the effect, in terms of volume of material, is small compared to that involved in groundwater percolation.

DOE's review and analysis of relevant information on basalt-hosted groundwater, as a proxy for water entering a cooled, intruded drift, showed that pH and ionic strength of water prior to entering basalt reservoirs, as well as variations within the actual composition of the basalt, are likely to have little effect on the pH and ionic strength of the water exiting the invaded drifts. DOE's sensitivity analyses using waters from basaltic aquifers also showed that the liquid influx composition has an insignificant effect on the in-package chemistry model estimates. DOE analyses using compositions of waters equilibrated with the ambient-temperature Columbia River Plateau basalts and Iceland basalts also found that the pH and ionic strength of the incoming water would have little influence on the resulting pH and the ionic strength of water passing through a basalt-filled repository.

The NRC staff finds that the chosen basalt compositions encompass a wide range of basalt types that include the characteristics of basalts expected in an igneous event at Yucca Mountain, and thus inform a satisfactory proxy for expected groundwater compositions for outgoing (effluent) groundwater flow after passing through basalt-filled drifts from an intrusive event at the proposed repository. This is because the basalt rock types in DOE's analyses encompass the same compositional ranges as expected for a future basaltic igneous event at Yucca Mountain (SNL, 2007ae). Thus, the NRC staff finds it acceptable that DOE's analyses of uncertainties associated with the expected composition of a future repository-filling basalt would not significantly affect the chemical composition of the effluent groundwater (SAR Section 2.3.11.3.3.9). This finding supports the DOE conclusion that the modeled changes in groundwater chemistry after contact with a basalt-filled drift would be negligible. This conclusion follows from the relatively small volume of intruded basalt in comparison with the volume of host rock. Moreover, even following an intrusive igneous event, the effluent groundwater composition would be dominantly controlled by reaction with the contents of the failed waste package (i.e., spent fuel or high-level waste glass, and corrosion products from internal components), rather than the intruded basalt, as described in SNL (2007ae, Section 6.8.10).

The NRC staff's review of release and transport of radionuclides following a possible igneous intrusion is further detailed in SER Section 2.2.1.3.4 (specifically, Sections 2.2.1.3.4.3.2 on waste form degradation and 2.2.1.3.4.3.4 on colloid formation and stability).

Summary and Findings of NRC Staff's Review of the Igneous Intrusion Modeling Case

The NRC staff concludes that DOE has provided sufficient information to support the modeling approach used to represent intrusive igneous events in the performance assessment of the proposed repository. The NRC staff concludes, on the basis of its evaluation of DOE-provided information and its own independent evaluation, that DOE has adequately considered how cooling basaltic magma would affect the characteristics of the invaded and disrupted repository drifts that are relevant to hydrologic flow and transport and to the calculated dose. The DOE modeling approach relies on the assumption that intersection of any igneous intrusive feature into the repository footprint fills all of the repository drift with basaltic magma and that the magma removes all barrier capabilities from all waste packages and drip shields in all drifts. The NRC staff finds this assumption acceptable because it does not underestimate the risk from a potential intrusive igneous event. Moreover, the NRC staff finds that DOE has evaluated the uncertainties associated with this assumption that could increase the DOE dose estimate. Therefore, NRC staff concludes that DOE has acceptably represented the potentially significant effects of igneous intrusive events in the performance assessment. The igneous intrusion case provides insight into repository performance in the case of failure of engineered barrier components (drip shields and waste packages). The model shows that the technical basis for the capability of those barrier components is based on, and consistent with, the technical basis for the performance assessments (reviewed in SER Section 2.2.1.4.1) used to demonstrate compliance with 10 CFR 63.113(b) and (c).

2.2.1.3.10.3.3 NRC Staff's Review of DOE Volcanic Eruption Modeling Scenario

DOE proposed that in some potential igneous intrusive events that intersect the repository footprint, a rising dike would reach the surface and develop a conduit at a location along the intrusion, and that magma would be extruded. If a conduit is located wholly or partially in a repository drift, waste from disrupted waste packages could be entrained by magma flow up the conduit and erupted from a volcano at the surface. Compared with the intrusion scenario, in which the contents of all waste packages in the repository are made available for hydrologic transport, DOE concluded that, for the volcanic scenario, only a limited amount of high-level waste could be entrained directly into a conduit or conduits (SAR Section 2.3.11.4), as explained next.

In the type of basaltic volcanic activity DOE predicted for the case of an eruption through the proposed repository, a dike reaches the surface and activity begins along a fissure (an elongated system of vents, which is the surface expression of the dike; see SAR Sections 2.3.11.2.1 and 2.3.11.4.1.1 and SAR Figure 2.3.11.5). In DOE's model, magma flow to the surface in the dike usually localizes to a single, or a few, points over a period of hours to a few days, as observed at past basaltic eruptions and as previously discussed in SER Section 2.2.1.3.10.3.2. Such behavior was seen in analogous historic events [e.g., the 1943–1952 eruption of Paricutín in Mexico, the 1973 Heimaey eruption in Iceland, and the 1975 Tolbachik eruption in Kamchatka (Pioli, et al., 2008aa; Thorarinsson, et al., 1973aa; Doubik and Hill, 1999aa)]. DOE studies of igneous products exposed in the rock record also inferred a similar progression for some prehistoric basaltic eruptions (e.g., SAR Section 2.3.11.4; SNL, 2007ae; Valentine, et al., 2006aa; Keating, et al., 2008aa). At this point in the modeled eruption, a conduit is considered to develop below the point of localization, with the main vent at the surface. This conduit and vent system feeds an explosive and lava-flow-forming Strombolian-style eruption. DOE adopted a violent Strombolian style for the entire model eruption considered on the basis of the characteristics of the young Lathrop Wells scoria cone near Yucca Mountain (see SER Section 2.2.1.2.2). DOE recognized that conduits grow (widen) downwards from the surface in the plane of the dike, as detailed in SAR Section 2.3.11.4.2.1.2 and SNL p. 6-46 (2007ae), and thus, in DOE's repository-disruption scenario, intersect a drift through the top of the drift.

DOE characterized subsurface volcanic conduits as flaring inward down from the top of the surface vent, such that conduit diameters at repository depths will be smaller than those observed near the surface. DOE characterized the size and shape of conduits using studies at exposed local analogous volcanoes (SAR Section 2.3.11.4 and Figure 2.3.11-6;

SNL, 2007ae; Valentine, et al., 2006aa; Keating, et al., 2008aa) and theoretical considerations and model studies (e.g., Wilson and Head, 1981aa; Valentine, et al., 2007aa). In the performance assessment, DOE represents subvolcanic conduits as simple cylinders (SAR Section 2.3.11.4.1). DOE used the area of the conduit that intersects a drift to calculate the mass of waste that the conduit entrains. DOE concluded that entrained waste is mixed uniformly in the volume of magma that is subsequently erupted at the surface. From a risk perspective, the DOE performance assessment calculates that the expected annual dose from the igneous volcanic modeling case alone is approximately 0.1 percent of the dose calculated for the intrusive scenario (SNL, 2007ag). This difference between the volcanic and intrusive scenarios arises, in part, because DOE concluded that the volcanic scenario entrains and erupts approximately 0.1 percent of the amount of high-level waste that is disrupted during the intrusive case. The NRC staff finds this conclusion to be acceptable, as detailed next. Thus, the NRC staff's review of the subsurface processes associated with the volcanic case focuses on the DOE basis for concluding that a volcanic conduit, or conduits, would entrain a limited amount of waste.

Development of Conduits and Likelihood of Ejecting Waste in a Volcanic Eruption

In the DOE-developed model, one to three eruptive conduits may occur along the thickest dike. DOE treats the predicted location of a single conduit along a dike, the most likely occurrence, as random (SAR Section 2.3.11.4.2.1). In SAR Section 2.3.11.2.2, DOE developed a basis to determine the likelihood that at least one conduit will form through the repository footprint and, more specifically for risk significance, through an emplacement drift containing waste packages if a dike intersected the repository. The DOE model for conduit formation is based on observations at basaltic volcanoes and supported by calculations constrained by information obtained from studies of analogous eroded volcanoes (SNL, 2007ae).

On the basis of observations of Quaternary volcanoes in the Yucca Mountain region, where in most cases only one volcano developed along a dike (Keating, et al., 2008aa), to treat uncertainty, DOE heavily weighted the distribution of the likely number of conduits that might develop along a dike toward one conduit per eruption (SAR Section 2.3.11.2.1.2; SNL, 2007ae). DOE determined that the presence of repository drifts would not affect the rise of a dike, nor subsequent eruptive processes, because the drifts would be negligible in volume compared to the volume of rock the dike transects. The NRC staff finds this an acceptable assumption, given the expected small dike size, on the order of 1 to 2 m [3 to 6 ft] wide below repository depths (Keating, et al., 2008aa), and the energy of a propagating dike. DOE determined that 85 percent of past eruptive events have formed a single conduit, 10 percent formed 2 conduits, and 5 percent formed 3 conduits, and thus concluded, based on the typical lengths of dikes in the region, that multiple conduits should be spaced between 0.4 and 2 km [0.25 and 1.2 mi] apart. DOE also considered five alternative conceptual models to represent the location of a conduit along a dike. On the basis of field analogs, models, and studies presented in SNL (2007ae,ag), DOE concluded that a model with random conduit locations along an existing dike is the only supportable approach, which NRC staff finds acceptable because conduits do not have any predictable location along surface expressions of dikes in analogous examples (Doubik and Hill, 1999; Hill and Connor, 2000aa; Valentine. et al., 2006aa; Valentine and Krogh, 2006aa; Keating, et al., 2008aa).

To calculate the likelihood that at least one volcanic conduit will form through an emplacement drift and entrain waste, DOE used numerical models to simulate the number of dikes that could penetrate the repository footprint using dike characteristics from CRWMS M&O (1996aa). For each simulation, DOE calculated the length of the dike, or dikes, located inside and outside the repository footprint and found there was a 60 percent chance that more than one dike would form in an intrusion and extrusion event. For the widest dike in each simulation, DOE constrained its model to form one to three conduits at random locations along that dike and determined whether this location coincided with the repository footprint (SAR Section 2.3.11.4.2.1.3; SNL, 2007ar). Using this approach, DOE estimated that there was a 20 to 35 percent chance, with a mean of 28 percent, that at least one conduit would form within the repository footprint. This value reflects the relatively small size of the repository footprint in comparison with the total area that dikes could impact (SAR Section 2.3.11.2.2). On the basis of alternative volcanic event characteristics and behavior, DOE acknowledged that the conditional likelihood of at least one eruptive center (conduit) within the repository footprint might range from 43 to 78 percent (SAR Section 2.3.11.2.2.6). However, DOE concluded that, on the basis of features of Yucca Mountain area volcanoes, a mean conditional eruption probability of 0.28 (28 percent) times the probability of dike intersection with the repository footprint was most consistent with basaltic volcanic events that are expected to include multiple dikes and in which conduit(s) form on the widest dike. On this basis, the NRC staff estimates that the mean conditional probability of a conduit forming within the repository, using the mean intrusive probability from the PVHA expert elicitation of 1.7×10^{-8} per year (SAR Section 2.3.11.4.2.1), is 4.8×10^{-9} per year ($1.7 \times 10^{-8} \times 0.28$).

The 28 percent conditional factor DOE provided is for a conduit that develops within the repository footprint, but which may not necessarily eject waste. DOE then developed a second conditional probability, 0.296 (the NRC staff rounded this to 0.3, or 30 percent), to represent the fraction of conduits within the repository footprint that may actually intersect a drift containing waste packages and eject the waste contents through a volcanic vent (SAR Section 2.3.11.4.2.1). This factor accounts for the spatial distribution of waste emplacement drifts within the repository footprint area and the likely orientation of dikes.

The NRC staff reviewed DOE information regarding the likelihood for conduit development at repository drifts. From studies of the characteristics of basaltic volcanism at the Yucca Mountain region and elsewhere (Hill and Conner, 2000aa; Doubik and Hill, 1999aa) and DOE and independent confirmatory and external studies of conduit development in basaltic volcanism (BSC, 2003ab; Detournay, et al., 2003aa; Pioli, et al., 2008aa), the NRC staff concludes that DOE acceptably characterized the number and spacing of volcanic conduits. The NRC staff finds that the DOE conclusion that the processes leading to conduit development along a dike are reasonably represented as randomized along the widest dike segment. This is acceptable because there is no predictable pattern controlling conduit formation at other analogous basaltic volcanoes. The NRC staff reviewed the DOE methodology that developed the 28 percent factor for conduit development in the repository and the 30 percent factor for conduit intersection with a drift. The NRC staff concludes that DOE acceptably implemented randomized conduit development in developing these factors, and that even if the conduit development factor was significantly higher, the implied risk would change by only a small amount (e.g., using a factor of 100 percent would increase the amount of waste disrupted and ejected to ~0.3 percent of that disrupted in the intrusive case, versus the predicted value of 0.1 percent). Given the relatively small volume and rapid infilling time of the intersected drifts, the NRC staff concludes that the presence of repository drifts will not significantly affect the localization process for conduit development. Thus, the NRC staff concludes that DOE has acceptably evaluated the likelihood of conduit development at intersected drifts.

Eruptive Conduit Growth and Size, and Impact on Waste Packages and Waste

According to DOE's scenario presented in SAR Section 2.3.11.4.2.1, where one or more conduits intersect repository drifts, all the waste packages within the area of the conduits are assumed to be destroyed, and all the waste is assumed to be incorporated into the erupting magma (SNL, 2007ag). The waste is assumed to mix with magma and be carried up the conduit toward the surface, where the magma–waste mixture would be explosively ejected into the atmosphere or flow as lava along the ground.

DOE considered the failed waste packages directly intersected by a conduit to provide no protection against waste release, so, in the DOE model, the conduit size at repository depth directly determines the number of waste packages disrupted. More specifically, DOE calculated the number of waste packages intersected by conduits as a cumulative distribution function based on a distribution for the number of conduits, a distribution for conduit diameters, and the likelihood factors for location of the conduits on the dikes, which includes the design configuration of the subsurface repository. Accordingly, DOE considered additional parameters including waste package size and spacing; drift location and dimensions; and distributions for dike length, orientation, thickness, and number of dikes in an intrusive event. DOE concluded that rising magma in a dike that enters a drift will slow, relative to that in the solid rock pillars between drifts; thus, the dike segment above drifts will lag slightly in breaching the surface. From that conclusion, DOE proposed that vents and conduits are more likely to form between drifts than above them. In most realizations DOE tested, this led to a condition where the volcanic conduit forms along the dike in the rock pillars between drifts and not the drift itself; thus, the most likely value for the number of disrupted waste packages in the model is zero. A range of zero to seven waste packages (SNL, 2007ar) intersected by a conduit during an eruption was modeled in the TSPA, which NRC staff finds acceptable, based upon its evaluation of the previous assumptions.

In DOE's model, uncertainty in conduit size is bounded by a size distribution based on observed host-rock fragments in violent-Strombolian deposits at the Lathrop Wells volcano (Doubik and Hill, 1999aa; SNL, 2007ae, Section 6.4 and Appendix C) and on field studies at analogous sites, which DOE interpreted as suggesting that the diameter is largest at the surface and decreases with depth. DOE gave a distribution for conduit diameters from approximately 4 m [13 ft] (bounded by dike width) to a mean value of 15 m [50 ft] and a 95th percentile value of 21 m [69 ft] for an expected conduit diameter at repository depth (SAR Section 2.3.11.4.2.1.2).

With DOE's volcanic scenario analysis, conduits developed only where the trend of a dike intersected a drift (SAR Section 2.3.11.4.1.1.1). DOE concluded that it is highly unlikely that a secondary conduit will form at some point along the drift away from the dike intersection. This conclusion was based on DOE's finding that magma will solidify quickly and pressures will be insufficient to allow the formation (or maintain the opening) of a secondary dike fed from the magma in the drift. In the analysis involving pyroclastic flow of magma inside a drift (an alternative conceptual model discussed in SER Section 2.2.1.3.10.3.2 with respect to the intrusive case), DOE assessed one situation where it assumed that a secondary fracture had already formed and a secondary opening was created on the drift-top wall (BSC, 2005af). DOE applied a multiphase fluid dynamics analysis to this scenario (Darteville and Valentine, 2005aa; SAR Section 2.3.11.3.2.3). Simulated results exhibited intermediate behavior with a down-drift multiphase flow on the roof and a return flow on the floor. The whole system with these two openings formed a clearly defined recirculation pattern in the drift with some materials leaving the system and some materials recycling back into the drift along the roof. Simulations also showed that this scenario leads to relatively high dynamic pressures compared with a single-conduit situation. Other simulations indicated that blockage of the volcanic conduit might also create secondary breakouts at a point away from the location of initial dike intersection (SAR Section 2.3.11.3.2.2). Although DOE acknowledged that the chance of these scenarios occurring was unlikely, it concluded that such scenarios could lead to a one to two order-of-magnitude increase in the amount of waste released during a volcanic igneous event (essentially equivalent to the waste content of a single drift, ~70–100 waste packages), which would cause no more than a one to two order-of-magnitude increase in expected annual dose (SAR Section 2.3.11.3.2.2).

The NRC staff's review of the DOE approach to modeling the development of subvolcanic conduits is guided by the information in SAR Section 2.3.11.3.2.2 regarding dose sensitivity to the waste source term. It also depends on analyses in SAR Section 2.4.2 that demonstrate that the volcanic case contributes approximately 0.1 percent of the total dose for the igneous scenario, which the NRC staff find acceptable, as discussed in SER Section 2.4.2.2.1.2.3.2. The NRC staff determined that the DOE mean conduit diameter of 15 m [49 ft] (SAR Section 2.3.11.4.2.1.2) is largely based upon estimates of the conduit diameter for Lathrop Wells volcano; however, the calculation for the conduit diameter given in SNL Appendix F (2007ae) and adopted in the SAR appears to be in error. The diameter was recalculated in a journal article on the Lathrop Wells volcano using the same information by Valentine, et al. (2007aa) at ~8 to 9 m [~26 to 30 ft], which the NRC staff has reviewed and determined to be the correct value. In contrast, many of the smaller conduit diameters that DOE used in supporting this parameter value are from eroded volcanoes in the Yucca Mountain region that DOE concluded are not representative of expected basaltic igneous processes (e.g., CRWMS M&O, 1996aa). On the basis of this information and other published information about basaltic volcanic conduits located several hundred meters [~1,000 ft] below surface (e.g., Doubik and Hill, 1999aa; Valentine and Groves, 1996aa; Valentine and Krogh, 2006aa; Delaney and Gartner, 1997aa), the NRC staff concludes that uncertainty in the average and maximum conduit diameter may be a factor of five greater than DOE considered. The magnitude of this uncertainty, however, would increase the expected annual dose for volcanic igneous events by less than an order of magnitude. Because DOE calculates that the volcanic case contributes 0.1 percent of the total dose to the igneous scenario (SER Section 2.4.2.2.1.2.3.2), the NRC staff concludes that this increase in uncertainty in conduit diameter, and thus dose, would not be significant. Thus, the NRC staff concludes that the DOE approach for representing subvolcanic conduits is acceptable.

Evaluation of Magma–Waste Interaction and Mixing in a Drift and Conduit

In DOE's TSPA, the amount of waste incorporated into a volcanic conduit is determined by the area of a drift intersected by a stylized cylindrical conduit. This model assumes that waste from disrupted packages located outside the boundary of the conduit will not be entrained into the upward-flowing magma in the conduit. Additional DOE analyses (SAR Section 2.3.11.3.4.4) described how circulation of magma and gas might occur between a conduit and other parts of the intersected drift. However, DOE did not characterize the extent or magnitude of this circulation or evaluate the potential for this circulation to entrain small particles of degraded waste from elsewhere in a drift beyond the conduit. Additional degraded waste may be available, as DOE assumed that the waste form is instantly degraded when the waste packages fail during the intrusive event (SAR Section 2.3.11.3.2.4).

The NRC staff reviewed the DOE information in SAR Section 2.3.11.3.4.4 and associated published literature. The NRC staff concludes that although DOE did not evaluate the effects of potential magma circulation, the significance of these potential effects would be less than effects associated with secondary conduit development for the following reasons. As Menand, et al. (2008aa) discussed, magma circulation in an intersected drift has the potential to transport some sizes of waste particles into the erupting conduit, if flow conditions in the drift are appropriate. The NRC staff expects that only a relatively small amount of waste particles could potentially be transported by magma circulation because materials on the floor of the drift (e.g., pallets on which the waste canisters rest, damaged/degraded engineered barrier system materials, and the invert; SAR Section 2.1) would present obstacles to magma flow (SNL, 2007ag,ar; Detournay, et al., 2003aa; Darteville and Valentine, 2009aa). These obstacles would also present rough surfaces that would impede waste particles from entrainment in the circulating magma and thus limit the amount of waste that could be released in an eruption. The NRC staff, therefore, concludes that the potential increase in entrained waste due to magma circulation would be significantly less than an order of magnitude (i.e., much less than the amount of waste contained in a potentially intersected drift). By providing acceptable analysis showing that the expected annual dose increases linearly with increasing source term for the volcanic modeling case, as detailed in SAR Section 2.3.11.3.2.2 and SNL (2008ag, Appendix P), DOE has provided sufficient information for staff to conclude that the potential effects of magma circulation are not significant to dose estimates from the igneous scenario.

Further, the NRC staff reviewed the information DOE provided in SAR Section 2.3.11.3.2.2 to evaluate the potential effects of secondary conduits developing away from the location of dike intersection with the drift. The NRC staff concludes that the likelihood of secondary conduits relies on a series of unusual conditions and thus appears remote. However, the NRC staff accepts the DOE assumption that development of a secondary conduit could potentially lead to the eruption of all waste in an intersected drift, as the assumption is consistent with available information (Woods, et al., 2002aa; Menand, et al., 2008aa; Lejeune, et al., 2009aa; Darteville and Valentine, 2009aa). The NRC staff concludes that DOE acceptably accounted for the uncertainty associated with secondary conduit formation by considering an alternative model and providing an analysis in SAR Section 2.3.11.3.2.2. This analysis shows a hypothesized two orders-of-magnitude increase in the amount of waste entrained in an eruption might lead to a two orders-of-magnitude increase in expected annual dose for the volcanic modeling case [see also SNL (2008ag, Appendix P)]. The NRC staff thus concludes that DOE has acceptably addressed the significance of secondary conduit formation, because the likelihood of secondary conduits appears remote, and the significance to performance appears to be much smaller than the two orders of magnitude presented in SAR Section 2.3.11.3.2.2 (i.e., much less than 10 percent of the total igneous scenario).

In the DOE volcanic eruption modeling scenario, the number of waste packages intersected is an input in the TSPA for calculating the amount of waste erupted, along with the probability that a conduit will develop in a drift containing waste packages. DOE used a Monte Carlo technique to account for parameter uncertainties such as the future time at which an eruption might occur and the possibility that more than one eruption could happen in the future of the repository. DOE calculates a magma partitioning factor (SAR Section 2.3.11.4.2.2.2; SNL, 2007ab) to determine the amount of the waste partitioned into a potential volcanic tephra fall deposit, the only volcanic product that is significant to dose (SER Section 2.2.1.3.13.3.1). DOE determined that 10 to 50 percent of the total amount of waste entrained in an eruption will be in the resulting tephra fall deposit. The magma partitioning factor and the expected style of eruption (violent Strombolian) from the volcanic conduit(s) is evaluated as part of the abstraction for airborne volcanic transport in SER Section 2.2.1.3.13.

DOE concluded that the amount of waste particles incorporated into the erupting magma would only constitute a minor amount (trace phase) in the magma in all the applicant's scenarios and that its presence would not be expected to influence the eruptive behavior of the magma (SNL, 2007ab). The NRC staff finds that DOE's estimate of the amount of the waste that could become incorporated into the fallout deposit is adequately documented and supported. NRC staff finds that DOE's claim that the amount of waste transported into a conduit and into the tephra deposit would be on the order of 10^{-6} of the concentration of tephra at any point in the deposit (SAR Section 2.3.11.4.2.2.3) to be adequate. The NRC staff concludes that the waste particles will not affect the eruptive processes occurring in the magma (SNL, 2007ab) and that the style of eruption would not be influenced by the presence of the waste. Therefore, the NRC staff further concludes that it is appropriate for DOE to model dispersal and fall for airborne transport of radionuclide-contaminated tephra (reviewed in SER Section 2.2.1.3.13) on the basis of past and current similar-style volcanic activity.

Summary and Findings of NRC Staff's Review on the Volcanic Eruption Modeling Case

The NRC staff concludes that DOE has provided sufficient information to support the modeling approach used to represent volcanic igneous events in the performance assessment. DOE has adequately represented how potential volcanic conduits could form randomly along an igneous intrusion (dike) and entrain waste. The NRC staff concludes that DOE acceptably assumed that all waste packages located within the footprint of a potential drift-intersecting conduit would release all degraded waste into the erupting magma. The NRC staff has verified that DOE has accounted for uncertainties in the amount of waste that potentially could be disrupted and erupted during a modeled volcanic event by providing calculations of dose sensitivity to the amount of waste erupted (see SER Section 2.2.1.4.1.3.3.2). These calculations form the basis for staff to conclude that uncertainties associated with the potential effects of magma circulation in a drift, or the remote chance of secondary conduit development, would not affect dose significantly. The NRC staff concludes that DOE has provided an acceptable basis for the use of its magma partitioning factor and has supported that basis with information from suitable volcanic analogs. The NRC staff finds that DOE has acceptably represented the potentially significant effects of subsurface igneous ("subvolcanic") processes and events in the performance assessment. The applicant's volcanic scenario is evaluated in SER Section 2.2.1.3.13.

The NRC staff concludes that the incorporation factor of waste into magma that the applicant adopted (the entire contents of waste packages are assumed to be mixed into the magma) is conservative as it allows for greater amounts of incorporated waste than might realistically occur. There are inherent uncertainties in how much waste would be incorporated into erupting magma because no suitable natural analogs have been identified for this process. Potential waste incorporation is highly dependent on the behavior of the magma during the interaction, particularly on the extent of magma fragmentation, which is driven by degassing of the partially solidified magma and depends upon many other variables. Magma fragmentation may occur as deep as repository depths {300 m [~1,000 ft]} in violent Strombolian eruptions (Doubik and Hill, 1999aa). However, in most Strombolian eruptions, fragmentation occurs via explosive gas bubbles bursting at less than 100 m [328 ft] below the surface (Wilson and Head, 1981aa). The presence of repository drifts could cause deeper fragmentation than in typical volcanic environments (e.g., Woods, et al., 2002aa), and DOE's model for waste incorporation relies on a vigorously degassing, partly fragmenting magmatic environment in the drifts. Despite these uncertainties, the NRC staff concludes that these factors will not make a significant difference to the dose from the volcanic eruption modeling scenario, as calculated in the DOE TSPA, because the likely increase in the amount of waste incorporated into the erupted magma is within the range of uncertainty considered.

2.2.1.3.10.4 Evaluation Findings

The NRC staff has reviewed the SAR and other information submitted in support of the license application, which includes information required by 10 CFR 63.21(c)(1),(9), and (15) and finds, with reasonable expectation, that the relevant requirements of 10 CFR 63.114 and 63.342 are satisfied regarding the abstraction of igneous disruption of waste packages. In particular, the NRC staff finds that DOE has adequately

- Included appropriate data related to the geology, hydrology, and geochemistry of the surface and subsurface from the site and the region surrounding Yucca Mountain, and provided adequate information on the design of the engineered barrier system to define parameters and conceptual models used in the performance assessment calculation, in compliance with 10 CFR 63.114(a)(1)

- Accounted for uncertainty and variability in the parameter values used to model igneous disruption of waste packages, in compliance with 10 CFR 63.114(a)(2)

- Considered and evaluated alternative conceptual models that are consistent with currently available data and scientific understanding, in compliance with 10 CFR 63.114(a)(3)

- Provided a technical basis for the inclusion of FEPs affecting igneous disruption of waste packages, consistent with the limits on performance assessment in 10 CFR 63.342, and evaluated in sufficient detail those processes that would significantly affect repository performance, in compliance with 10 CFR 63.114(a)(4–6)

- Provided technical bases for the models of igneous disruption of waste packages used in the performance assessment to represent the 10,000 years after disposal, in compliance with 10 CFR 63.114(a)(7)

- Used performance assessment methods for the period of geologic stability consistent with the methods used to demonstrate compliance for the initial 10,000 years following permanent closure, in compliance with 10 CFR 63.114(b)

- Included in the performance assessment for the period of geologic stability those FEPs used for the initial 10,000 year period, consistent with specific limits, in compliance with 10 CFR 63.342(c)

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CHAPTER 14

2.2.1.3.12 Concentration of Radionuclides in Groundwater

2.2.1.3.12.1 Introduction

Safety Evaluation Report (SER) Section 2.2.1.3.12 provides the U.S. Nuclear Regulatory Commission (NRC) staff review of information the applicant provided in the Safety Analysis Report (SAR) (DOE, 2008ab) on the concentration of radionuclides in groundwater extracted by pumping and used in the annual water demand of the reasonably maximally exposed individual (RMEI). The NRC staff reviewed the methods and assumptions the applicant used to estimate groundwater radionuclide concentrations. The NRC staff review focused on SAR Sections 2.3.9 and 2.4.4. SAR Section 2.3.9 includes discussions of saturated zone radionuclide transport and groundwater. SAR Section 2.4.4 includes the applicant's analysis of repository performance, which states the proposed repository would meet the standards for the protection of groundwater.

2.2.1.3.12.2 Regulatory Requirements

The regulatory requirement for the annual well water demand to be used for evaluating the concentration of radionuclides in the groundwater is specified in 10 CFR 63.312(c), which states that the RMEI uses well water with average concentrations of radionuclides based on an annual water demand of 3,000 acre-ft [3.7×10^9 L]. In performing its review, the NRC staff followed the guidance provided in the Yucca Mountain Review Plan (YMRP) Section 2.2.1.3.12 (NRC, 2003aa). Consistent with the YMRP, the NRC staff considered risk information to determine how to evaluate the applicant's analysis of the concentration of radionuclides in groundwater.

2.2.1.3.12.3 Assessment of Well Water Concentration Estimates

In SAR Section 2.4.4, the applicant stated that it assumed that all radionuclides transported by groundwater from the Yucca Mountain disposal system in a given year are captured in the annual water demand of 3,000 acre-ft [3.7×10^9 L]. The applicant determined the annual mean concentrations of transported radionuclides in the saturated-zone groundwater by dividing the annual mass flux of radionuclides reaching the accessible environment boundary by the annual water demand (SAR Section 2.4.4.1.1.1). As presented by the applicant in SAR Section 2.3.9, this annual mass flux includes both those radionuclides explicitly transported in the Total System Performance Assessment model and those calculated assuming secular equilibrium in long-lived decay chains.

NRC Staff Review

YMRP Section 2.2.1.3.12 states that if the applicant assumes that all radionuclides that reach the RMEI in a given year are included in the pumping wells in the annual water demand of 3,000 acre-ft [3.7×10^9 L], then the NRC staff should conduct a simplified review focusing on the bounding assumptions. In SAR Section 2.4.4, the applicant stated that it assumed that all radionuclides transported by groundwater from the Yucca Mountain disposal system in a given year are captured in the annual water demand of 3,000 acre-ft [3.7×10^9 L]. Therefore, the NRC staff followed the simplified review approach, consistent with YMRP Section 2.2.1.3.12. The NRC staff verified that the applicant determined the annual mean concentrations of transported radionuclides in the saturated-zone groundwater by dividing the annual mass fluxes of radionuclides reaching the accessible environment boundary by the annual water demand (SAR Section 2.4.4.1.1.1). The NRC staff evaluation of the radionuclide mass flux is provided in SER Section 2.2.1.3.9. The applicant's saturated zone transport abstraction model (SAR Section 2.3.9) tracks transport of a set of contaminant radionuclides and assumes that long-lived, decay-chain daughter radionuclides are in secular equilibrium with their parents at the accessible environment boundary. As discussed in SER Section 2.2.1.3.9, this assumption may not be reasonable for cases where a long-lived parent radionuclide is more strongly sorbed than its decay products. In its response to the NRC staff's request for additional information, the applicant evaluated this effect and showed that for the conditions expected in the saturated-zone transport path, the magnitude of the predicted excess daughter activity is not significant for performance (DOE, 2009df). The NRC staff concludes in SER Section 2.2.1.3.9 that, including the uncertainty from possible excess activity of decay-chain daughter radionuclides, the applicant's representation of the annual mass fluxes of radionuclides reaching the accessible environment boundary is acceptable.

The NRC staff evaluations of the applicant's compliance with the individual protection standard and with the groundwater protection standard are provided in SER Sections 2.2.1.4.1 and Section 2.2.1.4.3, respectively.

2.2.1.3.12.4 Evaluation Findings

The NRC staff has reviewed the SAR and other information submitted to support the license application relevant to the concentration of radionuclides in groundwater, and finds, with reasonable expectation, that the applicant adequately demonstrated that the concentration of radionuclides in groundwater is acceptable because

- The applicant assumed a RMEI annual water demand in accordance with 10 CFR 63.312(c)

- The applicant adequately demonstrated that the RMEI uses well water with average concentrations of radionuclides by dividing the annual mass fluxes of radionuclides reaching the accessible environment boundary by the annual water use of 3,000 acre-ft [3.7×10^9 L]

2.2.1.3.12.5 References

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CHAPTER 15

2.2.1.3.13 Airborne Transport and Redistribution of Radionuclides

2.2.1.3.13.1 Introduction

Safety Evaluation Report (SER) Section 2.2.1.3.13 evaluates the U.S. Department of Energy's (DOE, or the applicant) information on airborne transport and deposition of radionuclides expelled by a potential future volcanic eruption following igneous disruption of waste packages. It also evaluates DOE information on the redistribution of those radionuclides in soil. This evaluation of DOE's performance assessment for the volcanic eruption modeling case is a sequel to the evaluation of possible igneous disruption of the proposed repository (DOE's igneous intrusion modeling case; see SER Section 2.2.1.3.10). This SER Section also evaluates redistribution of radionuclides in soil in the accessible environment, which in the DOE model arrives in the accessible environment via groundwater transport. The U.S. Nuclear Regulatory Commission (NRC) staff's evaluation is based on information in the DOE Safety Analysis Report (SAR) (DOE, 2009av), as supplemented by DOE responses (DOE, 2009bk–bm) to the NRC staff's requests for additional information (RAIs).

This SER Section addresses two of the 14 model abstraction sections indicated in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa): airborne transport of radionuclides (YMRP Section 2.2.1.3.11) and redistribution of radionuclides in soil (YMRP Section 2.2.1.3.13). The NRC staff's assessment of information in DOE's SAR for these two abstraction sections used the guidance in the YMRP to conduct a risk-informed review. Together, airborne transport of radionuclides during a potential future explosive volcanic eruption that generates tephra {fragments of cooled magma that are transported through the air, including ash particles that have diameters less than 2 mm [0.08 in]} and redistribution of radionuclides deposited on the landscape by that eruption constitute the DOE volcanic ash exposure scenario in its biosphere model for the Total System Performance Assessment (TSPA) (SAR Section 2.3.10.2.6). As part of the review of redistribution of radionuclides, NRC staff evaluated the DOE performance assessment for the scenario where radionuclide-contaminated groundwater may cause the reasonably maximally exposed individual (RMEI) to be exposed to a dose (SAR Section 2.3.10.2.3). SAR Figure 2.3.10-1 displayed a separate flow of information for the volcanic ash exposure scenario compared to the groundwater exposure scenarios in the DOE performance assessment. This SER Section reflects this separation of information and presents the NRC staff's review and evaluation, first for the volcanic ash exposure scenario and second for the groundwater exposure scenario.

For the volcanic ash exposure scenario, the NRC staff evaluated the following three abstracted models addressed in DOE's SAR:

1. Airborne transport, dispersion, and deposition of tephra and high-level waste

2. Redistribution by fluvial (running water or stream) transport of contaminated tephra within the Fortymile Wash catchment basin, mixing and dilution with uncontaminated sediment, and deposition of the tephra-sediment mixture on the Fortymile Wash alluvial fan at the RMEI location, near the Fortymile Wash alluvial fan apex

3. The downward migration of radionuclides in the soil at the alluvial fan in the accessible environment and RMEI location

The latter two abstracted models comprise the DOE performance assessment appropriate for redistribution of radionuclides in soil (YMRP Section 2.2.1.3.13), while the first abstracted model constitutes DOE's performance assessment for airborne transport of radionuclides (YMRP Section 2.2.1.3.11).

For the groundwater exposure scenario, this SER Section presents the NRC staff evaluation of the DOE surface soil submodel, which is also part of the performance assessment for redistribution of radionuclides in soil. For both exposure scenarios, the final outputs of the abstractions evaluated in this SER Section are radionuclide concentrations in soil, which are direct inputs to the DOE biosphere model for calculating annual doses to the RMEI (reviewed by NRC staff in SER Section 2.2.1.3.14). Associated with this, SER Section 2.2.1.3.4 presents the NRC staff evaluation of the radionuclide inventory, which is an input to the volcanic ash exposure scenario (SAR Figure 2.3.10-3). SER Section 2.2.1.4.1 assesses the demonstration of compliance with the postclosure individual protection standard and presents the NRC staff's evaluation of the overall TSPA.

2.2.1.3.13.2 Regulatory Requirements

The applicant is required to provide information specified in 10 CFR 63.21(c)(1), (9), and (15) that is related to abstraction of airborne transport and redistribution of radionuclides. The requirements in 10 CFR 63.114 (Requirements for Performance Assessment) and 63.342 (Limits on Performance Assessments) include postclosure performance assessment to demonstrate compliance with 63.113 (Performance Objectives for the Geologic Repository after Permanent Closure). Specific compliance with 63.113 is reviewed in SER Section 2.2.1.4.1.

The regulations for performance assessment in 10 CFR 63.114 require, in part, that a performance assessment

- Include data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the surface and subsurface from the site and the region surrounding Yucca Mountain [10 CFR 63.114(a)(1)]

- Account for uncertainty and variability in the parameter values [10 CFR 63.114(a)(2)]

- Consider and evaluate alternative conceptual models [10 CFR 63.114(a)(3)]

- Provide technical bases for either the inclusion or exclusion of features, events, and processes (FEPs), including effects of degradation, deterioration, or alteration processes of engineered barriers that would adversely affect performance of the natural barriers, consistent with the limits on performance assessment in 10 CFR 63.342, and evaluate in sufficient detail those processes that would significantly affect repository performance [10 CFR 63.114(a)(4-6)]

- Provide technical basis for the models used in the performance assessment to represent the 10,000 years after disposal [10 CFR 63.114(a)(7)]

The NRC staff's evaluation of inclusion or exclusion of FEPs is given in SER Section 2.2.1.2.1. 10 CFR 63.114(a) sets forth requirements for performance assessment for the initial 10,000 years following disposal. Regulations at 10 CFR 63.114(b) and 10 CFR 63.342 set forth requirements for the performance assessment methods for the time from 10,000 years through the period of geologic stability, defined in 10 CFR 63.302 as 1 million years following disposal. These sections provide that through the period of geologic stability, with specific limitations, the applicant

- Use performance assessment methods consistent with the performance assessment methods used to demonstrate compliance for the initial 10,000 years following permanent closure [10 CFR 63.114(b)]

- Include in the performance assessment those FEPs used in the performance assessment for the initial 10,000-year period (10 CFR 63.342)

This model abstraction of airborne transport and redistribution of radionuclides involves igneous activity. Thus, 10 CFR 63.342(c)(1) also applies to this abstraction, as this regulation requires that DOE assess the effects of seismic and igneous activity on the repository performance, subject to the probability limits in 63.342(a) and 63.342(b). Specific constraints on the analysis required for seismic and igneous activity are set forth in 10 CFR 63.342(c)(1)(i) and (ii), respectively.

In addition, the following requirements for characteristics of the reference biosphere to be used in the abstraction for redistribution of radionuclides in soil are specified in 10 CFR 63.305:

- FEPs that describe the reference biosphere must be consistent with present knowledge of the conditions in the region surrounding the Yucca Mountain site. [10 CFR 63.305(a)]

- DOE should not project changes in society, the biosphere (other than climate), or human biology or increases or decreases of human knowledge and technology; in all analyses done to demonstrate compliance with this part, DOE must assume that all of those factors are constant as they are at the time of submission of the license application. [10 CFR 63.305(b)]

- DOE must vary factors related to the geology, hydrology, and climate based upon cautious but reasonable assumptions of the changes in these factors that could affect the Yucca Mountain disposal system during the period of geologic stability, consistent with the requirements for performance assessments specified at 10 CFR 63.342. [10 CFR 63.305(c)]

- Biosphere pathways must be consistent with arid or semi-arid conditions. [10 CFR 63.305(d)]

The NRC staff's review of the SAR and supporting information follows the guidance laid out in the YMRP Sections 2.2.1.3.11, Airborne Transport of Radionuclides, and 2.2.1.3.13, Redistribution of Radionuclides in Soil, as supplemented by additional guidance for the period beyond 10,000 years after permanent closure (NRC, 2009ab). The YMRP acceptance criteria for model abstractions that provide guidance for the NRC staff's review of the applicant's abstraction of airborne transport and redistribution of radionuclides are

1. System description and model integration are adequate
2. Data are sufficient for model justification
3. Data uncertainty is characterized and propagated through the abstraction
4. Model uncertainty is characterized and propagated through the abstraction
5. Model abstraction output is supported by objective comparisons

In its review of SAR and supporting information, the NRC staff used a risk-informed approach and the guidance in the YMRP, as supplemented by NRC (2009ab), for aspects of climate and infiltration important to repository performance. The NRC staff considered all five YMRP criteria in its review of information provided by DOE. In the context of these criteria, only those aspects of the model abstraction that substantively affect the performance assessment results, as determined by the NRC staff, are discussed in detail in this SER Section. The NRC staff's determination is based both on risk information provided by DOE and on the NRC staff's knowledge gained through experience and independent analyses

2.2.1.3.13.3 Technical Review

In SAR Figure 2.3.11-1, DOE presented the information flow for the volcanic eruption modeling case. The DOE abstracted model on atmospheric dispersal and deposition of tephra constitutes its performance assessment of airborne transport of radionuclides. DOE's abstracted models for tephra redistribution and vertical radionuclide migration into the soil together comprise the performance assessment of redistribution of radionuclides in soil.

Airborne transport of radionuclides pertains to the volcanic ash exposure scenario, which involves a possible disruption of the Yucca Mountain repository by a future volcanic eruption. In this scenario, high-level radioactive waste is mixed with magma and ejected into the atmosphere incorporated within the volcanic tephra. The airborne transport abstracted model accepts the number of waste packages intersected by volcanic conduits, provided in SAR Section 2.3.11.4.2.1 and evaluated in SER Section 2.2.1.3.10, and estimates the concentration and thickness of radionuclide-contaminated tephra that could be deposited on the ground surface of the Yucca Mountain region (SAR Figure 2.3.11-1). As depicted in SAR Figure 2.3.11-1, DOE then uses this information as input to the volcanic ash exposure scenario (SAR Section 2.3.10) for estimating the dose to the RMEI via surface redistribution of contaminated tephra and by migration of radionuclides from tephra particles into the soil, as described next.

Using the various abstracted models for redistribution of tephra into soil enabled the applicant to calculate the time-dependent profile of radionuclide concentration in the contaminated soil horizon at the RMEI location. The DOE airborne transport abstracted model provides input on the tephra deposit for the tephra redistribution calculations of waste concentrations in redistributed tephra. Another DOE redistribution-related abstracted model uses this information to estimate the downward migration of radionuclides from tephra into soil at the alluvial fan of Fortymile Wash and calculates the concentration of waste in redistributed tephra at the RMEI location (SAR Figure 2.3.11-1). Waste concentration information from DOE's redistribution models is coupled with information on the radionuclide inventory (radionuclide activities per unit mass of waste) to yield radionuclide concentration profiles in soil. The fraction of tephra that can be resuspended and inhaled by the RMEI during activities such as soil tillage is also important; this is the dominant exposure pathway for the first 10,000 years after repository closure in DOE's performance assessment analysis.

In this section, the NRC staff also evaluates the DOE surface soil submodel for the groundwater exposure scenario, described in SAR Section 2.3.10. In this model, radionuclides are considered to be added to the surface soil from irrigation with contaminated groundwater. The surface soil submodel accepts the concentration of radionuclides in groundwater in the accessible environment (as provided in SAR Section 2.4.4 and reviewed in SER Section 2.2.1.3.12) and calculates loss of radionuclides from the surface soil via mechanisms such as radioactive decay, leaching into deeper zones, erosion of soil particles, and gaseous releases to the atmosphere. As depicted in SAR Figures 2.3.10-1 and 2.3.10-10, the output from the surface soil model is used by the rest of the DOE biosphere model, which is reviewed in SER Section 2.2.1.3.14.

NRC Staff Perspective on Risk

The volcanic ash exposure scenario and groundwater exposure scenario provide different contributions to repository performance in the DOE assessment. The NRC staff finds acceptable the applicant's assessment that the volcanic ash exposure scenario following an eruption does not significantly influence repository performance, because DOE shows that its mean dose contribution is more than a factor of 1,000 smaller than the overall peak dose within the initial 10,000 years and more than a factor of 10,000 smaller than the overall peak dose after 10,000 years (SAR Figure 2.4-18). Further, DOE's dose exposure assessment is consistent with the NRC staff's independent analyses (see SER Section 2.2.1.4.1.3.3.2). The remaining DOE modeling cases depicted in SAR Figure 2.4-18 constitute the groundwater exposure scenario. The groundwater exposure scenario dominates the overall peak dose within 10,000 years and after 10,000 years (SAR Figure 2.4-18). This risk information suggests that the NRC staff should focus on the surface soil submodel in the groundwater exposure scenario and conduct a simplified review of the volcanic ash exposure focusing on the bounding assumptions. In addition to a detailed review of the surface soil submodel in the groundwater scenario, the NRC staff also conducted a detailed review of the volcanic ash exposure scenario based on DOE's multiple barrier information (SAR Section 2.1.3), consistent with YMRP Section 2.2.1.3.

The NRC staff review of SAR Section 2.1.3 determined that no aspect of the information reviewed in this chapter is identified as a barrier. SER Section 2.2.1.1 describes the NRC staff's evaluation of multiple barriers. However, DOE did identify that a volcanic event could adversely affect the engineered barrier system's ability to prevent the release or reduce the release rate of radionuclides from the waste and to prevent or reduce the movement of radionuclides away from the repository (SAR Section 2.3.11.1) by destroying the waste packages and releasing the contained radionuclides in the erupting material (SAR Section 2.1.2.2.5). On the basis of the NRC staff review in the forthcoming SER Section 2.2.1.3.13.3.1 pertaining to the acceptability of DOE's implementation of the abstracted models in the TSPA analysis and adequacy of technical bases for models used in the performance assessment, the NRC staff concludes that the applicant's technical basis for barrier capability is based on, and is consistent with, the technical basis for the performance assessment used to demonstrate compliance with 10 CFR 63.113(b).

For the volcanic ash exposure scenario, the NRC staff's evaluation of DOE's performance assessment is presented in SER Section 2.2.1.4.1.3.2. The four input quantities identified for that evaluation (fraction of waste incorporated with tephra to the total waste erupted, tephra volume, tephra density, and ash areal concentration) directly relate to the airborne transport abstracted model reviewed in this SER Section. All three abstracted models evaluated in this SER Section for the volcanic ash exposure scenario account for the bulk transport of radionuclides in waste and do not include processes that separate radionuclides or transport radionuclides at different rates. For this reason, the evaluation of the abstracted models for volcanic ash exposure in this SER Section does not focus on individual radionuclide contributions to total dose, which are considered in SER Section 2.2.1.4.1.3.3.2.

However, an understanding of the dominant radionuclides, and their exposure pathways, that contribute to dose provides risk insights into important aspects of the performance assessment for the DOE volcanic eruption modeling case. SAR Figure 2.4-32 identified the contribution of radionuclides to mean annual dose for the volcanic eruption modeling case. SAR Table 2.3.10-15 identified the average percentage exposure pathway contributions to the annual dose for the volcanic ash exposure scenario. On the basis of its review of DOE information, the NRC staff concludes that at early times (i.e., before 500 years), the overall dose is dominated by six radionuclides: Sr-90, Cs-137, Pu-238, Pu-239, Pu-240, and Am-241. At longer times (i.e., after 5,000 years), the dose is dominated by Pu-239 and Pu-240. And at very long times (i.e., after 100,000 years), the dose is dominated by Ra-226 (SAR Figure 2.4-32). On the basis of its identification of the dominant dose contributors and review of information in DOE's SAR Table 2.3.10-15, the NRC staff concludes that inhalation of particulates from the resuspension of contaminated tephra deposits is the dominant exposure pathway for 10,000 years. After about 100,000 years, for Ra-226, the dominant exposure pathway is external exposure (SAR Table 2.3.10-15). The NRC staff's independent analyses, documented in NRC (2005aa, Volume 2, Appendix D), reached similar conclusions about the dominant exposure pathway. Therefore, DOE results are acceptable. DOE's representations of inhalation of particulates from resuspension of contaminated tephra deposits were found by the NRC staff to be important to dose results for a volcanic ash exposure scenario in prior independent NRC performance assessment results, as identified in NRC (2005aa, Volume 2, Appendix D). Thus, the NRC staff has reviewed the performance assessment to (i) focus on those processes that most affect the concentration of waste in the resuspendable layer and (ii) focus on those processes that most affect the concentration of waste in the soil layers that control external exposure.

For the groundwater exposure scenario, the surface soil submodel reviewed in this SER Section is one component of the abstracted model for the biosphere that calculates biosphere dose conversion factors. For the radionuclides Tc-99, I-129, Np-237, and Pu-242 discussed in SER Section 2.2.1.4.1.3 in the table on groundwater biosphere dose conversion factors and in SER Section 2.2.1.4.1.3.3, the pathways linked to the surface soil submodel account for up to 50 percent of the radionuclide biosphere dose conversion factor, as identified in SNL (2007ac, Tables 6.13-1 and 6.13-2).

On the basis of these risk considerations, the NRC staff conducted a risk-informed review of airborne transport of radionuclides and redistribution of radionuclides in the soil by evaluating the DOE information relative to the acceptance criteria in YMRP Sections 2.2.1.3.11.3 and 2.2.1.3.13.3. The NRC staff focused the review on those aspects of these model abstractions that impact compliance with the individual protection standard (or calculated dose to the RMEI). To assess the effect that the combined uncertainties could have on calculated dose, the NRC staff also focused on those aspects that could cause at least a factor of two effect on intermediate model outputs over the range of an individual parameter value.

As identified above, SAR Figure 2.3.10-1 displayed a separate flow of information for the volcanic ash exposure scenario compared to the groundwater exposure scenarios in the DOE performance assessment. The NRC staff's evaluations of the DOE information on the volcanic ash exposure scenario and the surface soil submodel for the groundwater exposure scenario are documented in the next SER Sections 2.2.1.3.13.3.1 and 2.2.1.3.13.2, respectively.

2.2.1.3.13.3.1 Assessment and Review of the Volcanic Ash Exposure Scenario

The NRC staff's evaluations of DOE's abstracted models on (i) airborne transport, dispersion, and deposition of tephra and high-level waste, (ii) redistribution of tephra, and (iii) the vertical movement of radionuclides in the soil at the alluvial fan in the accessible environment are presented separately below in three subsections. Each subsection first identifies those important aspects of the DOE abstracted model that were the focus of the NRC staff's review. The NRC staff then summarizes the DOE license application for the abstracted model, followed by its review and evaluation. After the NRC staff's evaluation of the vertical movement of radionuclides in the soil, the NRC staff's overall findings on the volcanic ash exposure scenario are summarized in a subsection (Summary Evaluation Findings on Volcanic Ash Exposure Scenario) at the end of SER Section 2.2.1.3.13.3.1.

In addition to reviewing the individual abstracted models, the NRC staff also reviewed how DOE implemented these models into the TSPA. Because the acceptability of DOE's implementation of the abstracted models in the TSPA analysis is dependent on the NRC staff's findings for the individual model abstractions, the NRC staff's evaluation of DOE's implementation of the abstracted models in the TSPA analysis is presented in the overall evaluation on the volcanic ash exposure scenario at the end of SER Section 2.2.1.3.13.3.1. To place the individual model abstractions into the framework of the TSPA analysis, the NRC staff summarizes DOE's implementation of the volcanic eruption modeling case next.

The applicant integrated abstracted models of TSPA analysis for the volcanic eruption modeling case (volcanic interaction with the repository, atmospheric transport, tephra redistribution, volcanic ash exposure) in a GoldSim modeling environment. The applicant used the initial radionuclide inventory from a "blended" waste package to calculate radionuclide transport, as described in the review of the previously mentioned submodels. A blended waste package inventory was calculated by using a weighted average of commercial spent nuclear fuel and codisposal waste packages and inventories. Following a conditional future eruptive event, tephra transport and redistribution are abstracted to occur instantaneously (i.e., radionuclide waste transport to the RMEI is instantaneous) (SAR Section 2.3.11.4.2.3.1). The time dependence of radionuclide diffusion (downward migration) into the soil at the RMEI location was accounted for in the tephra redistribution model. The radionuclide concentration in the soil at the RMEI location, in g/cm^2, was modified by a "decay factor" to account for radionuclide decay and ingrowth. The resultant source term was provided to the volcanic ash exposure submodel to calculate dose.

2.2.1.3.13.3.1.1 Airborne Transport Modeling

The NRC staff conducted a risk-informed review of airborne transport of radionuclides, concentrating on aspects important to the volcanic ash exposure scenario in the DOE performance assessment.

Important Aspects of Airborne Transport

The abstracted model for atmospheric transport of radionuclides determines the characteristics of contaminated tephra deposited on the surrounding landscape. The applicant's analysis results indicated that the following parameters for airborne transport were influential to the volcanic ash exposure scenario: magma partitioning factor; tephra volume; eruptive power and duration; and mean ash particle diameter, wind direction, and wind speed. The magma partitioning factor is a fraction between zero and one and acts as a direct multiplier on the eruption source term and eruptive dose, similar to the number of waste packages entrained into the erupting magma that pertains to the review in SER Section 2.2.1.3.10. DOE's analysis results [e.g., SNL (2007ab, Figures C–1 and C–2)] showed that waste concentration in tephra is sensitive to tephra volume, eruptive power, and mean ash particle diameter. DOE sensitivity analyses concluded that the initial tephra thickness at the RMEI location (near the Fortymile Wash alluvial fan apex) is strongly dependent on wind direction, wind speed, and mean ash particle diameter, and moderately dependent on eruptive power and eruptive duration, as identified in SNL (2007ab, Appendix C). The applicant found that other parameters of the airborne transport abstracted model were less influential on tephra thickness.

Summary of DOE's Approach on the Airborne Transport Model

The applicant's volcanic eruption modeling case was described in SAR Section 2.3.11.4. In SAR Table 2.3.11-1, DOE identified the FEPs included in the TSPA.

For a repository-drift-penetrating basaltic eruption, DOE modeled the contamination of tephra with waste and the amount of radionuclides contained within the tephra-fall deposit. On the basis of studies of analog volcanoes, DOE apportioned contaminated magma into three volcanic products: namely, lava, scoria cone-forming deposits (selectively composed of the largest tephra fragments), and more widespread tephra-fall deposits. To account for waste that is incorporated in volcanic ejecta that form scoria cones and lava flows, the applicant applied a magma partitioning factor for the fraction of waste incorporated with tephra to the total waste erupted. In the DOE model for the extrusive event, only waste incorporated with tephra contributes radiological dose to the RMEI; waste apportioned into lava flows and scoria cones does not contribute to dose. The amount of waste incorporated in tephra scales with the magma partitioning factor.

The applicant's igneous eruption modeling identified that all explosive phases of the most likely future eruption, on the basis of the interpreted behavior of the youngest volcano near the repository site (Lathrop Wells; Valentine, et al., 2007aa), are considered to be violent Strombolian (SAR Section 2.3.11.4.1). The eruption would produce plumes of tephra in the atmosphere that could transport particulates, including high-level radioactive waste, downwind from the vent. This process could deposit radionuclides at the RMEI location, either from direct sedimentation of contaminated ash particles from the volcanic plume or from the remobilization by wind or surface water of the radionuclide-contaminated volcanic ash after initial deposition. The DOE approach to determining waste concentration in the tephra is sensitive to the tephra volume. For example, smaller tephra volumes result in higher waste concentration in tephra (i.e., waste mass per unit mass of tephra) for the same number of waste packages entrained. In SNL (2008ag), DOE evaluated exposure to airborne concentrations of radionuclides that are contained in the tephra during the eruption (direct tephra-fall exposure) and found that it did not increase expected annual dose significantly due to the extremely short exposure duration. Readers of the following review are reminded that "tephra" refers to airborne magmatic fragments of all sizes, whereas "ash" refers specifically to particles less than 2 mm [0.08 in] in diameter.

A violent Strombolian-type eruption is characterized by the development of a sustained, buoyant plume of hot air and volcanic tephra that commonly rises several kilometers [a few miles] above the volcano. DOE modeled the dispersal processes as turbulent advection diffusion using the

Suzuki (1983aa) model. The Ashplume conceptual model and the ASHPLUME code (Jarzemba, et al., 1997aa), as used by the applicant, implement the Suzuki approach to model the dispersal of tephra on the basis of the diffusion of particles from an eruption column, horizontal diffusion of particles by atmospheric turbulence, horizontal advection by atmospheric circulation, and settling of particles by gravity. ASHPLUME accounts for incorporation and entrainment of waste particles into magma during a potential volcanic eruption through the repository and estimates the concentration (expressed as g/cm^2) and thickness of radionuclide-contaminated tephra deposited on the ground surface. Following a conditional eruptive event, tephra transport is abstracted to occur instantaneously.

In the DOE approach, wind direction significantly affects tephra dispersal and deposit location. Tephra deposits that might fall at the RMEI location are strongly dependent on the presence of northerly winds that would transport the tephra plume to the south from a volcanic vent within the repository area (SAR Figure 2.3.11-13). The tephra deposit at the RMEI location becomes negligible for winds without a strong northerly component (north, north-northwest, or north-northeast), as identified in SNL (2007ab, Figure C–7 and Table D–5). The majority of the anticipated wind vectors at the site result in tephra being deposited to the east of Yucca Mountain (SAR Figure 2.3.11-15). According to SNL (2007ab, Appendix K), this wind direction provides a source of material for remobilization within the Fortymile Wash catchment basin. Applicant-performed sensitivity analyses indicated that wind direction produced a greater contribution to dose than plume spread and divergence, as outlined in SNL (2007ab, Figure K–4c). These sensitivity analyses also demonstrated that increasing the wind speed causes the tephra deposit center mass to shift downwind.

NRC Staff's Review of the Airborne Transport Model

The NRC staff reviewed SAR Section 2.3.11 on the volcanic eruption modeling case, additional information provided in response to the NRC staff's request for additional information (DOE, 2009bk–bm), the supporting DOE information on atmospheric transport of contaminated tephra presented in SNL (2007ab), and information published in peer-reviewed literature (e.g., Suzuki, 1983aa; Hurst and Turner, 1999aa; Andronico, et al., 2008aa).

Model Integration

Potentially relevant FEPs in DOE's TSPA model were listed in SAR Table 2.2-1. Model abstractions comprise FEPs that have been screened in from the applicant-conducted scenario analysis. SER Section 2.2.1.2.1 documents the NRC staff's evaluation of the DOE scenario analysis and FEP screening. As part of the review of the volcanic modeling case, the NRC staff examined DOE's information on igneous-related FEPs. In SER Section 2.2.1.2.1.3.1, the NRC staff determined that DOE identified a complete list of FEPs for the volcanic exposure scenario, including airborne transport. In SER Section 2.2.1.2.1.3.2, the NRC staff determined that DOE acceptably screened all FEPs for the volcanic exposure scenario, including airborne transport. DOE excluded a related FEP from consideration due to low consequence (FEP 1.2.04.07.0B); this FEP is concerned with leaching of radionuclides from tephra on the surface into the subsurface and into groundwater, whereby radionuclides could be dispersed via the groundwater transport pathway. In SER Section 2.2.1.2.1.3.2, the NRC staff determined that DOE's exclusion of FEP 1.2.04.07.0B, identified in DOE (2009ab, Enclosure 1), is acceptable because the possible contribution to mean dose via this mechanism is considerably lower than contributions from the other modeling cases. The NRC staff's review of the airborne transport abstracted model evaluates the applicant's implementation of the only included FEP (1.2.04.07.0A) associated with airborne transport modeling.

FEP 1.2.04.07.0A describes finely divided waste particles that may be erupted from a volcanic vent and deposited on the land surface from a waste-particle-contaminated ash (tephra) cloud or plume. This FEP is included in the performance assessment through the modeling of an eruption that includes airborne transport and tephra deposition (SNL, 2008ab).

The NRC staff evaluated modeling assumptions and integration in the DOE airborne transport abstracted model. DOE assumed that the tephra in a future eruption would be dispersed by a violent Strombolian eruption column, characterized by heating of entrained air. In the DOE model, the vertical atmospheric transport of the fragmented magma and gas mixture is represented as a thermally buoyant plume. The column rises to an altitude of neutral buoyancy compared to the surrounding atmosphere, at which point it spreads laterally and the resulting plume (ash or tephra cloud) is transported downwind. DOE modeled the dispersal of ash using a well-accepted and peer-reviewed model that Suzuki (1983aa) originally developed. Because the scale of horizontal atmospheric turbulence is much greater than the scale of vertical turbulence for violent Strombolian plumes within tens of kilometers [up to 50 mi] of the vent, horizontal diffusion is the dominant factor in the model in determining the width of the plume as it advects downwind. Therefore, the Ashplume conceptual model DOE used to simulate tephra deposition is based on a two-dimensional advection-diffusion model in which turbulent diffusion is considered only in the horizontal plane. The NRC staff finds the applicant's treatment of these processes and assumptions to be acceptable because it is based on well-established modeling techniques (Sparks, et al., 1997aa).

In its tephra-fall modeling, DOE assumed violent Strombolian activity for the entire duration of the tephra-forming activity. The NRC staff finds that this assumption is reasonable. Although violent Strombolian eruptions may have interruptions in activity, modeling such an eruption as a continuous process will not underestimate the amount and character of tephra transported during an eruption and, thus, will not underestimate dose to the RMEI. Consequently, the NRC staff considers the DOE simplification of a single, energetic eruptive style to be acceptable. In addition, the applicant acceptably determined the initial plume rise velocity using a relationship among eruptive power, eruption duration, conduit diameter, and eruption column height and plume conditions for violent Strombolian eruptions, which would not underestimate the amount and character of tephra fall deposit (and incorporated waste) modeled for an eruption.

DOE calculated tephra and waste deposited at the RMEI location, a point located 18 km [11 mi] south of the volcanic vent, as outlined in SNL (2007ab, Section 6.5.2.1.17 and Table 8-2). DOE's results showed that the point assumption is conservative and does not underestimate the concentration of waste deposited in the 33-km^2 [13-mi^2] RMEI location used in the Fortymile Wash Ash Redistribution (FAR) model, as part of the Fortymile Wash catchment basin, as identified in DOE (2009bk, Enclosure 9). Results of supplemental ASHPLUME simulations, discussed in DOE (2009bk, Enclosure 9), demonstrated that single point values for tephra and waste that were measured 18 km [11 mi] south of the potential repository would typically represent the maximum concentrations that would be deposited at the RMEI site. The NRC staff finds the applicant's assumption to be acceptable because the DOE model simulation results referred to previously demonstrate that the point assumption is a conservative approach.

With respect to the particle sizes in the atmospheric transport model of tephra, the model Suzuki (1983aa) developed is appropriate for particles of a mean diameter greater than tens of micrometers [about 6×10^{-4}–1.2×10^{-3} in]. The NRC staff finds that this cutoff is generally accepted to be the lower limit for the importance of simple gravitational settling of particles because the fall of smaller particles is governed by different physical laws than the Stokes settling assumption of the Suzuki model (Suzuki, 1983aa; Heffter and Stunder, 1993aa). The mass of ash particles smaller than 15 microns [6×10^{-4} in] is less than 2 percent of the total ash mass in most of the applicant's model realizations (DOE, 2009bm), which the NRC staff confirms is consistent for violent Strombolian tephra deposits (Andronico, et al., 2008aa). Therefore, the NRC staff concludes that the range of tephra sizes that are considered in the Ashplume model is appropriate for evaluating the airborne transport of radionuclides by a violent Strombolian volcanic eruption.

Data Sufficiency and Data Uncertainty

The NRC staff evaluated data sufficiency and uncertainty in the ASHPLUME model. DOE values for many input parameters in the Ashplume model were developed using analogous small-volume basaltic volcanic systems (SNL, 2007ab,ae), which is the commonly used approach when modeling volcanic eruptions (e.g., Hill, et al., 1998aa). Analogous historic violent Strombolian eruptions cited as sources of parameters for use in the Ashplume model include Tolbachik, Russia (1975); Parícutin, Mexico (1943–1952); and Cerro Negro, Nicaragua (1850–1999). DOE developed eruption parameter distributions on the basis of empirical relationships from available field data from the deposits of the aforementioned eruptions. The NRC staff reviewed the range of important parameters DOE derived from analog volcanoes (SNL, 2007ab,ae). The NRC staff's evaluation included the parameters for magma partitioning factor, magma volume, ash particle diameter, eruptive power and duration, wind direction and speed, and an eruption column parameter (the diffusion constant, β). These parameters are evaluated individually in the following paragraphs.

Magma Partitioning Factor

The applicant accounted for a proportion of disrupted and erupted waste that is partitioned into erosion-resistant products (scoria cone and lava flows) by using a magma partitioning factor with a uniform distribution from 0.1 to 0.5 (also discussed in SER Section 2.2.1.3.10). This range is based on volumetric proportions of cones and lava flows to total erupted volume estimated from field measurements at analog volcanoes, as identified in SNL (2007ab, Section 6.5.2.22). DOE identified in SAR Section 2.3.11.4.1.1.3 that very little erosional modification of lava fields of nearby ~ 350,000-year-old volcanoes (Little Black Peak and Hidden Cone) has occurred. The applicant also indicated that little if any cone scoria at the ~ 77,000-year-old Lathrop Wells volcano has yet been remobilized to the base of the cone where it would be available for fluvial transport (SAR Section 2.3.11.4.1.1.3). On the basis of its review of SAR Section 2.3.11.4.1.1.3, the NRC staff finds the assumption that the proportion of magma ending up as cones and flows would not contribute to dose at the RMEI location to be reasonable.

DOE used data from eight analog volcanic eruptions to determine a range of 0.1 to 0.5 (mean 0.3) for the magma partitioning factor (CRWMS M&O, 2003; SNL, 2007ab) (i.e., from 0.1 to 0.5 of the total erupted magma volume will be dispersed as tephra); estimated doses from the volcanic ash exposure scenario are directly proportional to the magma partitioning factor. However, in the NRC staff's view, not all the analog eruptions cited by DOE showed significant violent Strombolian behavior, and those eruptions that did tended to have magma partitioning factors greater than 0.3. The applicant used analog data to support the parameter range for the magma partitioning factor, including the lower part of the range with values less than 0.3, as discussed in DOE (2009bk, Enclosure 8, Table 1). DOE (2009bk, Enclosure 8) stated the tephra component is small in many of these basaltic analog eruptions, which is not typical of violent Strombolian eruptions. The NRC staff finds that eruptions of volcanoes, such as Cinder Cone in California, cited as an analog in

DOE (2009bk, Enclosure 8), featured only very minor phases of the violent Strombolian activity representative of this style of eruption (Heiken,1978aa). Notwithstanding this limitation, the NRC staff estimates that constraining the magma partitioning factor to higher values between 0.3 and 0.5 (mean value of 0.4) would imply an increase in calculated doses by a factor of 1.33 compared to results for the full parameter range (between 0.1 and 0.5, with a mean value of 0.3). The NRC staff finds that this small amount of uncertainty on estimated doses due to the magma partitioning factor is offset by the DOE conservative assumption for atmospheric transport that the entire eruption is modeled as violent Strombolian activity in the DOE TSPA. Therefore, the NRC staff concludes that the DOE Ashplume model adequately represents the airborne transport of radionuclides and does not underestimate doses for the volcanic ash exposure scenario.

Magma Volume

The NRC staff reviewed the data synthesis and documentation on likely magma volumes for future eruptions provided by the applicant in SNL (2007ab) and SNL (2007ae, Section 6.3.4.4). The NRC staff compared the DOE range of eruptive volumes to independent estimates (Jarzemba, 1997aa; Jarzemba, et al., 1997aa; Hill and Connor, 2000aa) and finds them to be in reasonable agreement. The upper end of this range was based on doubling of the Lathrop Wells tephra volume, which was "intended to capture the upper end of the range of uncertainty," and the lower end of the range was based on a calculated tephra fall volume for the smallest cone in the region, Northeast Little Cone, as identified in SNL (2007ae, Section 6.3.4.4) and DOE (2009bj, Enclosure 3). In the Ashplume model, DOE used the relationship among eruption power, eruption volume (rather than tephra volume), and eruption duration to constrain the range of total mass of tephra. As discussed in SNL (2007ab, Section 6.5.2.1), DOE constrained eruptive power on the basis of a few observed violent Strombolian eruptions. DOE showed that the tephra-fall volume in the DOE TSPA ranged from 0.004 to 0.14 km^3 [0.001 to 0.03 mi^3] with a mean value of 0.038 km^3 [0.01 mi^3], as identified in DOE (2009bj, Enclosure 3, Figure 1). Considering that the applicant uses the conservative assumption of an entirely violent Strombolian eruption, the NRC staff finds that the resulting range is reasonable because the mass fluxes of magma from the vent are within observed limits.

Ash Particle Diameter

The NRC staff reviewed the DOE log-triangular distribution for the mean ash particle diameter {minimum value of 0.001 cm [4×10^{-4} in]}, a mode value of 0.01 cm [4×10^{-3} in], and a maximum value of 0.1 cm [0.04 in] (SNL, 2007ab) and the presented rationale (SNL, 2007ae), which were derived by the applicant from data obtained at analogous small-volume basaltic volcanoes. The NRC staff considers this particle size range to be representative of available information on violent Strombolian eruptions (e.g., Andronico, et al., 2008aa; Pioli, et al., 2008aa). For comparison, NRC independently assessed basaltic tephra-fall deposits from the 1995 eruption of Cerro Negro (Hill, et al., 1998aa) and determined an average particle diameter of 0.07 cm [0.03 in]. This diameter is within the range of average ash-particle diameters considered by the applicant.

Eruptive Power and Duration

The applicant analyzed the eruptive parameters of analog volcanoes to develop the range and parameter distribution for eruptive power. The NRC staff compared the DOE eruptive power parameter values to an independent NRC estimate from Leslie, et al. (2007aa, Table 16-1). The DOE range was broader than the independently estimated NRC range, but the majority of the sampled values in the DOE TSPA fall within the NRC range. The geometric means for the two distributions are similar. The NRC staff finds the DOE parameterization for eruptive power acceptable because the applicant appropriately used analog volcano data to develop its parameterization for eruptive power. Duration, when considered with erupted volume, is one indicator of eruptive power, and hence, partly controls the height to which the eruption column reaches. The NRC staff finds the DOE parameterization for eruption duration acceptable because it is consistent with the range of observations from analog eruptions.

The upper part of the applicant's eruptive power range for possible future basaltic eruptions at Yucca Mountain leads to modeled eruption column heights of up to 8.2 km [5.0 mi] (SNL, 2007ab). The NRC staff finds this upper bound for a violent Strombolian eruption consistent with results from studies of historic eruptions (Pioli, et al., 2008aa). The NRC staff notes that SAR Sections 2.3.11.1 and 2.3.11.5 mentioned the ability of Ashplume to model eruption columns up to 13 km [8.1 mi] high. To build confidence in the Ashplume model, the applicant exercised this extended upper range for column height to model tephra dispersal for a volcano in New Zealand with a different eruption type, as identified in SNL (2007ab, Appendix J, p. J-23), and compared it with published results on tephra dispersal for that volcano. The NRC staff notes that column heights above 9 km [5.6 mi] are appropriate for modeling the eruption in New Zealand, but such column heights are not appropriate for a violent Strombolian eruption at Yucca Mountain, as identified in SNL (2007ab, Appendix E, p. E–3). The applicant clarified that column heights in the Ashplume realizations ranged from lower values of about 2 km [1.2 mi] up to a maximum value of 8.2 km [5.0 mi] in the DOE TSPA, as identified in DOE (2009bk, Enclosure 1). Consistent with Jarzemba (1997aa), DOE used observations from analog volcanoes to develop this range of eruptive power. Because the technical basis provided in DOE (2009bk, Enclosure 1) to support this range of column heights shows that the heights did not exceed 8.2 km [5.0 mi.], the NRC staff finds the supporting data and treatment of parameter uncertainty to be acceptable for modeling a future violent Strombolian-eruption-style event at Yucca Mountain.

Wind Direction and Speed

The applicant developed distribution functions for wind speed and wind direction from data provided by National Oceanic and Atmospheric Administration (NOAA) (2004aa). The full range of wind speeds from near zero to the maximum winds observed at the higher altitudes was represented in the wind-speed distribution used in TSPA analyses (SNL, 2007ab). The applicant accounted for uncertainty by stochastically sampling wind speed and direction for each eruption realization. DOE (2009bk, Enclosure 1) provided a technical basis by demonstrating that wind speed and wind direction are not correlated at different altitudes in the Ashplume model. DOE assigned the same wind speed for the top and lower heights within the column. NOAA (2004aa) showed that for the Yucca Mountain region, wind speeds tend to be greater at higher altitudes (a normal situation). Because DOE sensitivity analyses in SNL (2007ab, Appendix C) and DOE (2009bk, Enclosure 1) support modeled eruption column heights only up to 8.2 km [5 mi], whereas NOAA (2004aa) included wind data for higher altitudes up to 13 km [8 mi], NRC staff considers that the applicant's assignment for wind speed will not underestimate the dispersal characteristics of a violent Strombolian eruption. The NRC staff's review of the DOE sensitivity analyses (SNL, 2007ab) shows that the low concentration of tephra at the RMEI location after redistribution is relatively insensitive to variable wind conditions during an eruption. The NRC staff finds the assumption of constant wind speed and direction during an eruption acceptable because it will not significantly affect performance assessment results. For these reasons, the NRC staff finds that the applicant provided sufficient data and adequate documentation (SNL, 2007ab,ae) to support the selection of parameter ranges for wind speed.

Eruption Column Parameter

Although DOE determined through sensitivity analyses that the column diffusion constant (β) was not an influential parameter, the NRC staff reviewed the DOE application of this parameter in the Ashplume model because it controls the vertical distribution of the mass of tephra particles within the eruption column and helps determine the height at which particles exit the column and enter downwind atmospheric transport. The parameter range for β that DOE used is 0.01 to 0.5. Values at the lower end of the distribution lead to more of the tephra mass diffusing (falling) from the eruption column at relatively low altitudes in the modeled eruption (SNL, 2007ab). The applicant modeled (DOE, 2009bk) small tephra particle diameters {e.g., 0.005 cm [0.002 in]}, relatively high initial-rise velocities {e.g., 9,000 cm/s [3,543 in/s]}, and column diffusion coefficient values (β) less than 0.3 to support upward-concentration particle distributions at realistic heights for violent Strombolian eruption columns. Using the 1995 Cerro Negro eruption in Nicaragua as an analog, the NRC staff performed independent sensitivity analyses for β with a different theoretical model for tephra dispersal. The NRC staff results showed variations in deposit thickness up to a factor of approximately two over a range of distances on the order of 18 km [11 mi] and beyond (Hill, et al., 1998aa; Winfrey, 2005aa; Janetzke, et al., 2008aa). DOE (2009bk) estimated that varying β from 0.01 to 0.5 reduces the estimated waste concentration by less than 30 percent. On the basis of its review of the DOE result and general agreement with the confirmatory analyses cited above, the NRC staff concludes that the applicant's modeling of tephra dispersal from violent Strombolian eruption columns using β values less than 0.3 would not significantly affect dose estimates and is therefore acceptable.

Model Output: Waste Concentration in Tephra

The NRC staff evaluated the outputs from the DOE abstracted model, described in SN (2007ab, pp. 6–10), on airborne transport for the waste concentration in tephra and its spatial variation with distance and direction from the vent. Although DOE did not determine the waste incorporation ratio to be a significant parameter, the NRC staff evaluated the information DOE presented (SNL, 2007ab; SAR Section 2.3.11.4.2) and analyzed waste incorporated into magma based on its independent analyses (Codell, 2004aa). The NRC staff also independently determined that DOE's tephra and waste incorporation analysis combined particles according to the compatibility of their respective particle size distributions, which is an acceptable approach because it is consistent with conceptual models of eruptive conduit and fragmentation processes (Andronico, et al., 2008aa; CRWMS M&O, 2003; Pioli, et al., 2008aa; SNL, 2007ab; SNL, 2007ae; Sparks, et al., 1997aa). In DOE (2009bk, Enclosure 2) and DOE (2009bm, Enclosures 2 and 3), the applicant provided information on the modeled spatial variation of waste concentration in tephra, which demonstrates how much calculated concentrations varied within the same realization (i.e., how the concentration at downwind distances differed from waste concentrations in tephra closer to the vent). On a per-mass basis, waste constitutes a very small fraction of the mass in tephra deposits. Specifically, the applicant showed the mass of waste per unit mass of tephra was between 10^{-5} and 10^{-8} in deposits for two representative realizations in DOE (2009bk, Enclosure 2, Figure 1) and DOE (2009bm, Enclosure 3, Supplemental Figure 1). In DOE (2009bm, Enclosures 2 and 3), the applicant clarified that these values correspond to a single waste package and do not account for the partitioning of waste into scoria cone and lava flows.

In the report for Center for Nuclear Waste Regulatory Analyses Task P–15, (CNWRA, 2007aa, Attachment P–15, p. P15 B–4), the NRC staff compared the simulated waste concentrations in tephra to independent estimates of waste concentration in tephra from a subset of realizations, with the largest values for a single metric ton of waste erupted. The NRC staff also accounted for a typical amount of waste entrained in a simulated eruption. The NRC staff's estimates of waste concentration in tephra deposits were found to be in general agreement with the DOE values. On the basis of the variation in waste concentration provided by DOE, the NRC staff notes that variations in waste concentration over the land area representing the RMEI location (i.e., the alluvial fan of Fortymile Wash) are not expected to be large within individual realizations. The NRC staff finds acceptable DOE's explanation that these variations in waste concentration relate to particle size effects on atmospheric transport, because, on the basis of general tephra deposit characteristics, smaller sizes of combined tephra and waste particles would represent a greater portion of the deposit at farther distances from the vent, as described in DOE (2009bk, Enclosure 2, Section 1). Therefore, the NRC staff finds the applicant's model to be acceptable with respect to the fraction of waste in the tephra deposit.

Model Uncertainty

The NRC staff evaluated model uncertainty in the DOE abstracted model for airborne transport. DOE addressed model uncertainty by considering in SNL (2007ab) several different and widely accepted alternative models, including a Gaussian plume model (PUFF), a gas-thrust code (ASHFALL; Hurst and Turner, 1999aa), and TEPHRA (Winfrey, 2005aa). The applicant also evaluated an alternative igneous source term model developed by the NRC staff (Codell, 2003aa) to investigate the processes of waste fragmentation and incorporation into the tephra and determined that this alternative model was not significantly different from Ashplume (SNL, 2007ab). Codell (2003aa) reached the same conclusion as the applicant. The applicant specifically chose the Ashplume model (Jarzemba, et al., 1997aa) because it incorporates both tephra dispersal and waste incorporation necessary for performance assessment analyses. However, the applicant also accounted for model uncertainty by considering and evaluating several alternative conceptual models for downwind transport of tephra (and waste) from violent Strombolian eruptions. The results of all of these models are in general agreement (SNL, 2007ab), and, thus, the NRC staff concludes that the DOE treatment of model uncertainty is acceptable.

Model Support

The applicant supported its model results with (i) an independent technical evaluation, SNL (2007ab, Appendix E); (ii) a comparison to field observations from an analog eruption; and (iii) a comparison to another airborne transport code. With regard to the latter, the ASHFALL code (Hurst and Turner, 1999aa) represents sophisticated models incorporating the physics of tephra transport and deposition but does not include radionuclide transport. ASHFALL uses the same advective-diffusive relationships as Ashplume but employs time- and altitude-dependent wind conditions for tephra dispersal and more explicitly treats tephra particle settling velocities. The Ashplume code was used by the applicant in two sets of model runs to reproduce published output from the ASHFALL code for constant wind conditions and a variable wind field. The applicant's comparison of Ashplume and ASHFALL model computations in SNL (2007ab. Appendix J) indicated that Ashplume calculates tephra thicknesses that are within a factor of two of ASHFALL results. The applicant also supported its abstracted model with a comparison to field measurements of tephra thickness for the 1995 eruption at Cerro Negro, Nicaragua, outlined in SNL (2007ab, Appendix L). The NRC staff reviewed the information set forth in (i) to (iii) of this paragraph and found the calculated thicknesses to be in reasonable agreement with measured values described in SNL (2007ab, Figure L–6), and, thus, the NRC staff finds that the DOE model support for the atmospheric transport of radionuclides is acceptable.

Summary of NRC Staff's Findings for the Airborne Transport Model

On the basis of this evaluation, the NRC staff finds that the DOE ASHPLUME model calculations acceptably estimate the airborne transport of radionuclides. DOE acceptably integrated its model of airborne transport by incorporating the important processes associated with FEP 1.2.04.07.0A. Documentation the applicant provided in SNL (2007ab,ae) adequately described parameterization of the abstracted model. DOE acceptably

- Characterized and propagated data uncertainty by using distributions of important parameters that are technically defensible, reasonably account for uncertainties, and do not underrepresent the risk estimate

- Addressed model uncertainty by considering and evaluating several alternative conceptual models for airborne transport

- Supported its model by comparing the results of ASHPLUME to comparable ASHFALL computational results and to field measurements

Therefore, the NRC staff finds that the results of the airborne transport modeling are acceptable for further use in the volcanic ash exposure scenario.

In SER Section 2.2.1.4.1.3.3.2, the NRC staff evaluates the applicant's demonstration of compliance with the postclosure individual protection standard (or calculated dose to the RMEI) for the volcanic ash exposure scenario. In that evaluation, four input quantities of the airborne transport abstraction (fraction of waste incorporated with tephra to the total waste erupted, tephra volume, tephra density, and ash areal concentration) relate to the representation of the performance assessment for atmospheric transport and are evaluated in the following four paragraphs.

In the DOE model for an extrusive volcanic event, the amount of waste incorporated in tephra scales directly with the magma partitioning factor (SNL, 2007ab). On the basis of the relative proportions of eruptive products at analog volcanoes, the applicant selected a range between 0.1 and 0.5 for this parameter, which acts as a direct multiplier on the eruption source term and eruptive dose. Thus, the NRC staff finds that a fraction of 0.3 for waste incorporated with tephra to the total waste erupted is acceptable for use in the representation calculation in SER Section 2.2.1.4.1.3.2.

The applicant analyzed tephra-fall volumes for Quaternary Period (approximately last 2 million years) volcanoes in the Yucca Mountain region by comparison with fall-cone and cone-lava volume ratios for well-preserved young basaltic volcanoes. Violent Strombolian volcanic activity usually yields tephra-fall deposit volumes roughly twice those of the volcanic cone (Hill and Connor, 2000aa). For the Lathrop Wells volcano, an appropriate example of the type of eruptive event that could disrupt the potential repository at Yucca Mountain, the estimated tephra volume is 0.07 km^3 [0.017 mi^3] (SNL, 2007ae). Thus, the NRC staff finds that a tephra volume of 0.07 km^3 [0.017 mi^3] is acceptable for use in the representation calculation in SER Section 2.2.1.4.1.3.2.

Bulk *in-situ* density of tephra-fall deposits typically ranges from 0.3 to 1.5 g/cm^3 [0.01 to 0.05 lb/in^3] (Sparks, et al., 1997aa), but is rarely directly measured for basaltic volcanoes. Blong (1984aa) measured a range of tephra deposits that have a density of approximately 1.0 g/cm^3 [0.04 lb/in^3]. DOE (SNL, 2007ae) used 1.0 g/cm^3 [0.04 lb/in^3] for TSPA calculations on the basis of both this value from Blong (1984aa) and a normal distribution of deposit densities ranging from 0.3 to 1.5 g/cm^3 [0.01 to 0.05 lb/in^3] with a mean of 1.0 g/cm^3 [0.04 lb/in^3]. Thus, the NRC staff finds that a tephra density of 1 g/cm^3 [0.04 lb/in^3] is acceptable for use in the representation calculation in SER Section 2.2.1.4.1.3.2.

The ash areal concentration was derived from an assumed thickness of deposited tephra. In SNL (2007ac, Appendix G), DOE calculated an arithmetic mean of 0.97 cm (~ 1 cm) [0.54 in] for tephra thickness at the RMEI location for a wind direction fixed to the south. Because the 1-cm [0.54-in] assumption in DOE's areal concentration calculation is supported by modeling results of tephra thickness, the NRC staff finds that an ash areal concentration of 10,000 g/m^2 [0.014 lb/in^2] for an assumed 1-cm [0.54-in]-thick deposit is acceptable for use in the representation calculation in SER Section 2.2.1.4.1.3.2.

2.2.1.3.13.3.1.2 Tephra Redistribution in Fortymile Wash

The NRC staff conducted a risk-informed review of tephra redistribution, concentrating on those aspects important to the volcanic ash exposure scenario in the DOE performance assessment, as given next.

Important Aspects of Tephra Redistribution

DOE modeling of redistribution of tephra includes fluvial (running water or stream) transport of contaminated tephra within the Fortymile Wash catchment basin, mixing and dilution with uncontaminated sediment, and deposition of the tephra-sediment mixture on the Fortymile Wash alluvial fan at the RMEI location. These processes are modeled by DOE to occur instantaneously, thus not allowing for any radioactive decay of contaminated tephra before its deposition at the alluvial fan location. In the DOE model, on the alluvial fan, tephra is deposited in distributary channels by redistribution processes and on interchannel divides from airborne transport.

DOE performance assessment results for the volcanic ash exposure scenario are influenced by radionuclide concentrations in soil from both distributary channels and interchannel divides, as described in DOE (2009bk, Enclosure 5, Figure 1). Radionuclides in distributary channels contribute dose to the volcanic ash exposure scenario from the large number of realizations that result in an initial tephra deposit in the Fortymile Wash catchment basin. Fluvial sediment in distributary channels contributes more (two thirds, on average, in the DOE model) to the airborne particle concentration at the RMEI location than soils on interchannel divides (one third).

The NRC staff reviewed auxiliary Monte Carlo simulations by Pelletier, et al. (2008aa) that indicated the fluvial transport abstracted model reduced the concentration of tephra in sediment deposited in distributary channels by a factor of about 100 (arithmetic mean of 20 simulations), compared to the tephra concentration in the original tephra deposit. DOE expects any waste attached to tephra particles to remain attached during fluvial transport and, thus, expects that any reduction in tephra concentration from fluvial transport should reduce waste concentration by the same amount, as outlined in SNL (2007av, Section 5.2.5). On the basis of its review of DOE's sensitivity analyses in SNL (2007av, Section 6.6.1, Figures 6.6.1-1 to 6.6.1-3), the NRC staff notes that realizations with the largest waste concentrations were most sensitive to critical slope and scour depth (the depth to which flowing water will erode (pick up) and move sediment in a stream channel), in that order, and slightly sensitive to drainage density. Because DOE included waste dilution during fluvial transport in the FAR model, the NRC staff also focused its review on modeling assumptions and model support.

Summary of DOE's Approach on the Tephra Redistribution Model

The applicant's model of radionuclide redistribution in Fortymile Wash for the volcanic ash exposure scenario was described in SAR Section 2.3.11. In SAR Table 2.3.11-1, DOE identified the FEPs included in the TSPA model.

Following deposition of contaminated tephra (SER Section 2.2.1.3.13.2) from a potential eruption where a volcanic conduit intersects waste packages, the DOE tephra redistribution model accounts for the mobilization of contaminated tephra in the Fortymile Wash catchment basin, dilution of contaminated tephra with uncontaminated sediments in fluvial (stream) channels, and fluvial deposition at the location of the RMEI. Fortymile Wash lies east of the proposed repository, which DOE showed to be the most likely direction for tephra dispersal at typical heights for violent Strombolian eruption columns (SAR Figure 2.3.11-15). DOE developed the FAR Version 1.2 code, referred to hereafter as the FAR model, and incorporated this code into its TSPA as a dynamically linked library. The tephra redistribution is abstracted to occur instantaneously, as noted above, (i.e., radionuclide waste transport to the RMEI is instantaneous) (SAR Section 2.3.11.4.2.3.1). Eolian (wind-induced) processes are not included in the tephra redistribution model.

In the DOE model described in SNL (2007av), tephra is mobilized and transported downstream if it is initially deposited either on slopes steeper than a critical slope angle or in active channels with stream power exceeding a threshold value. Critical slope parameter values were determined from field measurements at analog sites. DOE determined active channel networks from digital elevation model data and drainage density estimates, on the basis of calibrations to field observations. Channel geomorphology in the Fortymile Wash catchment basin was based on recent observations and is modeled as time invariant. Effects on surface slope, elevation, stream power, and drainage density due to the presence of an initial tephra deposit and its weathering over time were not modeled. DOE considered these effects within the context of existing parametric values and propagated uncertainty, and exclusion of these effects from the model is not expected to significantly change the model results, as described in DOE (2009bk, Enclosure 10, Section 1).

DOE used scour depth estimates to determine the mixing and dilution of tephra with uncontaminated channel sediments. After the 1995 flood event in Fortymile Wash, DOE measured scour depth and estimated a total scour depth to account for the cumulative effect of flood events over time for use in the DOE TSPA. Sediment transport time is not explicitly accounted for in the model for fluvial remobilization and tephra dilution. Instead, a simplification was made such that remobilized tephra would be instantaneously diluted in fluvial sediments and directly deposited at the alluvial fan (i.e., fluvial remobilization, dilution, and deposition occur at the same simulation timestep as initial tephra-fall deposition).

The Fortymile Wash alluvial fan is located at the southern end of the drainage system; DOE modeled it as active (distributary) stream channels and areas between channels (interchannel divides). In the DOE tephra redistribution model, the whole alluvial fan is assumed to be an area occupied by the RMEI (SAR Figure 2.3.11-13). Parameter values for the area of the Fortymile Wash alluvial fan and the fraction of that area associated with channels were determined from field measurements and soil geomorphic mapping. In the model of a future volcanic eruption, initial radionuclide concentrations on interchannel divides arise from original tephra-fall deposits across the fan. Redistributed tephra mixed with ambient sediment from the Fortymile Wash drainage system is deposited in distributary channels and not on interchannel divides. Radionuclide concentrations in distributary channels, therefore, include a mixture of redistributed tephra from the Fortymile Wash drainage system and any original tephra-fall deposits. DOE assumed redistributed tephra is transported as bedload material, which neglects the potential for silt-sized material to be transported in the suspended (streamflow-borne) load, past the RMEI location, and into the Amargosa River Valley. DOE considered alternative modeling approaches during the development and validation of the scour-dilution-mixing approach in its tephra redistribution abstracted model (SNL, 2007av). DOE also referred to model-confidence building, supporting comparisons, and sensitivity analyses documented in SNL (2007av) and a published application of the scour-dilution-mixing model to the area around the Lathrop Wells Volcano (Pelletier, et al., 2008aa).

Time-dependent radionuclide concentrations with soil depth in stream channels and on channel divide surfaces are the ultimate outputs of the tephra redistribution model. The later SER Section 2.2.1.3.13.3.1.3 evaluates the time-dependent vertical migration of radionuclides in soil for the volcanic ash exposure scenario. In the biosphere model, reviewed in SER Section 2.2.1.3.14.3, the FAR model outputs are combined with biosphere dose conversion factors in the DOE TSPA to estimate annual doses to the RMEI (SAR Figure 2.3.10-10).

NRC Staff's Review of the Tephra Redistribution Model

The NRC staff reviewed SAR Section 2.3.11 on the volcanic eruption modeling case, additional information provided in response to the NRC staff's request for additional information (DOE, 2009bk–bm), the supporting DOE information on tephra redistribution presented in SNL (2007av), and information published in peer-reviewed literature (e.g., Pelletier et al., 2008aa).

Model Integration

Model abstractions comprise FEPs that have been screened in from the scenario analysis. In SER Section 2.2.1.2.1.3.1, the NRC staff determined that DOE had identified a complete list of FEPs for the volcanic exposure scenario, including tephra redistribution. In SER Section 2.2.1.2.1.3.2, the NRC staff determined that DOE had acceptably screened all FEPs for the volcanic exposure scenario, including tephra redistribution. DOE did not exclude any FEPs associated with this abstracted model. The NRC staff's review of the tephra redistribution abstracted model evaluates the applicant's implementation of the included FEP 1.2.04.07.0C, which accounts for the surface transport processes that redistribute radionuclides following the initial tephra-fall deposition. The NRC staff's evaluation of DOE's modeling assumptions used for the fluvial transport in the FAR model is addressed in the following paragraphs.

The applicant determined that wetter future climates would increase vegetation on hillslopes and reduce the amount of remobilized tephra from hillslopes into channels. The NRC staff finds that the applicant's determination is acceptable because it reasonably interprets the influence of vegetation on erosion. The applicant also pointed out that additional precipitation from a wetter climate in the future could increase the scour depth in channels, which was shown in SNL (2007av, Figure 6.6.1-2) to reduce initial radionuclide concentrations in channels. The

NRC staff finds DOE's approach, which does not quantify fluvial redistribution effects from wetter climates, is acceptable because this modeling approach would not underestimate radionuclide concentrations in sediment, as shown in SNL (2007av, Section 5.1.1).

The NRC staff evaluated the implementation of the channel area fraction in the FAR model, which also considered its coupling with the biosphere model in the DOE TSPA. The applicant assessed (i) the relative susceptibility of the two surfaces, interchannel divides and active fluvial channels, to airborne resuspension and (ii) the assumption that dose contributions from these two surfaces are proportional to their respective fractions of the total area of the alluvial fan. DOE concluded that differences in these two surfaces were accounted for in the DOE TSPA estimates for airborne particle concentration in DOE (2009bk, Enclosure 5, Figure 1). The applicant also showed that dose contributions for the volcanic ash exposure scenario were much greater from interchannel divides than from channels for thousands of years after repository closure. Contributions to dose from channels become important after the first 10,000 years as the importance of the inhalation pathway decreases relative to the other biosphere pathways. On the basis of its review of these results, the NRC staff concludes that differences in the relative susceptibility of the two surfaces to airborne resuspension, which could affect the relative dose contributions in addition to the area fraction, would not significantly affect dose estimates. Therefore, the NRC staff finds the DOE proportional relationship on dose contribution from these two surfaces, on the basis of their respective fractions of the total area of the alluvial fan, acceptable. The NRC staff also finds that the applicant's rationale for not explicitly modeling long-term changes to active channels within the depositional fan of Fortymile Wash is reasonable because it is not possible to accurately predict the location and size of future channels in this type of depositional system. On the basis of the reasons previously described in this paragraph, the NRC staff concludes that the DOE justification for the area fraction of active channels in the Fortymile Wash alluvial fan and associated parameters in their model is reasonable.

Although the applicant acknowledged that a future eruption could alter the channel geomorphology to some degree by deposition of fresh tephra, it assumed that eruption-induced changes would have little effect on the overall geomorphology of the Fortymile Wash catchment basin, as described in SNL (2007av, Section 5.1.4). The NRC staff finds the applicant's approach of basing fluvial redistribution on current channel geomorphology to be reasonable for modeling future fluvial redistribution and the associated uncertainties in estimating radiological doses. This approach is consistent with information from analog volcanoes such as Parícutin (Segerstrom, 1950aa, 1966aa; Inbar, et al., 1994aa), where original channels were reestablished in the decades following a tephra-fall eruption. SNL (2007av, Sections 5.1.3 and 6.1.2) provide DOE's justification for not modeling overbank deposition (i.e., deposition of fluvial tephra and sediment on interchannel divides) at the Fortymile Wash alluvial fan. The NRC staff concludes that this approach is reasonable because of the absence of fluvial overbank deposits on existing interchannel divides of the Fortymile Wash fan, as detailed in SNL (2007av, Appendix A).

Rather than accounting for significant rainfall and flooding events individually and tracking the movement of redistributed tephra from each event over time, the FAR model applies a representative deposit for long-term redistribution and dilution of tephra in the same simulation time step. Because FAR model results are integrated with biosphere modeling in the DOE TSPA, the NRC staff considered the coupling of the FAR and biosphere models in the evaluation of FAR model assumptions concerning time dependency. The applicant assessed the replenishment of contaminated fluvial deposits over time with respect to time-dependent estimates of resuspended airborne particle concentrations in the DOE TSPA, according to

DOE (2009bk, Enclosure 4). DOE clarified that the active outdoor category for RMEI activities includes time spent walking outdoors on uncompacted soil or tephra. The applicant also acknowledged that airborne particle concentrations would be higher in the DOE TSPA for walking on uncompacted tephra deposits following an eruption than during preeruption conditions. The NRC staff reviewed the DOE parameter ranges for airborne particle concentrations from resuspension following a volcanic eruption and finds them to be supported by published literature referenced in BSC (2006ad). On the basis of its review of information provided by the applicant [BSC (2006ad); SNL (2007av); DOE (2009bk, Enclosure 4)] and independent field measurements that the NRC staff conducted at analog volcanic sites [Hill, et al. (2000ab); Benke, et al. (2009aa)], the NRC staff concludes that the applicant's estimates of inhalation of resuspended particulates from tephra-fall deposits or redistributed tephra in fluvial sediments do not underestimate dose. The NRC staff also determines that the applicant's model integration in its TSPA for the volcanic ash exposure scenario, including the coupling of the assumption for instantaneous tephra remobilization in Fortymile Wash with time-dependent values for airborne mass loading, is acceptable for estimating doses over time. This approach is acceptable because, by using this instantaneous assumption, DOE couples the largest amount of remobilized contamination near the RMEI location with higher values for airborne mass loading following the volcanic event, which will overestimate doses from remobilized tephra. Because the applicant did not model further mixing and dilution, which could reduce radionuclide concentrations in the instantaneous deposit from subsequent flooding events in the Fortymile Wash drainage basin, as identified in SNL (2007av, Section 5.2.4), the NRC staff finds the applicant's assumption that the drainage system is open (SAR Section 2.3.11.4.4.3) to be acceptable. Therefore, the NRC staff concludes that modeling a representative deposit for long-term redistribution and dilution of tephra adequately represents the fate of material in this ephemeral drainage system and is an acceptable approach for calculating radiological consequences. Overall, the NRC staff finds the technical basis for this abstracted model in SNL (2007av) to be adequately described because SNL (2007av, Section 5) adequately addresses the assumptions of the FAR model.

In SNL (2007av, Section 1.2), the applicant acknowledged that eolian (wind) sediment transport is a significant geomorphic process in the Yucca Mountain region. The applicant clarified in DOE (2009bk, Enclosure 12, Section 1.1) that airborne resuspension of tephra deposits at the RMEI location is included in the DOE TSPA as a local-scale eolian redistribution process. The NRC staff notes that the DOE abstracted model did not model long-range eolian redistribution of tephra explicitly. In SNL (2007av, Section 5.2.2), the applicant also accounted for potential local-scale eolian transport of contamination in channel sediments onto interchannel divide surfaces by increasing the range for the channel fraction of land area in the Fortymile Wash alluvial fan.

As described in SNL (2007av, Section 5.2.2), direct tephra-fall deposition and fluvial processes dominate radionuclide concentrations in the soil and air at the RMEI location. DOE considered the long-range eolian transport of freshly deposited tephra south to the location of the RMEI from the Fortymile Wash catchment basin to be negligible on the basis of the applicant's characterization of the prevailing direction for strong southerly winds, as identified in SNL (2007ab, Appendix D) and Pelletier and Cook (2005aa). The applicant also considered relevant wind data in CRWMS M&O Site 9 (1997aa) and determined that southerly winds exhibited higher wind speeds compared to northeasterly winds, as described in DOE (2009bk, Enclosure 12, Section 1.2). Higher speeds for south-to-north winds would tend to drive the net transport of contaminated tephra toward the north and away from the RMEI location, as DOE (2009bk, Enclosure 12, Section 1.2) identified. The applicant, therefore, concluded that eolian transport of radionuclides deposited in the Fortymile Wash catchment basin to the RMEI would be negligible, as stated in SNL (2007av, Section 5.2.2). DOE further concluded that not modeling these eolian effects would tend to overestimate tephra and waste concentration at the location of the RMEI. The NRC staff concludes that predicted future south-to-north surface wind directions and associated eolian transport would tend to dilute radionuclide concentrations in the soil and the air at the RMEI location. The NRC staff, thus, finds that the applicant has provided a sufficient technical basis for not explicitly modeling long-range eolian transport of contaminated tephra to the RMEI location because the applicant's approach will not underestimate airborne concentrations of radionuclides at that location or dose to the RMEI.

Fluvial transport of sediment and tephra in Fortymile Wash is modeled by DOE as bedload transport. From its review of grain-size distributions and textural considerations by the applicant in SNL (2007av, Section 5.2.3), the NRC staff finds the DOE modeling approach to be acceptable because bedload transport would be the primary mode of fluvial transport of tephra and sediment in Fortymile Wash. Using grain-size data from analog volcanic eruptions, the applicant expects a range of tephra particle sizes with an approximate median value of 0.01 to 1.0 mm [0.0004 to 0.04 in], as identified in SNL (2007av, Section 5.2.3). With diameters between 0.002 to 0.05 mm [0.00008 to 0.002 in], as described in BSC (2006ah, Section 6.5.3.2), silt-sized particles represent a small portion of this range. The applicant considered that silt-sized material could be transported past the RMEI location in the suspended load (rather than in the bedload) and concluded that not modeling suspended load transport and deposition is conservative, as stated in SNL (2007av, Section 5.2.3). Because the amount of silt-sized material in the fluvial system is small with respect to the total amount of fluvial material and the DOE FAR model does not exclude tephra or waste associated with silt-sized particles from contributing to dose, the NRC staff finds that the DOE approach for modeling bedload transport and not including other transport mechanisms, such as suspended load, is acceptable. The NRC staff finds it acceptable that DOE neglected other fluvial transport processes in the DOE FAR model because the applicant's approach will not underestimate dose.

The NRC staff expects that density effects on combined waste-tephra particles would have a minor-to-negligible impact on bulk transport, mixing with clean sediment, and waste concentration in redistributed tephra for the large volumes of tephra. This is due to the relatively small ratio of waste concentrations per unit tephra mass, outlined in DOE (2009bk, Enclosure 2, Figure 1) and SER Section 2.2.1.3.10.3.2.3, which supports the applicant's conclusion in SNL (2007av, Section 5.1.5) that other processes (physical, chemical, or biological) for concentrating radionuclides at the RMEI location are negligible. To enhance confidence in its system description and model integration, the applicant had critical reviews performed on an earlier version of the technical basis document and included the review and resolution of comments in SNL (2007av, Appendix C, Section 7.3.2). The NRC staff evaluated the review comments and DOE responses and determined that this process provided additional support for the technical basis used in the FAR model.

Data Sufficiency and Data Uncertainty

The following subsections consider the data sufficiency and propagation of uncertainty presented for the FAR model parameters of critical slope, scour depth, and drainage density. On the basis of its review of the parameter distributions and treatment of uncertainty the applicant provided in SNL (2007av), the NRC staff finds that data synthesis and documentation adequately supported the range of sampled parameters, for the reasons discussed in each of the following subsections.

Critical Slope

The applicant collected field data from several analog volcanic sites near Flagstaff, Arizona (i.e., Rattlesnake Crater, Cochrane Hill, Moon Crater, and Cinder Cone), to determine the critical slope parameter range and represent the steepest slope for stable tephra deposits on hillslopes, as outlined in SNL (2007av, Section 6.5.2). The applicant assessed the measurement scale of the field observations for critical slope and representativeness of the 30 by 30-m [97 by 97-ft]-grid cell size for the digital elevation map of the Fortymile Wash drainage system. Slopes were measured at a scale of tens of meters [tens to hundreds of feet] in the field, and DOE concluded that the slope angles are representative of hillslopes in DOE (2009bk, Enclosure 6, Section 1.1). The NRC staff acknowledges that estimating slopes at the scale of an entire hillslope is appropriate for modeling tephra remobilization in the Fortymile Wash drainage system because the potential for tephra mobilization off hillslopes into channels depends on the entire hillslope traversed. DOE clarified that the representation of topography in the FAR model does not smooth steeper slopes and assessed the appropriateness of field data from analog volcanic sites to estimate fluvial erosion of tephra deposits in the Yucca Mountain region. Because the applicant assessed sheetwash and rilling at the field sites (where DOE measured critical slope values) and provided a justification for why rapid postdepositional erosion was not observed and why it is not expected at Yucca Mountain, as described in DOE (2009bk, Enclosure 6, Sections 1.2–1.4), the NRC staff finds acceptable DOE's conversion of field measurement data at analog volcanic sites to its parameter distribution for critical slope, as identified in SNL (2007av, Section 6.5.2). For the aforementioned reasons, the NRC staff finds that DOE parameter uncertainty for critical slope adequately represents fluvial erosion in Fortymile Wash for post-eruption conditions.

Scour Depth

SNL (2007av, Section 6.5.6) described the site-specific field data used by DOE to establish the parameter distribution for scour depth. Parameter values for scour depth were inferred from scour chains, installed by the U.S. Geological Survey (USGS) at the Narrows section of Fortymile Wash in the 1980s that were subsequently buried by flood sediment deposition about 10 years later in 1995, as shown in SNL (2007av, Table 6.5.6-1). In DOE (2009bk, Enclosure 10, Section 1), the applicant explained that other stream flow data (e.g., from the Amargosa Valley station) do not represent tributary conditions of the upper drainage basin. Therefore, on the basis of the scour-chain information cited above, the NRC staff finds DOE's use of scour measurements that were taken only at the Narrows, for determining scour depth and discharge in Fortymile Wash, is acceptable. Although these scour measurements indicated an upper-bound scour depth of 152 cm [5.0 ft], the applicant chose to limit the upper bound of the scour depth parameter to 122 cm [4.0 ft], as identified in SNL (2007av, Section 6.5.6 and Table 6.5.10-1), to add conservatism into the calculation. The average measured scour-depth value from the scour chain data, 73 cm [2.4 ft], was chosen as the lower bound (SNL, 2007av). In SNL (2007av, Figure 6.6.1-2), DOE showed that dilution would be reduced by selecting a shallower scour depth because a smaller amount of uncontaminated sediment would be mixed with contaminated tephra. The NRC staff finds the applicant's interpretation of these field data for scour depth, and the selected range for this parameter acceptable, because restricting scour depth to smaller values overestimates tephra and waste concentrations in fluvial sediment at the location of the RMEI.

The applicant based its determination of the parameter range for scour depth on site-specific field measurements at Fortymile Wash for current conditions without a surplus of tephra. The applicant assessed the potential effect of fresh tephra in Fortymile Wash on scour depth in channels and concluded that scour depth would not be affected by the proportion of tephra in channel sediments. According to DOE (2009bk, Enclosure 10, Section 1), the expected grain sizes of tephra are similar to the observed grain sizes for channel bed material; therefore, different hydraulic conditions were not expected for a fluvial deposit mixing sediment and tephra, as outlined in SNL (2007av, Section 5.2.3) and Pelletier, et al., (2008aa, p. 236). For distances up to a few kilometers [2–4 mi] from a volcanic vent, the NRC staff recognizes that violent Strombolian tephra deposits will consist of particles in the centimeter to millimeter [0.4 to 0.04 in] grain-size range, similar to grain-size distributions in bedload sediment from Fortymile Wash. The NRC staff confirmed that the applicant used values for scour depth that tend to underestimate dilution and overestimate dose, as described in DOE (2009bk, Enclosure 10, Section 1). On the basis of its review of this information, the NRC staff finds acceptable the DOE technical basis for scour depth and finds that the scour depth distribution adequately represents posteruption conditions when fresh tephra may be present in Fortymile Wash.

The NRC staff reviewed the applicant's scaling approach outlined in SNL (2007av, Section 6.3.3, Step 3) for computing scour depth at different locations in the Fortymile Wash drainage basin. DOE sampled values for the maximum scour depth within the drainage system and computed scour depth at other locations in SNL (2007av, Eq. 6.3-14). SNL (2007av, Figure 6.3.3-6) illustrates the variability of scour depth within the drainage basin. The NRC staff concludes that scour depth is dependent on the volumetric water flow (discharge) and acknowledges that the contributing area [outlined in SNL (2007av, Figure 6.3.3-5)] is a common approach for representing water flow at specific locations in the drainage system. Leopold, et al. (1966aa) studied another ephemeral drainage system in a semiarid area of New Mexico and found that most of the erosion occurred in a small percentage of the basin. This observation is consistent with the variability in scour depth the applicant presented in SNL (2007av, Figure 6.3.3-6). The applicant's scaling approach for scour depth is acceptable because the NRC staff finds that the variability in scour depth is based on estimates of slope angle [described in SNL (2007av, Figure 6.3.3-3)], contributing area , and stream order (Leopold, et al., 1964aa) in the drainage basin. The NRC staff concludes that the greatest scour depth is expected to occur where the main drainage channel narrows because scour depth is dependent on volumetric water flow and channel width. The applicant used site-specific field measurements at the Narrows to establish the parameter values for the maximum scour depth, which the NRC staff finds acceptable because, as stated above, values used for scour depth tend to underestimate dilution and overestimate dose.

Drainage Density

The drainage density is the ratio of the total length of streams to the area of the drainage system (length per unit area). Compared to critical slope and scour depth, performance assessment outputs are less sensitive to drainage density for the volcanic ash exposure scenario. DOE estimated drainage density from simulations of 34 channel heads on the eastern slope of Yucca Mountain. In SNL (2007av, Eq. 6.3-7 and p. 6-16), the applicant used the reciprocal of the drainage density as a stream power threshold for determining active channels within the Fortymile Wash catchment basin; in SNL (2007av, Section 6.5.6 and Figure 6.5.6-3), the applicant compared modeled channel head locations to actual locations and selected the drainage density that yielded the smallest average distance difference. The NRC staff finds this approach acceptable for determining the drainage density because it is based on a site-specific comparison and an established scientific relationship for stream power and contributing areas. The applicant assigned the parameter range for drainage density by considering other values that provided good agreement between observed and calculated channel heads on

Yucca Mountain. The NRC staff finds that the resulting parameter range for drainage density adequately accounts for uncertainty because it is based on a site-specific comparison and an acceptable interpretation of the results of the comparison.

DOE assigned values to the above-mentioned parameters (critical slope, scour depth, and drainage density) in the FAR model, according to the 30 by 30-m [97 by 97-ft] grid cells on the digital elevation (topographic) map of the Fortymile Wash drainage system. In the DOE TSPA analysis, sampling of these parameters is performed in the FAR model independently, without correlation between them [SNL (2007av, Table 6.5.10-1)]. The NRC staff concludes that representing topography and associated surface processes in the FAR model at a 30 by 30-m [100 by 100-ft] scale is a reasonable approach because a strong correlation between critical slope, scour depth, and drainage density is not expected, as these parameters characterize different aspects of the system (i.e., material stability on hillslopes; intrachannel flow and mixing; and active channel threshold within the network, respectively). The NRC staff finds that the DOE approach for parameter correlation and the spatial resolution used to determine parameter values in its model will not underpredict the consequences and alternative approaches will not significantly affect the dose.

Model Uncertainty

The NRC staff reviewed the DOE consideration of alternative models in Pelletier, et al. (2008aa) and SNL (2007av, Sections 6.2.2, 6.3.3, and 7.2.4). As previously described in the subsection entitled NRC Staff Perspective on Risk (SER Section 2.2.1.3.13.3), the NRC staff determined that the DOE abstracted model for fluvial transport and tephra dilution significantly reduces radionuclide concentration and influences the results of the volcanic ash exposure scenario. The FAR model is based on a modeling approach for the natural processes of scour, dilution, and mixing developed by DOE specifically for the Fortymile Wash drainage system. The classic dilution-mixing model has been generally considered the standard approach (Hawkes, 1976aa; Marcus, 1987aa) in the past. In DOE (2009bk, Enclosure 7, Section 1), the applicant considered the classic dilution-mixing model as appropriate only for tributary systems that discharge into the sea or a lake but not well suited for Fortymile Wash, because it is a tributary-distributary inland drainage system in a desert region. The applicant also identified other shortcomings with classic dilution-mixing models, such as the inability to model the vertical distribution of contamination in sediments and the dilution of contaminated tephra with uncontaminated sediments. Because the applicant considered scour-dilution mixing as the predominant mode of dilution, it concluded in DOE (2009bk, Enclosure 7, Section 1.1) that the scour-dilution-mixing model more accurately represented the processes at Fortymile Wash. The applicant further concluded in DOE (2009bk, Enclosure 7, Section 1.2) that dilution-mixing models were not directly applicable.

The NRC staff concludes that the FAR model adequately represents the Fortymile Wash drainage system and finds that DOE adequately considered alternative conceptual models to the scour-dilution-mixing approach used in the FAR model. Furthermore, the NRC staff finds the applicant’s treatment of alternative conceptual models sufficient because the analytic methods used by DOE in the FAR model are well established and the scour-dilution-mixing model is an appropriate model for the Fortymile Wash tributary-distributary inland drainage system in a desert region. Therefore, NRC staff finds that the FAR model would not underpredict the radiological consequences to the RMEI and is supported by applicable data.

Model Support

The NRC staff evaluated the applicant's model support for the FAR model. The applicant supported its model results with (i) independent technical evaluations, as outlined in SNL (2007av, Appendix C) and (ii) a peer-reviewed journal article (Pelletier, et al., 2008aa) that included a site-specific comparison for fluvial redistribution and dilution of tephra from the Lathrop Wells volcano. The NRC staff reviewed the independent technical evaluators' comments and found that DOE sufficiently responded to and resolved comments. The NRC staff also reviewed Pelletier, et al. (2008aa) and agrees with the article's conclusion that a scour-dilution-mixing approach is suitable for estimating downstream contamination concentrations when bedload transport dominates and overbank sedimentation is not significant. The NRC staff highlights that the applicant used a modeling assumption to constrain the tephra thickness in a channel routed to the Fortymile Wash alluvial fan to not exceed the scour depth, as described in SNL (2007av, p. 6-24). In SNL (2007av, Section 7.3.1.2), the applicant also pointed to the 77,000-year-old analog volcano at Lathrop Wells for observations of long-term storage of tephra below the scour depth. Although tephra stored below the scour depth in channels connected to the main Fortymile Wash channel or in unconnected channels cannot contribute to the concentration of radionuclides at the Fortymile Wash alluvial fan, total tephra stored below the scour depth was estimated by the applicant on the order of 3 to 7 percent of the amount of tephra-fall in the Fortymile Wash drainage system, as identified in DOE (2009bk, Enclosure 11). Because increases in scour depth tend to result in greater dilution of tephra with uncontaminated sediment in the DOE FAR model, the NRC staff concludes that tephra storage below the scour depth does not significantly affect the estimates of radionuclide concentrations in sediment at the location of the RMEI. The NRC staff acknowledges that sediment storage in channels is a well-recognized geomorphological phenomenon and finds that the DOE model for fluvial transport and dilution of tephra in Fortymile Wash has been adequately supported. For these reasons, the NRC staff finds the DOE abstracted model for the fluvial transport of tephra in Fortymile Wash is adequately supported.

Summary of NRC Staff's Findings for the Tephra Redistribution Model

On the basis of the evaluation described above, the NRC staff finds that the DOE FAR model provides an acceptable approach for calculating the redistribution of tephra to the RMEI location. DOE acceptably integrated its model of tephra redistribution by incorporating the important processes associated with the ash redistribution via soil and sediment transport FEP. The NRC staff finds that the parameter determinations for the redistribution of tephra and incorporated waste in the DOE abstracted model for fluvial transport in the Fortymile Wash catchment basin are adequately described and justified. The NRC staff concludes that the data sufficiently support this abstracted model in the TSPA. The NRC staff finds that data uncertainty was adequately characterized and propagated through the abstracted model for fluvial transport in Fortymile Wash. Parameter ranges were adequately described and justified. The NRC staff finds that DOE's treatment of alternative conceptual models is acceptable because no alternatives that are consistent with the presented data would affect the results significantly. Because DOE supported its model results with independent technical evaluations and a site-specific comparison for fluvial redistribution and dilution of tephra from the Lathrop Wells volcano, the NRC staff finds DOE's model support acceptable.

2.2.1.3.13.3.1.3 Downward Migration of Radionuclides in Soil

The NRC staff conducted a risk-informed review of the model of downward migration of radionuclides in soil, concentrating on those aspects important to the volcanic ash exposure scenario in the DOE performance assessment, as given next.

Important Aspects of Downward Migration of Radionuclides in Soil

In the DOE TSPA, both long- and short-term inhalation of radionuclides resuspended from the soil into the air significantly contribute to the total dose for the volcanic ash exposure scenario for 100,000 years. Because the short-term inhalation contribution is dominated by a much faster rate of reduction in airborne mass loading, the vertical migration of radionuclides in soil has greater potential influence on long-term inhalation dose. As previously discussed, contributions to total dose from the inhalation of particulates diminish after 100,000 years.

The DOE results indicated that processes for vertical migration of radionuclides into soil contained in the abstracted model for the volcanic ash exposure scenario result in a small reduction in radionuclide concentrations over time. The NRC staff obtained quantitative insights by investigating intermediate output files from the DOE TSPA. For unplowed soil, the reduction of radionuclide concentration due to vertical migration out of the resuspendable layer is gradual and slows with increasing time following initial deposition. On average, the radionuclide concentration in the resuspendable layer required approximately 20, 150, 700, and 4,000 years to decrease by a factor of 2, 4, 8, and 16, respectively, from its initial value in the DOE TSPA. For plowed soil, radionuclides are uniformly mixed within the tillage depth, and the time-dependent reduction in radionuclide concentration due to vertical migration is small (reduction by a factor of about 2 in 10,000 years). For fluvial channels, radionuclides are assumed to be well mixed within the fluvial sediment deposit, and the time-dependent reduction in concentration due to vertical migration of radionuclides out of either the resuspendable layer or tillage depth is minimal (leading to a reduction of less than a factor of 2 in 10,000 years). Sensitivity analyses performed by DOE indicated that radionuclide concentration is most sensitive to the diffusivity rate in soil on interchannel divides, followed by a lesser sensitivity to the diffusivity rate in fluvial channels. There was a negligible sensitivity to different values of permeable depth on the interchannel divides, as described in SNL (2007av, Section 6.6).

Summary of DOE’s Approach on the Radionuclides in Soil Model

The applicant’s modeling of radionuclide concentration migration into soil for the volcanic ash exposure scenario was described in SAR Sections 2.3.10, Biosphere Transport and Exposure and 2.3.11, Igneous Activity. In SAR Tables 2.3.10-1 and 2.3.11-1, DOE identified the FEPs included in the TSPA model.

Calculation of radionuclide concentrations with soil depth at the RMEI location is one of the main elements of the DOE FAR model for tephra redistribution. The applicant developed this part of the FAR model specifically for the Fortymile Wash alluvial fan, consisting of active channels and interchannel divide surfaces. The exposure of the RMEI to radionuclides in soil was modeled for two layers: (i) a thin upper surface layer from which particles can be suspended into the atmosphere by disturbances and (ii) a thicker, lower surface layer that may undergo mixing by agricultural practices such as tillage (SAR Section 2.3.10.2.6).

The FAR model includes the downward migration of radionuclides into soil for the volcanic ash exposure scenario. Incorporated into the DOE TSPA as a dynamically linked library, the FAR model is connected to the surface soil submodel of the DOE biosphere model, Environmental Radiation Model for Yucca Mountain Nevada (ERMYN), which calculates biosphere dose conversion factors for the volcanic ash exposure scenario on the basis of unit concentrations of radionuclides in volcanic ash deposited on the ground. The surface soil submodel is included in the biosphere analysis of all exposure pathways for the volcanic ash exposure scenario (refer to SAR Figures 2.3.10-8 and 2.3.10-10). Biosphere dose conversion factors are combined with time-dependent radionuclide concentrations in soil from the FAR model to estimate annual doses for the volcanic ash exposure scenario.

The applicant modeled time-dependent vertical migration of radionuclides in soil within the FAR tephra redistribution model as a diffusive process in one dimension. Values for radionuclide diffusivity and permeable depth differed between those areas on interchannel divides and those in fluvial channels. Field data on Cs-137 concentration profiles from the upper Fortymile Wash alluvial fan were used to determine radionuclide diffusivities and the associated uncertainties.

Permeable depths in soils were determined from field measurements in pits dug on interchannel divides of the Fortymile Wash alluvial fan and from USGS data on scour depth in fluvial channels. Although advection is not explicitly modeled, the applicant identified that diffusivity data accounted for all transport mechanisms, including advection and bioturbation. The applicant does not include effects of future climate change on the modeled processes and parameters in the tephra redistribution model, because DOE concluded that processes associated with future climate change would only decrease radionuclide concentrations in soils (SAR Section 2.3.11.4.4.3). In the FAR model, radionuclides are restricted from migrating into a deeper horizon by use of reduced permeability. The reduced permeability was assumed to be caused by a greater carbonate or clay content than the minor content in surface and near-surface soils. The applicant's approach limited possible reduction of radionuclide concentrations in the surface layer due to vertical migration over long time periods.

For the volcanic ash exposure scenario, the surface soil submodel calculates radionuclide mass concentrations in the tilled surface soil layer and in the thin resuspendable layer for noncultivated soil. In the DOE TSPA, radionuclide concentrations in the resuspendable layer and tilled soil are applied to different environmental exposure pathways. Weighting factors for land usage (e.g., fractions of land that are tilled and not tilled) are not included in the dose calculations. Igneous eruption dose calculations include weighting factors for the fraction of land area apportioned into active fluvial channels and interchannel divides. Volcanic material (basaltic tephra) is assumed to be mixed uniformly in tilled surface soil. Concentrations of radionuclides in tilled surface soil are factored into the pathway analysis for ingestion of contaminated crops and animal products. Inhalation and external exposure pathway calculations are dependent on radionuclide concentrations in the resuspendable layer. Because erosion and other surficial processes are accounted for in the tephra redistribution model, the DOE surface soil submodel does not include these processes for the volcanic ash exposure scenario.

NRC Staff's Review of the Downward Migration of Radionuclides in Soil Model

The NRC staff reviewed SAR Sections 2.3.10 and 2.3.11 on the volcanic eruption modeling case, additional information provided in response to the NRC staff's request for additional information (DOE, 2009bk), and the supporting DOE information on tephra redistribution presented in SNL (2007av).

Model Integration

Model abstractions comprise FEPs that have been screened in from the scenario analysis. In SER Section 2.2.1.2.1.3.1, the NRC staff determined that DOE identified a complete list of FEPs for the volcanic exposure scenario, including downward migration in soil. In SER Section 2.2.1.2.1.3.2, the NRC staff determined that DOE had acceptably screened all FEPs for the volcanic exposure scenario, including downward migration in soil. DOE did not exclude any FEPs associated with this abstracted model. The NRC staff's review of the abstracted model for downward migration in soil evaluates the applicant's implementation of the included FEPs: (i) FEP 1.2.04.07.0C, (ii) FEP 2.3.02.01.0A, (iii) FEP 2.3.02.02.0A, and (iv) FEP 2.3.02.03.0A.

As discussed in the next sections, the NRC staff finds that the DOE TSPA analysis adequately incorporates important FEPs and couplings between different models associated with the vertical migration of radionuclides in soil for the volcanic ash exposure scenario. SER Section 2.2.1.3.13.1.2 evaluates the treatment of climate in the DOE FAR model. The technical basis for this abstracted model was described in SNL (2007av); modeling assumptions were described in SNL (2007av, Section 5). DOE neglected effects due to future wetter climates in this abstracted model on the basis that wetter climates could increase radionuclide diffusivities in soil, increase vertical migration, reduce the radionuclide concentrations in surface soil layers, and thus reduce estimated doses. The NRC staff finds this DOE modeling approach acceptable because wetter climates would likely increase radionuclide migration to deeper soil layers, as identified in Till and Moore (1988aa, Eq. 2), and thus result in lower estimated doses. Therefore, the NRC staff concludes that neglecting the potential effects of a future wetter climate on radionuclide migration would not underestimate dose.

The NRC staff evaluated critical modeling assumptions for this abstracted model. The applicant assumed that all radionuclides migrate into the soil at the same rate because, in temperate climates, weathering of radionuclides from the soil surfaces into deeper soil layers is mainly a physical, rather than a chemical, process. This conclusion is supported by the similar rate of radionuclide migration for radionuclides of different chemical characteristics (Anspaugh, et al., 2002aa). The NRC staff considered the small reduction of radionuclide concentrations over time credited in the DOE model and finds the field data adequately support the reduction the applicant presented in SNL (2007av).

The DOE assumption for not explicitly modeling advection (i.e., flow by liquid movement) of radionuclides is acceptable to the NRC staff because including advective transport would likely increase the removal of radionuclides from the soil surface and thus reduce calculated doses. In addition, independent critical reviews from non-DOE, academic-based experts were conducted on an earlier version of the technical basis document, and the applicant included the review comments and responses in SNL (2007av, Appendix C, Section 7.3.2). The NRC staff reviewed those independent technical evaluations and finds that DOE sufficiently responded to and resolved comments by independent evaluators that pertained to the vertical migration of radionuclides in soil.

Data Sufficiency

The NRC staff evaluated data sufficiency in the DOE abstracted model for the downward migration of radionuclides in soil. This abstracted model consists of parameters for permeable depth in fluvial channels and on interchannel divides, soil diffusivity of radionuclides in channels and on interchannel divides, and land fraction of the Fortymile Wash alluvial fan attributed to channels. Parameter values for permeable depth in channels were inferred from USGS data on scour depth in channels and the abovementioned site-specific field data from soil pits, as described in SNL (2007av, Section 6.5.5.2). Diffusivity rates for radionuclide migration in soils on interchannel divides and in fluvial channels were determined from site-specific field data of Cs-137 profiles. These profiles resulted from contaminated fallout deposited approximately 50 years earlier from atmospheric nuclear weapons testing. The applicant performed soil-geomorphic mapping of the Fortymile Wash alluvial fan to determine the fraction of the fan area that has been subjected to fluvial erosion and deposition within the past 10,000 years, as identified in SNL (2007av, Appendix A). Diffusivity rates for fluvial channels were determined from measurements from surfaces the applicant characterized as active channels. Measured data from older terraces were used to calculate the diffusivity rate for interchannel divides. On the basis of these two field data sets, the applicant specified separate diffusivity parameters for older surfaces of the interchannel divides and younger surfaces in fluvial channels. Because the time-dependent reduction of radionuclide concentrations in soil is slow in the DOE TSPA, the NRC staff concludes that modeling of the downward migration of radionuclides tends to overestimate dose to the RMEI. The NRC staff finds the data sufficiently support this abstracted model in the TSPA because they are from site-specific field observations. On the basis of the information evaluated in this paragraph, NRC staff finds the parameter determinations are adequately described and justified. For these reasons, the NRC staff finds the DOE use of measured diffusivity rates for cesium in soil acceptable for modeling the reduction of radionuclide concentrations with time in the DOE TSPA for the volcanic ash exposure scenario.

Data Uncertainty

In the DOE abstracted model for the vertical migration of radionuclides in soil, parameter distributions are applied to account for data uncertainty. As discussed in the preceding paragraphs, the NRC staff reviewed these parameter distributions and concludes that the range of uncertainty in these parameters is consistent with the technical basis used to develop the parameters. A permeable depth in fluvial channels of 200 cm [79 in] was derived from field measurements and is used as a constant value, as outlined in SNL (2007av, Section 7.1.3 and Table 4.1-4). The applicant supported this constant value with an argument that the permeable depth in fluvial channels could be much deeper. The NRC staff notes that neglecting deeper permeable layers is conservative because increases in permeable depth reduce long-term radionuclide concentrations in soil. Given the minimal effect that vertical migration has on radionuclide concentrations in fluvial channels over time, the NRC staff finds that this treatment of uncertainty in permeable depth is acceptable.

Model Uncertainty

The NRC staff evaluated model uncertainty for the downward migration of radionuclides in soil following an eruption. For conditions after a potential future volcanic eruption that intersects the repository and entrains waste, radionuclides on the ground surface would originate as radionuclide contamination in basaltic tephra deposits, as discussed above. The applicant used site-specific data from the deposition and migration of fine particulates into current surface soils of the Fortymile Wash alluvial fan, which are not rich in basaltic material, to support its model for the downward migration of radionuclides following deposition in the volcanic ash exposure scenario (SAR Section 2.3.11.4.2.3; SNL, 2007av). The applicant provided a technical basis for neglecting the effects of fresh basaltic tephra on radionuclide diffusion in soil. The technical basis for radionuclide migration was provided in two parts: one for channel sediments and the other for soils on interchannel divides.

For channel sediments, the applicant used field observations at Lathrop Wells, described in SNL (2007av, Section 7.3.1.1), to show that basaltic and nonbasaltic sediments in the drainages exhibited similar grain sizes and transport rates. The applicant also found basaltic and nonbasaltic sediments to be well mixed. The applicant reported a significant amount of dilution of fresh basaltic tephra with nonbasaltic sediments during fluvial transport in the Fortymile Wash drainage basin. In particular, tephra concentrations in channel sediments were less than 20 percent at the RMEI location in the DOE TSPA. For these reasons, the applicant concluded that determining separate diffusion rates of radionuclides in basaltic tephra was not necessary for estimating the downward migration of radionuclides in mixed channel sediments. The NRC staff finds the DOE diffusion model adequately represents uncertainty for radionuclide migration in channel sediments for the volcanic ash exposure scenario because DOE's model is based on field observations.

For soils on interchannel divides, the applicant concluded in DOE (2009bk, Enclosure 3) that differences in diffusivity for a basaltic tephra deposit on ambient soils would be negligible because tephra thicknesses at the RMEI location would be thin {less than 0.33 cm [0.85 in] for about 90 percent of TSPA simulations with a primary tephra-fall deposit near the RMEI location} and grain-size ranges for tephra and ambient soils on interchannel divides are similar. The NRC staff notes that the DOE diffusion model does not permit radionuclides to migrate below the lower boundary of the surface soils, defined by parameters for the permeable depth. For very long times after the volcanic event, radionuclide concentration profiles in surface soil layers become uniformly distributed with depth in the DOE model. As previously discussed in SER Section 2.2.1.3.13.3.1.3, the NRC staff finds this assumption to be acceptable because neglecting deeper permeable depths would reduce the long-term radionuclide concentrations in soil. The NRC staff finds the applicant's parameter ranges and selected distributions acceptable because they represent expected conditions of the Yucca Mountain region for the volcanic ash exposure scenario.

DOE did not identify any alternative conceptual models that would likely affect the timing or magnitude of dose. The DOE model slowly distributes radionuclides from surface layers to deeper layers and restricts radionuclides from migrating below permeable soil layers. The NRC staff finds that the DOE diffusion model will not underestimate radiological dose. The NRC staff also finds that the approach DOE used is conservative and acceptable because there are no alternative models consistent with available information that would significantly increase radiological dose.

Model Support

The NRC staff evaluated model support for the downward migration of radionuclides in soil. DOE used a one-dimensional diffusion model for the downward migration of radionuclides in soil with measurements of Cs-137 radioactivity profiles in soils at Fortymile Wash. Pelletier, et al. (2005aa) published a peer-reviewed journal article that supported use of a diffusion model for radionuclide migration in soil at the Fortymile Wash alluvial fan. The article also included the synthesis of field data for determining radionuclide diffusivities in soil at the fan. In SNL (2007av, Section 7.2.6), the applicant verified the appropriateness of the DOE diffusion model for characterizing the radionuclide concentration profiles in soil. DOE supported its abstracted model with a geomorphic characterization of the Fortymile Wash alluvial fan, as identified in SNL (2007av, Appendix A), and derivation of its mathematical model for diffusion, described in SNL (2007av, Appendix E). For the aforementioned reasons, the NRC staff finds that DOE's diffusion model for radionuclide migration in soil has been adequately supported for its use in simulating exposure after an igneous eruption.

Summary of NRC Staff's Findings on the Downward Migration of Radionuclides in Soil Model

On the basis of the evaluation described above, the NRC staff finds that the DOE FAR model provides an acceptable approach for calculating the downward radionuclide migration in the soil at the location of the RMEI. DOE acceptably integrated its model of downward radionuclide migration in soil by adequately incorporating the important processes associated with the four included FEPs. The NRC staff finds that the parameter determinations and their uncertainties for the downward radionuclide migration are adequately described and justified. The NRC staff concludes that the data are sufficient to support this abstracted model in the TSPA because they are from site-specific field observations. The NRC staff finds that data uncertainty was adequately characterized and propagated through the abstracted model for downward radionuclide migration in soil. The NRC staff finds that DOE's treatment of alternative conceptual models is acceptable because no alternatives that are consistent with the presented data would affect the results significantly. Because DOE supported its abstracted model with a geomorphic characterization of the Fortymile Wash alluvial fan and derivation of its mathematical model for diffusion, the NRC staff finds the DOE model support acceptable.

Summary of NRC Staff's Findings on the Volcanic Ash Exposure Scenario

The NRC staff finds that DOE has provided sufficient information to address the acceptance criteria described in YMRP Sections 2.2.1.3.11.3 and 2.2.1.3.13.3 for the airborne transport of radionuclides, tephra redistribution in Fortymile Wash, and downward migration of radionuclides abstracted models, and also for their implementation in the GoldSim modeling environment of the TSPA analysis. The NRC staff finds that the license application considered appropriate data from the site and surrounding region, considered uncertainties and variability in parameter values, and used alternative conceptual models in the analyses. Specific FEPs have been included in the license application, and appropriate technical bases have been provided for inclusion or exclusion FEPs. The license application included specific degradation, deterioration, and alteration processes, and the effects of these processes were considered in evaluating annual dose. The NRC staff finds that the license application included adequate technical bases for models used in the performance assessment. The NRC staff finds that the performance assessment is sufficient for time periods after 10,000 years and through the period of geologic stability. Therefore, the NRC staff finds that the applicant has satisfied the requirements of 10 CFR 63.21(c)(1), (9) and (15), 63.114(a), 63.114(b), 63.305, and 63.342 for the volcanic ash exposure scenario.

2.2.1.3.13.3.2 Assessment and Review of Groundwater Exposure Scenarios

For the groundwater exposure scenario, the surface soil submodel addresses the vertical movement of radionuclides in the soil that follows from irrigation with contaminated groundwater (SAR Figure 2.3.10-1) and calculates a time-dependent profile of radionuclide concentration in the contaminated soil horizon at the RMEI location. Radionuclide contamination in groundwater can result from waste package failure due to corrosion, mechanical disruption, or potential disruption by intruding magma. Radionuclide contamination in groundwater serves as an input to the surface soil submodel. SER Section 2.2.1.3.12 documents the NRC staff's review of the DOE approach to estimating radionuclide contamination in groundwater. This section addresses the vertical movement of radionuclides in the soil from contaminated groundwater irrigation together with background precipitation. As described next, the applicant's results indicate the influence of the surface soil submodel on the DOE-calculated repository performance.

The NRC staff conducted a risk-informed review of the DOE surface soil submodel using YMRP Section 2.2.1.3.13. The NRC staff reviewed the important aspects of the groundwater exposure scenario in the DOE performance assessment.

Important Aspects of the DOE Surface Soil Submodel

The NRC staff reviewed the SAR and assessed the extent to which the DOE surface soil submodel influences the DOE calculation of repository performance. The surface soil submodel is used to estimate radionuclide doses for the groundwater exposure scenarios, including the seismic ground motion and igneous intrusion scenarios. SAR Figure 2.4-18(a and b) showed that the seismic ground motion and igneous intrusion scenarios dominate the estimated total mean annual dose for 10,000 and 1 million years after repository closure. Total doses from the groundwater exposure scenarios are controlled by multiple radionuclides and exposure pathways. Because the surface soil submodel is a component of the DOE biosphere model ERMYN (SAR Figure 2.3.10-9; BSC, 2006ah; SNL, 2007ac), its importance within the DOE TSPA depends on specific exposure pathways and radionuclides.

SAR Figures 2.4-20 and 2.4-30 indicated that a combined set of radionuclides—C-14, Tc-99, I-129, Ra-226, Np-237, Pu-238, Pu-240, and Pu-242—can contribute significantly to total dose over time. Per the NRC staff perspective on risk discussed near the beginning of this SER Section 2.2.1.3.13.3, the NRC staff used this DOE information and radionuclide set to assess the influence of the surface soil submodel on TSPA results. Pathways linked to the surface soil submodel (radon inhalation and external exposure) account for more than 80 percent of the Ra-226 biosphere dose conversion factor, as described in SNL (2007ac, Tables 6.13-1 and 6.13-2). Pathways linked to the surface soil submodel account for approximately 50 percent of the Tc-99, Pu-239, Pu-240, and Pu-242 biosphere dose conversion factors, as identified in SNL (2007ac, Tables 6.13-1 and 6.13-2). In SNL (2007ac, Tables 6.13-1 and 6.13-2), pathways linked to the surface soil submodel accounted for less than 35 percent of the Np-237, I-129, and C-14 biosphere dose conversion factors. On an individual radionuclide basis, the DOE surface soil submodel can have a large influence on estimated doses from Ra-226, a moderate influence on doses from Tc-99 and various plutonium isotopes, and a small influence on other radionuclide doses. Because no single, dominant radionuclide exists and Ra-226 contributes only a fraction to the total dose [SAR Figure 2.4-20(b)], the NRC staff concludes that these DOE results indicate the limited influence of the DOE surface submodel on the DOE-calculated repository performance.

Summary of DOE's Approach on the Surface Soil Submodel

The applicant's surface soil submodel is one component of the DOE biosphere model, which is described in SAR Section 2.3.10, Biosphere Transport and Exposure. In SAR Table 2.3.10-1, DOE identified the FEPs included in the TPSA model.

The surface soil submodel calculates the radionuclide concentrations in both cultivated field and garden surface soils following radionuclide release in the groundwater pathway. The output from the surface soil submodel serves as input for various biosphere submodels (animal, ingestion, external, plant, and air). The outputs of the biosphere model are biosphere dose conversion factors, which are factors that provide the dose-per-unit concentration in a medium such as water, for groundwater exposure (SAR Figure 2.3.10-9). Biosphere dose conversion factors are combined with the time-dependent radionuclide concentrations in groundwater from the saturated zone transport models to calculate annual dose to the RMEI from groundwater exposure (SAR Figure 2.3.10-9). The applicant's calculation used the groundwater exposure and volcanic ash exposure doses to demonstrate compliance with the individual protection standards at 10 CFR 63.311 and 10 CFR 63.321 (SAR Section 2.3.10.1). Results from the surface soil model are used to determine potential doses from inhalation of suspended soil particles, consumption of radionuclide-containing crops, soil ingestion by humans and animals, exposure to radioactive gases from the surface soil, and external exposure to radionuclide-containing soils. SER Section 2.2.1.3.14 documents the NRC staff evaluation of the biosphere dose conversion factors and biosphere submodels, other than the surface soil submodel; the NRC staff evaluation of the surface soil submodel follows.

In the surface soil submodel, radionuclides are considered to be added to the soil from irrigation using contaminated groundwater. They may decrease through the mechanisms of radioactive decay, leaching into deeper zones, erosion of soil particles, and gaseous release to the atmosphere (i.e., radon and carbon dioxide). Two soil layers are considered: a thin upper surface layer from which particles can be suspended into the atmosphere by disturbances and a thicker, lower surface layer that may undergo mixing by agricultural practices such as tilling the land.

NRC Staff's Review of the Surface Soil Submodel

The NRC staff reviewed SAR Sections 2.3.10 on the surface soil submodel and the supporting DOE information on the surface soil submodel presented in SNL (2007ac) and BSC (2006ah).

Model Integration

Model abstractions comprise FEPs that have been screened in from the scenario analysis. In SER Section 2.2.1.2.1.3.1, the NRC staff determined that DOE had identified a complete list of FEPs for the groundwater exposure scenario. DOE screened out the transport of radionuclides past these soil layers to greater depths in its analysis of FEPs. In SER Section 2.2.1.2.1.3.2, the NRC staff determined that DOE had acceptably screened all FEPs for the groundwater exposure scenario, including those associated with the surface soil submodel. The NRC staff's review of the downward migration modeling in soil evaluates the applicant's implementation of the included FEPs: (i) FEP 2.3.02.01.0A, (ii) FEP 2.3.02.02.0A, and (iii) FEP 2.3.02.03.0A.

DOE considered two soil layers in the surface soil submodel: (i) a thin surface layer that is susceptible to particles being suspended in the atmosphere from disturbances such as agricultural activities and wind and (ii) a lower, thicker layer that is approximately the thickness of the plow or till zone. Radionuclide concentrations for primary radionuclides and two long-lived decay products are calculated for varying climate conditions. Radionuclide concentrations are assumed to be uniform in the thin resuspendable layer and uniform in the thicker surface layer, if tilling is practiced.

The NRC staff evaluated the modeling assumptions and integration for the surface soil submodel. Radionuclides in the surface soil submodel originate from contaminated groundwater used for irrigation. The applicant used a unit concentration for each radionuclide of 1 Bq/m^3 in the irrigation water to determine normalized biosphere dose conversion factors. TSPA computes the radionuclide doses by multiplying these normalized factors by the radionuclide concentrations in the groundwater. DOE allowed the radionuclide concentration absorbed on soils to build up toward equilibrium conditions before estimating the potential dose to the RMEI. NRC staff notes that when equilibrium conditions are obtained, longer irrigation periods would not affect radionuclide concentrations in soil. So that the potential dose to the RMEI at earlier times would not be underestimated, DOE determined radionuclide concentrations in soil by assuming irrigation with contaminated well water for "prior irrigation" periods up to 1,000 years before estimating the potential dose to the RMEI. The NRC staff finds the applicant's prior irrigation period and buildup of radionuclides acceptable because the use of higher radionuclide concentrations in the surface soil would tend to overestimate dose.

The mathematical model DOE used, outlined in SNL (2007ac, Eq. 6.4.1-1, p. 6-73), to represent addition and removal of radionuclides in the surface soil is a differential equation that considers the dominant governing processes. The differential equation relates the rate of radionuclide accumulation to the difference between the rate of radionuclide addition and the rate of radionuclide loss in a volume of soil. This type of differential equation is known as an equation of continuity and is commonly used to track mass movement through systems (Bird, et al., 1960aa). Inherent in the equation is mass balance that accounts for the difference between radionuclide addition and removal per unit time. The differential equation accounts for changes in storage or radionuclide concentration in the soil's surface. Although the mathematical model and associated parameters DOE used do not account for all phenomena at a small (pore-level) scale, the NRC staff finds the overall behavior at larger scales, for which the parameters are justified, is appropriately represented because small scale phenomena are captured in the parameter determination. This mathematical model describes radionuclide movement at a scale where parameter values do not vary significantly with relatively small changes in spatial scales. For analyses evaluating potential doses to the receptor (RMEI), the NRC staff finds this modeling approach acceptable. For the reasons described above, NRC staff finds the conceptual and mathematical surface soil submodels are acceptable for determining average radionuclide concentrations in the surface soil resulting from irrigation with radionuclide-containing groundwater.

Radioactive decay, transport (i.e., leaching) to deeper soil, erosion of soil particles, and gaseous release to the atmosphere of radon and carbon dioxide are the dominant mechanisms for removal of radionuclides from the surface soil layers. Short-lived radioactive decay products (i.e., having half-lives shorter than 180 days) are assumed by DOE to be in equilibrium with the long-lived primary radionuclides. The NRC staff acknowledges this assumption as a common approach in environmental modeling and finds that it will not underestimate the effects of short-lived radionuclides on dose because any nonequilibrium in radionuclides having half-lives shorter than 180 days, produced by decay from long-lived parent radionuclides, will not affect the average annual dose to the RMEI. The potential removal of radionuclides that are incorporated into crops, which could then be harvested from the fields and gardens, is not considered by DOE. Radionuclides incorporated in these plants and animal wastes are assumed to be returned to the soil as fertilizer. Because crops grown in Amargosa Valley are assumed by DOE to be consumed by the RMEI or local livestock, the NRC staff finds that the modeling assumption to neglect radionuclide removal in crops and include radionuclide return to soil does not underestimate the dose. For the reasons described above, the NRC staff finds the applicant's assumptions for radionuclide return to soil and removal in the surface soil submodel to be acceptable.

Data Sufficiency and Uncertainty

The NRC staff evaluated data sufficiency and uncertainty for the irrigation rate source term. The irrigation rates were determined separately for field and garden crops. Irrigation rates directly affect radionuclide concentrations in the soil because more radionuclide mass is added to the soil when the irrigation rate is higher. An average irrigation rate was calculated from irrigation rates from several crops on the basis of current practices at Amargosa Valley, Nevada. Vegetables and fruit were assumed to be grown in gardens, whereas grains and forage were assumed to be grown in fields. An average irrigation rate was used to account for crop rotation in fields and gardens in Amargosa Valley. DOE accounted for crop overwatering to prevent the buildup of soluble salts in the rooting zone. Overwatering introduces more contaminated groundwater than is needed to grow the crops. The NRC staff views overwatering to be a standard and acceptable approach for determining the irrigation rate for crops, because limiting salt buildup in soils is desired and practiced worldwide (Hillel, 1971aa, p. 229). The NRC staff finds that the DOE assumptions for the irrigation source term in the surface soil submodel do not underestimate the potential dose. The NRC staff finds the DOE irrigation source term acceptable because it is consistent with present knowledge of the Yucca Mountain region and consistent with semiarid conditions.

The NRC staff evaluated data sufficiency and uncertainty for surface soil submodel parameters. DOE developed parameters for the surface soil submodel from surveys of land use in Amargosa Valley (e.g., type of crops grown, crop rotation, and crop acreage). The applicant applied documented physical and chemical properties (e.g., soil properties, radionuclide properties/characteristics). DOE generally field checked or verified the data obtained from the surveys against independent records. The data were also typically collected over several years to account for variability. Documented physical and chemical parameters were obtained from measurements and analyses by independent groups, such as the U.S. Department of Agriculture's soil surveys and established literature sources. DOE determined parameter values using relationships between parameters and measured quantities, which have been published in the scientific literature (e.g., Food and Agriculture Organization and the U.S. Department of Agriculture Natural Resources Conservation Service), and documented its analyses in the Biosphere Model Report (SNL, 2007ac). The applicant used parameter distributions to account for parameter uncertainty. The NRC staff reviewed the parameter distributions and concludes that these distributions are representative of the range of conditions in Amargosa Valley. Parameters were also adjusted by DOE to reflect differing climate conditions, where appropriate. For example, the average irrigation rate was adjusted to represent projected future climates. The NRC staff concludes that adjusting the surface soil submodel parameters to account for climate changes, on the basis of cautious and reasonable assumptions, is consistent with the regulatory requirements in 10 CFR 63.305. Because DOE addressed parameter uncertainty for differing climate conditions in a similar manner to the representation of the present-day climate (discussed above), the NRC staff finds that the resulting parameter distributions adequately address uncertainty.

Model Uncertainty and Support

The NRC staff evaluated the model support and applicant's treatment of model uncertainty for the surface soil submodel. In SNL (2007ac, Section 6.3.3), DOE concluded that there are no alternative conceptual models for the biosphere evaluation. For its review of the redistribution of radionuclides in soil, the NRC staff evaluated this conclusion in terms of the surface soil submodel. Because the DOE surface soil model relies on first principles of mass balance to represent radionuclide redistribution in soil, the NRC staff finds this conclusion to be reasonable. The NRC staff is not aware of an alternative approach to representing radionuclide redistribution in soil that uses a conceptual model that is significantly different from the first-principles approach used in the DOE surface soil model. The applicant compared ERMYN with two other established models that assess radionuclide concentrations in soil, GENII (Napier, et al., 2006aa) and BIOMASS ERB2A (International Atomic Energy Agency, 2003aa), to evaluate the technique used to solve the mathematical model. The mathematical development all these models used, including the surface soil submodel used in ERMYN, is similar after the terms are combined or redefined, as identified in SNL (2007ac, Section 7.3.1.1).

The applicant explained differences in the models and concluded that the calculations were equivalent. The NRC staff finds that the differences were adequately explained, and differences in these models are not expected to significantly affect performance assessment results. An external review conducted by independent experts [SNL (2007ac, Section 7.6)] provides additional confidence in the ERMYN model. This external review covered a broader scope, including surface soil components; it did not explicitly address the surface soil submodel. Nonetheless, the external review concluded in SNL (2007ac, Section 7.6) that the overall ERMYN model was a well-constructed, transparent, and complete biosphere modeling tool. On the basis of its review of the comparisons to established models, the NRC staff finds acceptable the applicant's assessment of model uncertainty and model support for the surface soil submodel.

Summary of NRC Staff's Findings on the Groundwater Exposure Scenario

On the basis of the previously described evaluation, the NRC staff finds that the surface soil submodel provides an acceptable approach for calculating the radionuclide concentrations in both cultivated field and garden surface soils via the groundwater pathway. DOE acceptably integrated its surface soil submodel by adequately incorporating the important processes associated with the three included FEPs. Furthermore, the applicant adequately and transparently described the governing processes of radionuclide buildup, retention, and removal in the surface soil. The NRC staff finds that the parameter determinations and their uncertainties for the surface soil submodel are adequately described and justified. The NRC staff concludes that the data are sufficient to support the abstracted model for radionuclide transport in the soil because they are based on documented soil properties of the Yucca Mountain region. The NRC staff finds that data uncertainty was adequately characterized and propagated through the abstracted model by stochastic sampling of parameter ranges. The NRC staff finds that DOE adequately addressed uncertainty in the conceptual model through comparisons to soil submodels in two other established biosphere models. Because DOE supported the surface soil model results by comparison to results from other published biosphere models, the NRC staff finds DOE's model support acceptable.

The NRC staff finds that DOE has provided sufficient information to address the acceptance criteria described in YMRP Section 2.2.1.3.13.3 for the surface soil submodel and its implementation in the GoldSim modeling environment of the TSPA analysis. The NRC staff has reviewed the information in the SAR and other information submitted in support of the license application and has found that the requirements of 10 CFR 63.21(c) (1), (9), and (15), 63.114(a) and 10 CFR 63.114(b) are satisfied. Regarding the requirements of 10 CFR 63.305, with respect to characteristics of the reference biosphere used in the groundwater exposure scenario, the NRC staff has reviewed the information in the SAR and other information submitted in support of the license application and has found that the requirements are satisfied.

2.2.1.3.13.4 Evaluation Findings

The NRC staff has reviewed the SAR and other information submitted in support of the license application, which includes information required by 10 CFR 63.21(c)(1),(9), and (15), and finds, with reasonable expectation, that the relevant requirements of 10 CFR 63.114, 63.305, and 63.342 are satisfied regarding the abstraction of airborne transport and redistribution of radionuclides. In particular, the NRC staff finds that DOE has adequately

- Included appropriate data related to the geology, hydrology, and geochemistry of the surface and subsurface from the site and the region surrounding Yucca Mountain to define parameters and conceptual models used in the performance assessment calculation, in compliance with 10 CFR 63.114(a)(1)

- Accounted for uncertainty and variability in the parameter values used to model airborne transport and redistribution of radionuclides, in compliance with 10 CFR 63.114(a)(2)

- Considered and evaluated alternative conceptual models that are consistent with currently available data and scientific understanding, in compliance with 10 CFR 63.114(a)(3)

- Provided a technical basis for the inclusion of FEPs affecting airborne transport and redistribution of radionuclides, consistent with the limits on performance assessment in 10 CFR 63.342, and evaluated in sufficient detail those processes that would significantly affect repository performance, in compliance with 10 CFR 63.114(a)(4-6)

- Provided technical bases for the models of airborne transport and redistribution of radionuclides used in the performance assessment to represent the 10,000 years after disposal, in compliance with 10 CFR 63.114(a)(7)

- Used performance assessment methods for the period of geologic stability consistent with the methods used to demonstrate compliance for the initial 10,000 years following permanent closure, in compliance with 10 CFR 63.114(b)

- Included in the performance assessment for the period of geologic stability those FEPs used for the initial 10,000-year period, consistent with specific limits, in compliance with 10 CFR 63.342(c)

The NRC staff finds that, with respect to the requirements of 10 CFR 63.305, for consideration of the redistribution of radionuclides, DOE has adequately

- Used FEPs to describe the reference biosphere that are consistent with present knowledge of conditions in the region surrounding the Yucca Mountain site, in compliance with 10 CFR 63.305(a)

- Not projected changes in society, the biosphere (other than climate), or human biology or increases or decreases of human knowledge and technology, and assumed that all of those factors are constant as they are at the time of submission of the license application, in compliance with 10 CFR 63.305(b)

- Varied factors related to the geology, hydrology, and climate based upon cautious but reasonable assumptions of the changes in these factors that could affect the Yucca Mountain disposal system during the period of geologic stability, consistent with the requirements for performance assessments specified at 10 CFR 63.342, in compliance with 10 CFR 63.305(c)

- Used biosphere pathways consistent with arid or semi-arid conditions, in compliance with 10 CFR 63.305(d).

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CHAPTER 16

2.2.1.3.14 Biosphere Characteristics

2.2.1.3.14.1 Introduction

Safety Evaluation Report (SER) Section 2.2.1.3.14 evaluates the U.S. Department of Energy's ("DOE" or "applicant") postclosure performance assessment model used to calculate biosphere transport and the annual dose to the reasonably maximally exposed individual (RMEI), as presented in DOE's Safety Analysis Report (SAR) (DOE, 2008ab). The sources of radionuclides used in the applicant's biosphere model calculations are calculated by other models in its performance assessment analysis. Those models calculate repository releases from postclosure engineered-barrier-system failures and then model the transport of the released radionuclides from the repository location to the biosphere. Results from these other transport models provide the sources of radionuclides from two primary biosphere media: groundwater and soil contaminated with tephra deposits. In the model, tephra (hereafter, volcanic ash) is deposited on the ground from postulated volcanic events. The applicant's biosphere model then calculates the subsequent transport of these radionuclides within the biosphere through a variety of exposure pathways (e.g., soil, food, water, air) and applies dosimetry modeling to convert the RMEI exposures into annual dose.

In 10 CFR 63.2, the reference biosphere is defined as "the description of the environment inhabited by the [reasonably maximally exposed individual]." The RMEI is defined by regulation (10 CFR 63.312) as a hypothetical adult who (i) lives in the accessible environment above the highest concentration of radionuclides in the plume of contamination; (ii) has a diet and living style representative of current Amargosa Valley, Nevada, residents; (iii) uses well water with average concentrations of radionuclides based on an annual water demand of 3,000 acre-ft [3.7×10^9 L]; (iv) drinks 2 L [0.528 gal] of water per day from groundwater extracted from wells drilled at the location specified in (i) of this paragraph; and (v) is an adult with metabolic and physiological considerations consistent with present knowledge of adults.

DOE estimated the dose to the RMEI on the basis of the concentrations of radionuclides in groundwater and in contaminated ash. These concentrations were calculated by DOE's model abstractions for saturated zone transport (SAR Revision 1, Section 2.3.9), extrusive (volcanic eruption) atmospheric dispersal (SAR Section 2.3.11.4.5.2), and volcanic ash redistribution (SAR Section 2.3.11.4.5.3). These model abstractions, which provide inputs for the biosphere calculations within the Total System Performance Assessment (TSPA) model, are reviewed in SER Sections 2.2.1.3.9 and 2.2.1.3.13, respectively. This SER Section focuses on the U.S. Nuclear Regulatory Commission (NRC) staff's (staff) review of the performance assessment calculations of biosphere transport and dose to the RMEI described in SAR Section 2.3.10. The NRC staff's evaluation of biosphere modeling of radionuclide concentrations in soil can be found in SER Section 2.2.1.3.13.

In SAR Section 2.3.10, DOE analyzed the characteristics of the Yucca Mountain region and Amargosa Valley, Nevada, for its biosphere transport and RMEI dose calculations. The applicant identified features, events, and processes (FEPs) and developed biosphere conceptual and mathematical models for use in its TSPA computer model. The applicant described environmental conditions, resident lifestyle, exposure media, environmental transport pathways, and human exposure pathways it used for evaluating the impacts of repository performance on dose to the RMEI.

Exposure pathways in the DOE biosphere model are based on assumptions about residential and agricultural uses of the water and indoor and outdoor activities. These pathways include ingestion, inhalation, and direct exposure to radionuclides deposited to soil from irrigation (SAR Section 2.3.10.1). Ingestion pathways include drinking contaminated water, eating crops irrigated with contaminated water, eating food products produced from livestock raised on contaminated feed and water, eating farmed fish raised in contaminated water, and inadvertently ingesting soil. Inhalation pathways include breathing resuspended soil, aerosols from evaporative coolers, and radon gas and its decay products.

DOE's approach to biosphere modeling was twofold. The applicant used a standalone computer code entitled Environmental Radiation Model for Yucca Mountain Nevada (ERMYN) to calculate biosphere dose conversion factors, which were used as inputs in DOE's TSPA code. The TSPA multiplied the appropriate biosphere dose conversion factor by either a soil concentration or a water concentration to obtain the dose to the RMEI for each exposure scenario (i.e., volcanic ash, groundwater). The substance of DOE's biosphere modeling approach is contained primarily in the ERMYN code.

This SER Section evaluates the technical bases for the applicant's conceptual and mathematical biosphere models, input parameter selections, parameter uncertainty propagation, model support, and model implementation and integration within the applicant's performance assessment evaluation. These evaluations are organized by subsections that address specific components of the applicant's biosphere model (or model development process), including system description and model integration, biosphere transport pathways, human exposure, dosimetry, and integrated biosphere modeling results. The NRC staff's review evaluates both the biosphere modeling in the ERMYN code and how the applicant used the ERMYN output (the biosphere dose conversion factors) to calculate RMEI dose in the TSPA model.

2.2.1.3.14.2 Regulatory Requirements

The applicant is required to provide information specified in 10 CFR 63.21(c)(1) that is relevant to the abstraction of the biosphere characteristics modeling. The requirements in 10 CFR 63.114 (Requirements for Performance Assessments) and 10 CFR 63.342 (Limits on Performance Assessments) include postclosure performance assessment to demonstrate compliance with 10 CFR 63.113 (Performance Objectives for the Geologic Repository after Permanent Closure). Compliance with 10 CFR 63.113 is reviewed in SER Section 2.2.1.4.1.

The regulations for performance assessment in 10 CFR 63.114 require, in part, that a performance assessment

- Include data related to the geology, hydrology, and geochemistry (including disruptive processes and events) of the surface and subsurface from the site and the region surrounding Yucca Mountain [10 CFR 63.114(a)(1)]

- Account for uncertainty and variability in the parameter values [10 CFR 63.114(a)(2)]

- Consider and evaluate alternative conceptual models [10 CFR 63.114(a)(3)]

- Provide technical bases for either the inclusion or exclusion of FEPs, including effects of degradation, deterioration, or alteration processes of engineered barriers that would adversely affect performance of the natural barriers, consistent with the limits on performance assessment in 10 CFR 63.342, and evaluate in sufficient detail those processes that would significantly affect repository performance [10 CFR 63.114(a)(4-6)]

- Provide technical basis for the models used in the performance assessment to represent the 10,000 years after disposal [10 CFR 63.114(a)(7)]

The NRC staff's evaluation of inclusion or exclusion of FEPs is given in SER Section 2.2.1.2.1. Requirements for performance assessment for the initial 10,000 years following disposal are specified in 10 CFR 63.114(a). Requirements for the performance assessment methods for the time from 10,000 years through the period of geologic stability, defined in 10 CFR 63.302 as 1 million years following disposal, are specified in 10 CFR 63.114(b) and 10 CFR 63.342. These sections provide that through the period of geologic stability, with specific limitations, the applicant

- Use performance assessment methods consistent with the performance assessment methods used to demonstrate compliance for the initial 10,000 years following permanent closure [10 CFR 63.114(b)]

- Include in the performance assessment those FEPs used in the performance assessment for the initial 10,000-year period (10 CFR 63.342)

The applicant's model abstraction for biosphere characteristics is subject to the specific constraints given in

- 10 CFR 63.102(o), specifying the implementation of total effective dose equivalent (TEDE)

- 10 CFR 63.305, specifying the required characteristics of the reference biosphere

- 10 CFR 63.311(b), requiring inclusion of all potential pathways of radionuclide transport and exposure

- 10 CFR 63.312, specifying the required characteristics of the RMEI

The NRC staff's review of the SAR and supporting information follows the guidance in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa), Sections 2.2.1.3.14, Biosphere Characteristics, as supplemented by additional guidance for the period beyond 10,000 years after permanent closure (NRC, 2009ab). The YMRP acceptance criteria for model abstractions that provide guidance for the NRC staff's review of the applicant's abstraction of biosphere characteristics are

1. System description and model integration are adequate
2. Data are sufficient for model justification
3. Data uncertainty is characterized and propagated through the abstraction
4. Model uncertainty is characterized and propagated through the abstraction
5. Model abstraction output is supported by objective comparisons

In its review of the SAR and supporting information, the NRC staff used a risk-informed approach for aspects of biosphere characteristics important to repository performance. The

NRC staff considered all five acceptance criteria provided in the YMRP in its review of information provided by DOE. In the context of these criteria, only those aspects of the model abstraction that significantly affect the performance assessment results, as determined by the NRC staff, are discussed in detail in this SER section. The NRC staff's determination is based both on risk information provided by DOE and on NRC independent analyses and staff knowledge gained through experience.

2.2.1.3.14.3 Technical Review

The NRC staff's technical review of DOE's biosphere characteristics model abstraction evaluated both the biosphere model and the model development process. Considering the YMRP acceptance criteria and the technical organization of the DOE's biosphere characteristics model abstraction, the review focused on five topics: (i) system description and model integration, (ii) biosphere transport pathways, (iii) human exposure, (iv) dosimetry, and (v) the integrated biosphere modeling results. These reviews are documented in subsections of this SER Section. The system description and model integration review evaluated the applicant's overall conceptualization of the biosphere, including FEPs that were selected and included in the applicant's biosphere conceptual models.

The NRC staff's detailed review focused on the most risk-significant parts of the applicant's biosphere model. Risk insights that apply to both the applicant's TSPA results and to the applicant's detailed abstraction modeling of the biosphere (i.e., using the ERMYN code to generate the biosphere dose conversion factors) informed the NRC staff's review. These risk insights focused the NRC staff's detailed review on those aspects of the applicant's biosphere modeling that contributed most to the calculated RMEI dose results in the TSPA.

The NRC staff analyzed the risk-significant aspects of the biosphere model abstraction in the TSPA code by evaluating the applicant's sensitivity analysis results using the TSPA code. These results indicated that biosphere dose conversion factors (ERMYN code outputs) significantly affected the TSPA results, as identified in SNL (2008ag, Appendix K, pp. FK-63 to FK-65). The NRC staff, therefore, performed a detailed technical review of the applicant's biosphere model.

The NRC staff's risk-informed review evaluated a subset of the applicant's biosphere model abstraction to determine the acceptability of the applicant's overall methodology. This detailed review focused on the subset of radionuclides and exposure pathways that were the most risk significant in the applicant's performance assessment analysis. These radionuclides and exposure pathways are summarized in SER Tables 16-1 and 16-2.

On the basis of the applicant's documentation of its performance assessment results, the NRC staff developed SER Tables 16-1 and 16-2 using the following two-step approach. First, the NRC staff identified those radionuclides that individually account for the largest fraction of the applicant's peak total mean annual dose results from its TSPA analysis, as shown in SAR Figure 2.4-20. Next, the primary pathways that account for the largest fraction of the applicant's calculated biosphere dose conversion factors for each identified radionuclide were included. The applicant's pathway contributions to each radionuclide-specific biosphere dose conversion factor were documented in SAR Table 2.3.10-11. On the basis of this analysis of the applicant's results, the NRC staff concludes that the radionuclides and pathways identified in SER Tables 16-1 and 16-2 are the most risk-significant contributors to the applicant's TSPA results.

Table 16-1. Exposure Pathways and Radionuclides Found To Be the Most Risk Significant in the DOE Performance Assessment for the 10,000-Year Simulation Period

Radionuclide*	Source of Radionuclides†	Route of Exposure‡	Primary Pathways‡
Tc-99	Estimated Releases to Groundwater	Ingestion	42% drinking water
			37% animal product
C-14		Ingestion	59% fish
			22% drinking water
Pu-239		Inhalation	50% particulates
			24% evaporative cooler aerosols
		Ingestion	19% drinking water
I-129		Ingestion	60% drinking water
			28% animal products

*Radionuclides presented in order of their contribution to the DOE peak total mean annual dose results in SAR Figure 2.4-20.
†Modeling cases that contribute most to the DOE total mean annual dose are based on release to groundwater as shown in SAR Figure 2.4-18 and SAR Section 2.4.2.2.1.2.
‡Routes of exposure and primary pathways from SAR Table 2.3.10-11. Various pathways not listed contribute the remaining percentage of each radionuclide dose.

Table 16-2. Exposure Pathways and Radionuclides Found To Be the Most Risk Significant in the DOE Performance Assessment for the 1-Million-Year Simulation Period

Radionuclide*	Source of Radionuclides†	Route of Exposure‡	Primary Pathways‡
Pu-242	Estimated Releases to Groundwater	Inhalation	51% particulates
			24% evaporative cooler aerosols
		Ingestion	19% drinking water
Np-237		Inhalation	35% evaporative cooler aerosols
			21% particulates
		Ingestion	29% drinking water
Ra-226		Inhalation	74% radon
I-129		Ingestion	60% drinking water
			28% animal products

*Radionuclides presented in order of their contribution to the DOE peak total mean annual dose results in SAR Figure 2.4-20.
†Modeling cases that contribute most to the DOE total mean annual dose are based on release to groundwater as shown in SAR Figure 2.4-18 and SAR Section 2.4.2.2.1.2.
‡Routes of exposure and primary pathways from SAR Table 2.3.10-11. Various pathways not listed contribute the remaining percentage of each radionuclide dose.

The NRC staff focused its detailed review on the subset of radionuclides and pathways that are the most risk-significant contributors to the applicant's performance assessment results. An example is provided here to clarify how this approach identifies the most risk-significant contributors to the applicant's results. For the applicant's 1-million-year results presented in SAR Figure 2.4-20(b), the peak total mean annual dose is approximately 0.02 mSv/yr [2 mrem/yr]. Four radionuclides contribute approximately 0.015 mSv/yr [1.5 mrem/yr] (75 percent) to that value. The NRC staff identified these four radionuclides as the most risk-significant contributors because they represent the smallest number of radionuclides that comprise the largest fraction of the peak mean dose. The remaining 17 radionuclides in the applicant's analysis each contributed a small fraction to the peak mean dose. The pathways for these four radionuclides were then individually evaluated to identify the subset of pathways that contributed the largest fraction to the dose contribution from the radionuclide using information provided in SAR Table 2.3.10-11. For example, Pu-242 is responsible for 30 percent of the applicant's peak mean dose. The NRC staff evaluated the pathways through which Pu-242 contributed to the dose and found that the inhalation of particulates pathway was responsible for 51 percent of the Pu-242 dose, the inhalation of evaporative cooler aerosols pathway was responsible for 24 percent of the Pu-242 dose, and the drinking groundwater pathway was responsible for 19 percent of the Pu-242 dose. Twelve other pathways are responsible for the remaining 6 percent of the Pu-242 dose; therefore, three pathways were identified as being the most risk significant for Pu-242. This example illustrates the NRC staff's approach to identifying the radionuclides and their pathways that are the most risk significant to the applicant's performance assessment calculation.

SER Table 16-1 contains the radionuclides and their pathways that are the most risk-significant contributors to the applicant's performance assessment dose results for the time period of 10,000 years following disposal, as specified by 10 CFR 63.311(a)(1). SER Table 16-2 contains the radionuclides and their pathways that are the most risk-significant contributors to the applicant's performance assessment dose results for the time period after 10,000 years following disposal but within the period of geologic stability, as described in 10 CFR 63.311(a)(2). The radionuclides listed in SER Tables 16-1 and 16-2 include radionuclides found to be important contributors to dose results in prior independent NRC performance assessment results, as identified in NRC (2005aa, Volume 2, Appendix D).

While the NRC staff's technical review evaluated all of the applicant's biosphere modeling documentation, the NRC staff's review focused on the applicant's biosphere submodels and input parameters that are risk significant in the biosphere dose conversion factor calculations. The NRC staff's identification of risk-significant pathways is in SER Tables 16-1 and 16-2. In particular, the NRC staff's review of the data supporting the biosphere transport pathway input parameters (SER Section 2.2.1.3.14.3.2) focused on parameters in the applicant's transport submodels. These transport submodels include plant uptake, animal uptake, fish uptake, and air modeling. Similarly, the NRC staff's review of the applicant's data supporting input parameters for the human exposure submodels, evaluated in SER Section 2.2.1.3.14.3.3, focused on the inhalation and ingestion exposure submodels because those are the routes of exposure that are most risk significant in SER Tables 16-1 and 16-2.

2.2.1.3.14.3.1 System Description and Model Integration

In SAR Section 2.3.10.2 and in supporting references, the applicant described the biosphere characteristics of the Yucca Mountain region; of Amargosa Valley, Nevada, that impact its residents; of included FEPs; and of the biosphere conceptual models in the ERMYN code that were used to calculate biosphere dose conversion factors. This section documents the NRC staff's review of these descriptions. As discussed next, this review evaluated whether the applicant's included FEPs and conceptual models satisfy applicable regulatory requirements found at 10 CFR 63.114(a)(5); 10 CFR 63.305; 10 CFR 63.311(b); and 10 CFR 63.312(a), (b), (d), and (e). An additional part of the NRC staff's review evaluated integration (i.e., couplings, consistency, and assumptions) of the TSPA biosphere model abstraction with other TSPA model abstractions.

Features, Events, and Processes

The applicant described the Yucca Mountain region characteristics in SAR Section 2.3.10.2.1. This information addressed topics including climate, topography and soils, native flora and fauna (i.e., plants and animals), local communities, infrastructure (including water sources), and agricultural conditions. Information on the characteristics of Amargosa Valley, Nevada, residents (summarized in SAR Section 2.3.10.2.2) originated predominantly from local and national surveys. The SAR addressed topics such as diet and lifestyle factors, including the use of evaporative coolers, gardening, employment, commuting, housing, and metabolic considerations. The applicant documented the screening approach for the FEPs in SAR Section 2.2.1.2 and listed all the FEPs that were evaluated for the TSPA model in SAR Table 2.2-1. FEPs that were included in the biosphere model are listed in SAR Table 2.3.10-1 and are reviewed in this SER section. The NRC staff's review of excluded FEPs is documented in SER Section 2.2.1.2.1.3.2.

The NRC staff's review evaluated the technical bases the applicant used to support its disposition of included FEPs in the performance assessment with respect to the following: (i) whether the applicant provided satisfactory technical bases for including biosphere FEPs in compliance with 10 CFR 63.114(a)(5); (ii) whether the included FEPs are consistent with present knowledge of the conditions in the region surrounding the Yucca Mountain site and in compliance with 10 CFR 63.305(a); and (iii) whether the applicant included all biosphere-related FEPs that could significantly change the magnitude or timing of the radiological exposures to the RMEI in its performance assessment, in compliance with 10 CFR 63.114(a)(5).

The NRC staff evaluated the applicant's technical bases for included FEPs and reviewed the applicant's descriptions of how each included biosphere FEP was incorporated into the performance assessment calculation. In this review, the NRC staff verified that the FEPs that could significantly contribute to the RMEI dose were included in the performance assessment calculation and that information supporting the FEPs was based on present knowledge of the Yucca Mountain region conditions. Because many FEPs are general in nature (e.g., climate change, biosphere characteristics, plant uptake), the NRC staff evaluated whether the performance assessment evaluation included the specific aspects of a feature, event or process that would be expected to contribute to RMEI dose, and therefore should be included in the model. As part of its review of the applicant's analyses, the NRC staff also incorporated its understanding of Yucca Mountain region characteristics obtained from extensive prelicensing experience and independent analyses of the biosphere characteristics at Yucca Mountain (e.g., LaPlante and Poor, 1997aa).

Based on the preceding analysis, the NRC staff finds that the applicant's list of included FEPs is consistent with the NRC staff's independent assessment of the characteristics of the Yucca Mountain region and Amargosa Valley. The NRC staff finds that features included in the applicant's performance assessment (i.e., wells, soil type, agricultural land use, irrigation, animal farms, fisheries, and human lifestyle and characteristics) were appropriately supported by the technical bases in the SAR and supporting references and are representative of the present knowledge of the Yucca Mountain region. The NRC staff finds that processes included in the applicant's performance assessment (i.e., radionuclide accumulation in soils, atmospheric transport, plant and animal uptake, radioactive decay, ingestion, inhalation, external exposure, and radiation dose) were appropriately supported by the technical bases in the SAR and supporting references that represented present knowledge of the Yucca Mountain region. Therefore, the NRC staff finds that the applicant's list of included FEPs is complete and acceptable for use in its biosphere model. In addition, the NRC staff has not identified any additional FEPs that are excluded from the applicant's biosphere model that would be expected to significantly increase the dose or affect the timing of dose to the RMEI. On the basis of these considerations, the NRC staff concludes that the applicant's inclusion of biosphere FEPs complies with the requirements of 10 CFR 63.114(a)(5) and 10 CFR 63.305(a).

Conceptual Models

In SAR Section 2.3.10.2.3, the applicant considered FEPs and NRC regulatory requirements for the reference biosphere model in identifying applicable RMEI exposure pathways and developing RMEI exposure scenarios. An exposure scenario, in general, describes a set of facts, assumptions, and inferences about how exposure occurs. In the applicant's Yucca Mountain biosphere model, an exposure scenario is a conceptual model that describes the biosphere characteristics, which lead to the RMEI's exposure to radionuclides that enter the biosphere from different transport routes (here, groundwater or volcanic ash). DOE's conceptual representations of the exposure pathways for groundwater and volcanic ash exposure scenarios were provided in SAR Figures 2.3.10-6 and 2.3.10-8. DOE incorporated these conceptual representations into mathematical submodels in the ERMYN code. The mathematical submodels in the ERMYN code were depicted in SAR Figures 2.3.10-9 and 2.3.10-10 and described in SAR Sections 2.3.10.2.5 and 2.3.10.2.6.

The NRC staff evaluated the applicant's conceptual models and associated mathematical submodels in the ERMYN code for consistency with the applicable NRC regulations for the biosphere model and the performance assessment analysis: 10 CFR 63.305(b) and (d); 10 CFR 63.311(b); and 10 CFR 63.312(a), (b), (d), and (e). The NRC staff reviewed both the applicant's groundwater exposure scenario (the modeling of biosphere characteristics that lead to the RMEI exposure to radionuclides from contaminated groundwater) and the volcanic ash exposure scenario. These reviews, documented in the subsections that follow, evaluated whether DOE's conceptual representations of the biosphere model included all potential pathways of radionuclide transport and exposure to comply with 10 CFR 63.311(b). These reviews also evaluated whether the applicant has complied with requirements for the description of the biosphere and RMEI in 10 CFR 63.305 and 10 CFR 63.312, respectively, that apply to this conceptual-model-level of analysis. Input parameters and data for the reference biosphere and RMEI are discussed in SER Sections 2.2.1.3.14.3.2, 2.2.1.3.14.3.3, and 2.2.1.3.14.3.4.

Groundwater Exposure Scenario Conceptual Model

The NRC staff's review of the applicant's conceptual model of the groundwater exposure scenario included biosphere FEPs, their functional relationships, and the resulting exposure pathways for modeling biosphere transport and dose to the RMEI. The NRC staff's evaluation of functional relationships among FEPs considered how the applicant accounted for interactions among related FEPs in the biosphere conceptual model so that all potential pathways could be identified. For example, farming practices, such as soil irrigation, can lead to soil accumulation of radionuclides, which can contribute to plant uptake of radionuclides from that soil.

The applicant's groundwater exposure scenario includes a RMEI who is assumed to be an adult who lives in the accessible environment above the highest concentration of radionuclides in the plume of contamination, as required by 10 CFR 63.312(a) and (e). The RMEI is assumed to use wells to draw groundwater from the contamination plume for domestic and agricultural purposes. The NRC staff finds acceptable the general RMEI and biosphere characteristics in the applicant's model that addresses the use of groundwater for drinking, irrigating crops, watering livestock, raising fish, and operating evaporative coolers.

The applicant's conceptualization of dose to the RMEI involves three routes of exposure: external exposure, inhalation, and ingestion. The inhalation dose portion of the applicant's conceptual model includes RMEI inhalation of radionuclides in (i) resuspended soil particles, (ii) gaseous emissions from the soil and their radioactive decay products, and (iii) aerosols generated by evaporative coolers. The ingestion dose portion of the applicant's conceptual model includes (i) drinking water; (ii) crops, including leafy vegetables, other vegetables, fruits, and grains; (iii) animal products, including meat, poultry, milk, and eggs; (iv) freshwater fish; and (v) soil. The meat category is a combination of all edible portions of beef, pork, and wild game (BSC, 2005ab).

Based upon this information, on NRC staff's review of the biosphere characteristics in the applicant's groundwater exposure scenario conceptual model (described in SAR Section 2.3.10.2), and on the associated mathematical models described in SAR Section 2.3.10.3, the NRC staff finds that these models include all potential pathways of radionuclide transport and exposure in compliance with 10 CFR 63.311(b). This finding is based on the applicant's inclusion of pathways that (i) the NRC staff finds characteristic of the Yucca Mountain region, (ii) are commonly included in dose models and assessments, and (iii) are based on unique site-specific considerations (e.g., evaporative coolers, fish farming, wild game, and radon).

The NRC staff finds that the applicant's groundwater conceptual model and associated mathematical models also comply with the biosphere and RMEI requirements in 10 CFR 63.305(b) because the NRC staff finds that DOE has not projected changes in society, the biosphere (other than climate), human biology, or increases or decreases of human knowledge or technology. These factors in the DOE models remain constant throughout the evaluation period with characteristics that existed when the license application was submitted.

The NRC staff finds that DOE has complied with 10 CFR 63.305(c) by allowing factors related to the geology, hydrology, and climate to vary based upon cautious, but reasonable, assumptions of the changes in these factors that could affect the Yucca Mountain disposal system during the period of geologic stability. This is consistent with the performance assessment requirements specified in 10 CFR 63.342. Examples of variation in geology, hydrology, and climate include varying soil characteristics, precipitation and irrigation, and analysis of different climate states, as further evaluated below (see Integration of Biosphere Model in the TSPA). The NRC staff also finds that the biosphere pathways in the applicant's groundwater conceptual model are consistent with arid or semiarid conditions by including the use of groundwater to meet domestic and agricultural water needs, frequent irrigation with overwatering, resuspension of dry soils by winds, and water evaporation to cool residences. This complies with the requirements of 10 CFR 63.305(d).

Volcanic Ash Exposure Scenario Conceptual Model

The NRC staff's review of the volcanic ash exposure scenario conceptual model evaluated the included biosphere system FEPs, their functional relationships, and the included exposure pathways for modeling biosphere transport and dose to the RMEI. The NRC staff finds that the applicant's volcanic ash exposure scenario includes RMEI exposure to radioactive waste entrained in (i) volcanic ash that is deposited from the initial plume of released radionuclides directly to the ground at the location of the RMEI and (ii) regional volcanic ash deposits that are redistributed via eolian (carried by wind) and fluvial (carried by water) processes. Other models in the applicant's performance assessment address the transport of contaminated volcanic ash to the ground in the biosphere. Those models are reviewed in SER Section 2.2.1.3.13.

Because the applicant calculates the RMEI dose from volcanic ash in the soil on the basis of the potential routes of exposure to the RMEI that are similar to those already reviewed for the groundwater scenario (including external, inhalation, and ingestion exposures), this review emphasizes the aspects of the volcanic exposure scenario conceptual model that are not duplicated in the groundwater exposure scenario conceptual model. The applicant's use of common submodels with the groundwater scenario is acceptable because the methods for calculating dose to the RMEI from radionuclide-contaminated soil are the same whether the radionuclide concentration of the soil results from irrigation or volcanic ash deposition.

The applicant's conceptualization of inhalation dose in the volcanic ash exposure scenario includes resuspension of radionuclides in soil particles and release of Rn-222 (radon) gas. The NRC staff finds that inclusion of resuspension modeling is consistent with the applicant's description of an arid climate characterized by low rainfall (SAR Sections 2.3.10.2.1 and 2.3.1) and is, thus, in compliance with 10 CFR 63.305(d). The applicant's inclusion of resuspension in its conceptual approach also addressed a variety of dust-generating activities and RMEI exposure environments (SAR Sections 2.3.10.2.6 and 2.3.10.3.2.2). Therefore, the NRC finds that the applicant has adequately addressed the potential dust inhalation exposure pathways. The applicant's ingestion dose calculations applied the same soil-contamination-based exposure pathways that were used in the applicant's scenario for groundwater exposure. The applicant considered that groundwater pathways, which did not include a soil component (e.g., evaporative coolers, ingestion of groundwater, and ingestion of fish), did not apply to the volcanic ash exposure scenario, based upon the applicant's exclusion of FEP 1.2.04.07.0B for ash in groundwater (SNL, 2008ab). The NRC staff's review of the applicant's exclusion of this FEP is documented in SER Section 2.2.1.2.1.3.2, which found the exclusion to have an adequate technical basis.

On the basis of characteristics of the applicant's volcanic ash exposure scenario conceptual model, the NRC staff concludes that the conceptual model includes all potential pathways of radionuclide transport and exposure in compliance with 10 CFR 63.311(b). This conclusion is based on the applicant's inclusion of pathways that the NRC staff finds are (i) characteristic of the Yucca Mountain region, (ii) commonly included in dose models and assessments, and (iii) based on unique site-specific considerations (e.g., various particulate resuspension exposure environments and radon). The NRC staff finds that the applicant's conceptual model also complies with the biosphere and RMEI requirements of 10 CFR 63.305(b) because the applicant has not projected changes in society, the biosphere (other than climate), human biology, or increases or decreases of human knowledge or technology. These factors in the DOE models remain constant throughout the evaluation period with characteristics that existed when the license application was submitted. The NRC staff also concludes that the applicant's model is consistent with arid or semiarid conditions by including biosphere pathways that are consistent with arid or semiarid conditions, such as the resuspension of dry soils and ash by wind, in compliance with the requirements of 10 CFR 63.305(d).

Integration of Biosphere Model in the TSPA

The NRC staff reviewed the integration between the biosphere model abstraction and other TSPA abstractions for couplings among models that share or utilize similar information and consistency of assumptions among models. This review was conducted because the abstraction models must be designed and implemented to function as intended within the larger TSPA model (i.e., they must correctly receive and pass data), and the abstraction models that share FEPs (e.g., climate change can affect both Yucca Mountain infiltration and biosphere conditions) are expected to have consistent representations of FEPs (e.g., assumptions) to avoid bias in TSPA calculations. Based upon the NRC's review of the applicant's integration of the TSPA biosphere model abstraction with other TSPA model abstractions, the NRC staff concludes that the integration of the biosphere abstraction is acceptable for calculating the biosphere transport and dose to the RMEI in the TSPA code. This conclusion is based on the evaluation of direct couplings between the TSPA biosphere abstraction and other TSPA model abstractions and evaluation of shared assumptions in the biosphere and other abstractions.

The NRC staff's evaluation of direct couplings between the TSPA biosphere abstraction and other TSPA model abstractions considered the flow of information from other abstractions to the biosphere abstraction within selected TSPA model files. Specifically, the applicant's model file for seismic ground motion for the 1-million-year modeling case passes radionuclide-specific saturated zone model results (radionuclide groundwater concentrations) to the biosphere model where those results are multiplied by the groundwater exposure scenario biosphere dose conversion factors. The applicant's model file for the volcanic eruption modeling case passes the ash redistribution model results (radionuclide and pathway-specific soil concentrations) for multiplication by the volcanic ash exposure scenario radionuclide and pathway-specific biosphere dose conversion factors. Thus, the NRC staff finds that couplings between the biosphere model abstraction and other model abstractions are in agreement with the applicant's documentation of the calculations (SAR Sections 2.3.10.5.1.2 and 2.3.10.5.2.2). These couplings are also consistent with the NRC staff's technical understanding of model integration.

The NRC staff also evaluated other, less direct, points of model integration addressing consistency between the TSPA biosphere abstraction and other abstractions, including assumptions. The NRC staff finds that the biosphere model abstraction includes biosphere dose conversion factors applicable to the radionuclides identified in the inventory analysis that was developed for the postclosure performance analyses (SAR Section 2.3.7.4.1.2). Consistency in the radionuclides evaluated in the TSPA abstractions is important to ensure that the biosphere model includes biosphere dose conversion factors for the radionuclides that are included in the other TSPA abstraction models. The NRC staff also evaluated how the applicant integrated climate evolution in its biosphere modeling with the climate evolution considered in the infiltration and unsaturated zone flow process models. The NRC staff evaluated the applicant's technical bases for this climate implementation approach for the biosphere and finds that DOE sufficiently demonstrated that the use of current climate biosphere dose conversion factors (i.e., not explicitly modeling separate climate states in the biosphere model) is conservative and adequate for use in the RMEI dose calculations throughout the period of geologic stability in the biosphere model. The applicant quantitatively evaluated the effects of climate change as follows. The applicant's analysis (i) evaluated biosphere model parameters on the basis of the expected parameters impacted by climate change, (ii) derived values for these parameters on the basis of its analysis of potential future climate states, and (iii) executed biosphere calculations for three separate climate states (present-day interglacial, monsoon, glacial transition). The results showed that future climate evolution in the biosphere lowers dose to the RMEI (SAR Section 2.3.10.5.1.1). The NRC staff finds the applicant's methods, including use of the same biosphere model with different sets of climate-dependent input parameters to evaluate the effect of climate change on biosphere dose conversion factor results, are acceptable for evaluating whether TSPA calculations should include separate sets of biosphere dose conversion factors for each climate state. The applicant's result was also consistent with the results from prior NRC-sponsored biosphere analyses (LaPlante and Poor, 1997aa). Both the NRC's and the applicant's analyses suggested that future climate states, which are expected to be cooler and wetter than the current climate, would result in the RMEI using less water (e.g., irrigation), and would, therefore, lower the amount of radionuclides deposited to soils and lower the calculated RMEI dose. Therefore, the NRC staff finds that DOE provided an adequate technical basis for its conclusion that the biosphere dose conversion factors calculated for the current climate state (modern interglacial climate) are conservative for calculating the dose to the RMEI in the TSPA biosphere abstraction throughout the period of geologic stability (SAR Section 2.3.10.5.1.1). The NRC staff finds that this evaluation of climate change meets the requirements of 10 CFR 63.305(b).

In summary, after reviewing the applicant's system description and model integration, the NRC staff concludes that the applicant has demonstrated compliance with the requirements in 10 CFR 63.114(a)(5); 10 CFR 63.305; 10 CFR 63.311(b); and 10 CFR 63.312(a), (b), (d), and (e). This conclusion is based on the NRC staff's review of information discussed in this subsection, including the applicant's description of the characteristics of the Yucca Mountain region, the documentation of FEPs the applicant has included in the biosphere model, and the integration of included FEPs into the conceptual models of the biosphere system.

2.2.1.3.14.3.2 Assessment of Biosphere Transport Pathways

A series of integrated submodels in the DOE ERMYN biosphere model calculates radionuclide transport through pathways within the biosphere. Five transport submodels (surface soil, plant uptake, animal uptake, fish uptake, and air) calculate environmental media concentrations used in the ERMYN calculations of biosphere dose conversion factor input parameters for the TSPA model. The NRC staff's review of the surface soil submodel is documented in SER Section 2.2.1.3.13.3.2. This section documents the NRC staff's review of the applicant's technical bases for input parameters, treatment of parameter uncertainty, and, as appropriate, evaluation of alternative conceptual models applicable to the biosphere transport submodels in ERMYN. The NRC staff's risk-informed review focused on transport submodels and applicable input parameters for exposure pathways that contribute most to the TSPA results, as discussed in SER Section 2.2.1.3.14.3.

These submodels address plant uptake, animal uptake, fish uptake, and air transport. Air transport includes localized resuspension of particulates from soil or ash, generation of indoor evaporative cooler aerosols, and the release of radon gas from soil or ash. While groundwater-release-related modeling cases are the primary contributors to the total TSPA dose results (as summarized in SER Section 2.2.1.3.14.3), the NRC's review of the applicant's biosphere transport models also included analysis and findings regarding risk-significant aspects of the volcanic-ash-related biosphere transport modeling. The applicant's TSPA dose results documented in SAR Figure 2.4-32 list Pu-239 and Pu-240 as the largest contributors to its calculated peak mean annual dose for the volcanic ash modeling case. The applicant further documented that the RMEI inhalation of resuspended particulates was the predominant pathway for the volcanic ash exposure scenario's biosphere dose conversion factors for Pu-239 and Pu-240 (SAR Table 2.3.10-15). The applicant's information is consistent with the NRC staff results that show inhalation of resuspended particulates is a predominant contributor to volcanic exposure scenario dose results (NRC, 2005aa; LaPlante and Poor, 1997aa). Transport submodels for plant uptake, animal uptake, and radon are also used in volcanic ash biosphere dose conversion factor calculations, but contribute less to the applicant's calculated peak mean annual dose for the volcanic ash modeling case (SAR Figure 2.4-32; SAR Table 2.3.10-15).

Plant Uptake Submodel

The applicant's plant uptake submodel in the ERMYN code (SAR Section 2.3.10.3.1.3) calculates plant radionuclide concentrations on the basis of direct deposition of irrigation water and dust on plant surfaces and root uptake from estimated soil radionuclide concentrations computed by the surface soil model (or provided as direct model input for volcanic ash biosphere dose conversion factor calculations, as discussed in SAR Section 2.3.10.3.2.1). For root uptake processes, soil-to-plant transfer factors are used as input parameters to calculate plant radionuclide concentration from the radionuclide concentration in the soil. DOE selected soil-to-plant transfer factors from laboratory and field study results obtained from available literature using the methods discussed in BSC (2004ap).

DOE evaluated soil-to-plant transfer factors from data obtained through a variety of references, including original data from literature reviews and biosphere analyses that selected and reported values from available sources. For its analysis, the applicant identified five unique crop groups: leafy vegetables, other vegetables, fruit, grain, and forage. For each crop group and radionuclide, DOE selected soil-to-plant transfer factor values that it considered most applicable to the Yucca Mountain biosphere conditions, based upon area soil characteristics and crop types (SAR Section 2.3.10.3.1.3). The applicant then calculated geometric means and standard deviations from the values selected from each reference. The applicant assumed that soil-to-plant transfer factors followed truncated lognormal distributions with a 99 percent confidence interval around the geometric mean of the selected point values (BSC, 2004ap).

The NRC's review of the applicant's soil-to-plant transfer factor input parameters focused on the adequacy of supporting data sources, the selection of values, the applicability of selected values to Yucca Mountain biosphere calculations, and the applicant's approach to propagating uncertainty and variability in selected values. The NRC staff also reviewed the magnitude of the applicant's selected geometric mean values and geometric standard deviations for key contributing radionuclides in relation to the values provided by the supporting references from which the DOE values were derived. Data sources the applicant used to derive input parameters included (i) commonly referenced original data compilations that evaluated a variety of peer-reviewed field and laboratory studies and (ii) other technical analyses that reported soil-to-plant transfer factors selected or derived from available source data or compilations. These references provide a technical basis for selecting composite transfer factor values because they include the most extensive international literature compilation of scientific data on the topic (International Atomic Energy Agency, 1994aa), as well as a variety of technical reports authored by various radiological assessment practitioners, including the NRC. Therefore, the NRC staff finds that the referenced documents are representative of available scientific data on soil-to-plant transfer factors, because the applicant's transfer factors reasonably incorporate available scientific data.

A number of approaches can be used to select soil-to-plant transfer factors from crop-specific values to derive representative values for plant groups from available data sources. The applicant averaged applicable point estimates from a combination of original data sources and other documented analyses; this approach results in selecting geometric mean values that are representative of values presented in the source documents. For example, as shown in BSC (2004ap, Table 6-2), the applicant evaluated the data for soil-to-plant transfer of technetium in leafy vegetables. This data included eight point values ranging from 9.5 to 180. The DOE-derived truncated lognormal distribution ranged from 3.8 to 550 with a geometric mean of 46. Therefore, the DOE approach resulted in a derived distribution that includes the range of values presented in the source documents.

In addition to reviewing the applicant's approach, the NRC staff evaluated a subset of transfer factor values for technetium, iodine, neptunium, americium, and plutonium, as provided in BSC (2004ap, Section 6.2.1.2). This subset includes radionuclides the NRC staff identified as risk significant to the applicant's TSPA results (SER Section 2.2.1.3.14.3). The NRC staff compared the applicant's values with values independently selected from the available literature and reported in prior NRC-sponsored documents and analyses (NRC, 1992ae; LaPlante and Poor, 1997aa). For example, considering the prior example of the applicant's transfer factor for technetium in leafy vegetables, where the source data ranged from 9.5 to 180, the values from NRC (1992ae) and LaPlante and Poor (1997aa) were 44 and 76, respectively. These results are consistent with the geometric mean of 46 that the applicant chose. On the basis of similar evaluations conducted by the NRC staff for the remaining radionuclides in the evaluated subset, the NRC staff finds the DOE-derived geometric mean values are within reasonable ranges of the NRC-reported values. The NRC staff concludes that the applicant has provided an adequate technical basis for the soil-to-plant transfer factors used in the biosphere model. This is based on the results of NRC staff's review of (i) the applicant's approach to deriving geometric mean soil-to-plant transfer factors and (ii) the resulting geometric mean values for a subset of the factors.

The applicant's approach to propagating uncertainty in the assumed lognormal distributions of soil-to-plant transfer factors generated ranges that, as demonstrated in the technetium-derived lognormal distribution example, encompass the values reported in the source documents. The NRC staff, therefore, concludes that the applicant's approach has an adequate technical basis for use in the TSPA biosphere calculations. By deriving a distribution that encompasses the values reported in the source documents, the applicant ensured that parameter sampling in the ERMYN code selects input parameter values from a distribution that encompasses the range of values that are reported in the source documents. Because the source documents are representative of the available scientific data on soil-to-plant transfer factors, this approach results in biosphere dose conversion factor calculations that use soil-to-plant transfer factors based on available scientific data.

For direct deposition of radionuclides on plant surfaces, the plant uptake model in the ERMYN code calculates the radionuclide concentrations in crops from leaf uptake and retention of intercepted irrigation water and dust (SAR Section 2.3.10.3.1.3). These calculations are based on the irrigation deposition rate or dust deposition rate, the fraction of irrigation that originated from above-plant spraying, the crop interception fraction, the translocation factor (fraction of deposited radionuclides that are absorbed and move to other parts of the plant), the weathering half-life (removal rate of contaminants from leaves), the crop growing time, and the crop yield, as identified in SAR Section 2.3.10.3.1.3 and SNL (2007ac, Sections 6.4.3.2 and 6.4.3.3).

The NRC staff's review of the applicant's sensitivity and uncertainty analysis of the groundwater biosphere dose conversion factors, as provided in SNL (2007ac, Section 6.13 and Table 6.13-3), finds that the direct deposition inputs had relatively low, or no, effect on biosphere dose conversion factor distributions because soil-based exposure pathways had a greater effect on the plant pathway contribution to biosphere dose conversion factor distributions. These results are consistent with the applicant's conservative modeling approach to calculating soil concentrations from irrigation deposition until equilibrium concentration is reached (see SAR Section 2.2.1.3.13.3.2). Therefore, the NRC staff conducted a general review of these inputs to verify that the applicant provided technical bases. The NRC staff review concluded that the applicant has provided the technical bases for the direct deposition input parameters in the ERMYN code. The NRC staff finds that the applicant's sensitivity analysis methods involved statistical correlation analyses of individual input parameter distributions with radionuclide-specific biosphere dose conversion factor distributions. These methods are acceptable because they are appropriate statistical analysis methods that the NRC and its licensees commonly use to quantify the relationship between individual model input parameter variability and the variability in model output.

Thus, on the basis of the information discussed in this subsection, the NRC staff finds that the applicant has accounted for uncertainty and variability in parameter values and has provided the technical bases for parameter ranges, probability distributions, or bounding values used in the plant uptake submodel in its performance assessment for the 10,000-year compliance period, in accordance with the requirements in 10 CFR 63.114(a)(2), and carried through the period of geologic stability, as required in 10 CFR 63.114(b).

Animal Uptake Submodel

SAR Section 2.3.10.3.1.4 described the applicant's ERMYN code animal uptake submodel. This submodel calculates radionuclide concentrations in human food products that are derived from livestock that ingest contaminated food and water. For the purpose of modeling, the applicant identified four distinct animal product groups: meat, milk, eggs, and poultry. The animal product radionuclide concentrations were calculated on the basis of estimated animal intakes of radionuclides from contaminated feed, water, and soil, as applicable to the groundwater or volcanic eruption modeling cases. Animal feed radionuclide concentrations are computed by the plant uptake submodel (e.g., the applicant assumes cows eat locally grown forage and chickens eat local grain). As discussed in BSC (2004ap), the applicant used animal product transfer coefficients as input parameters for the fraction of an animal's daily intake of a radionuclide that is transferred to a unit mass or volume of produced food product. The applicant's animal product transfer coefficients were selected using the same methods (BSC, 2004ap) described in the previous subsection for soil-to-plant transfer factors.

The NRC staff evaluated the sufficiency of the applicant's technical bases and supporting data for the selected values and uncertainty ranges for the animal product transfer coefficients used in the ERMYN code to calculate biosphere dose conversion factor input parameters for the TSPA model. Data sources the applicant used to derive the animal product transfer coefficient input parameters included (i) commonly referenced original data compilations that evaluated a variety of peer-reviewed studies and other technical reports and (ii) other technical analyses that reported animal product transfer coefficients selected or derived from available source data or compilations. These references provide a technical basis for selecting generally applicable animal product transfer coefficient values because they include the most extensive international literature compilation of scientific data on the topic (International Atomic Energy Agency, 1994aa) as well as a variety of technical reports authored by various radiological assessment practitioners, including the NRC. Therefore, the NRC staff finds that the referenced documents are representative of the available scientific data on animal product transfer coefficients. The documents were reviewed and compiled by experts in the field and provide an acceptable body of technical information to support the applicant's derivation of input parameters for the biosphere model.

In reviewing the applicant's derivation of animal product transfer coefficients for the biosphere model, the NRC staff finds that the applicant's approach of averaging applicable point estimates from a combination of original data sources and other documented analyses results in selecting geometric mean values that are representative of values presented in the source documents. For example, the data the applicant evaluated [provided in BSC (2004ap, Table 6-39)] for the transfer of technetium from feed to meat include 14 point values ranging from 1.0×10^{-4} to 8.7×10^{-3}. DOE derived truncated lognormal distribution ranges from 6.9×10^{-6} to 1.8×10^{-1}

with a geometric mean of 1.1×10^{-3}. Therefore, the DOE approach resulted in a derived distribution that included the range of values presented in the source documents.

In addition to the review of the applicant's approach, the NRC staff evaluated a subset of the applicant's transfer factor values for technetium, iodine, neptunium, americium, and plutonium, as identified in BSC (2004ap, Section 6.3.3). This subset included radionuclides the NRC staff identified as risk significant to the applicant's TSPA results (SER Section 2.2.1.3.14.3). The NRC staff compared the DOE geometric mean values with values independently selected from the available literature and reported in prior NRC documents and analyses (NRC, 1992ae; LaPlante and Poor, 1997aa). For example, the values for technetium from NRC (1992ae) and LaPlante and Poor (1997aa) are 8.5×10^{-3} and 1.0×10^{-4}, respectively. The NRC staff finds that these data are consistent with the geometric mean of 1.1×10^{-3} that the applicant derived. When used in the applicant's dose model, the applicant's higher value in this comparison is more conservative than the NRC-reported values because it would transfer more radionuclides to meat and, therefore, increase the dose to the RMEI relative to the NRC-reported values. On the basis of similar evaluations conducted by the NRC staff for the remaining radionuclides in the evaluated subset, the NRC staff finds that the DOE-derived geometric mean values are within reasonable ranges of the NRC-reported values. After reviewing the applicant's approach to deriving geometric mean animal product transfer coefficients and the resulting geometric mean values for a subset of the coefficients reviewed, the NRC staff concludes that the applicant has provided an adequate technical basis for the animal product transfer coefficients used in the biosphere model.

In reviewing the applicant's approach to propagating uncertainty in the assumed lognormal distributions of animal product transfer coefficients, the NRC staff finds that the approach generated ranges that encompass the values reported in the source documents and, therefore, has an adequate technical basis for use in TSPA biosphere calculations. DOE assumed a truncated lognormal distribution using the geometric standard deviation computed from the source data and applied a 99 percent confidence interval approach similar to that used for deriving parameter distributions for soil-to-plant transfer factors. By deriving a distribution that encompassed the values reported in the source documents, the applicant ensured that parameter sampling in the ERMYN code selects input parameter values from a distribution that encompasses the range of values that are reported in the source documents. Because the source documents are representative of the available scientific data on animal product transfer coefficients, this approach resulted in biosphere dose conversion factor calculations that use animal product transfer coefficients based on available scientific data.

In summary, on the basis of the information discussed in this subsection, the NRC staff finds that the applicant has accounted for uncertainty and variability in parameter values and has provided the technical bases for parameter ranges, probability distributions, or bounding values used in the animal uptake submodel in its performance assessment for the 10,000-year compliance period, in accordance with the requirements in 10 CFR 63.114(a)(2), carried through the period of geologic stability, as required in 10 CFR 63.114(b).

Fish Uptake Submodel

The applicant's ERMYN code fish uptake submodel (SAR Section 2.3.10.3.1.5) calculates radionuclide concentrations in fish raised in local fish farms that are assumed to use contaminated groundwater. The input parameter that most influences the results of this model is the bioaccumulation factor. This element-specific factor relates the concentration of radionuclides in the edible portion of the fish to the concentration of radionuclides in the contaminated water in which the fish is submerged. DOE selected bioaccumulation factors on the basis of a review of the applicable literature. The applicant's review included fish in all portions of the food chain as well as bottom-feeding fish. DOE assumed a lognormal distribution. For the fish farms noted in Amargosa Valley during the applicant's consumption survey, the fish were fed commercial feed that is not locally derived. Feed that is not locally derived would not be expected to be contaminated with radionuclides from a Yucca Mountain release scenario. Therefore, while the applicant applied a bioaccumulation factor that accounts for fish ingesting contaminated food and water, actual conditions suggest that only the water would be contaminated and that, therefore, the applicant's analysis would overestimate the radionuclide concentration in fish. The resulting dose to the RMEI from fish consumption is, therefore, overestimated.

The NRC staff evaluated the sufficiency of the applicant's technical bases and supporting data for the selected point values and uncertainty ranges for the bioaccumulation factors used in the fish uptake submodel. Data sources the applicant used to derive input parameters included (i) a commonly referenced original data compilation that evaluated a variety of technical reports and some peer-reviewed studies and (ii) other technical analyses that reported bioaccumulation factors selected or derived from available source data or compilations. The NRC staff finds the referenced documents are representative of available scientific data on fish bioaccumulation factors. The references were reviewed and compiled by experts in the field and include the most extensive international literature compilation of scientific data on the topic (International Atomic Energy Agency, 1994aa) as well as a variety of technical reports authored by various radiological assessment practitioners, including the NRC. Therefore, the NRC staff finds that these references provide a technical basis for the selection of generally applicable fish bioaccumulation values.

In reviewing bioaccumulation factor values, the NRC staff finds that the applicant's approach of averaging applicable point estimates from a combination of original data sources and other documented analyses resulted in the selection of geometric mean values that are representative of values presented in the source documents. For example, the values for fish uptake of carbon the applicant evaluated included eight point values ranging from 4.6×10^3 to 5.0×10^4 L/kg [5.5×10^2 to 6.0×10^3 gal/lb], with a DOE-derived geometric mean of 1.6×10^4 L/kg [1.9×10^3 gal/lb], as identified in BSC (2004ap, Table 6-64). On the basis of the importance of carbon and the fish pathway in the applicant's TSPA results relative to other radionuclides and pathways (SER Section 2.2.1.3.14.3), the NRC staff evaluated the applicant's supporting information for the fish bioaccumulation factor for carbon. The NRC staff compared the derived value with values independently selected from the available literature (NRC, 1992ae; International Atomic Energy Agency, 1994aa). The NRC staff found that the DOE-derived geometric mean value was within a reasonable range of the NRC- and the International Atomic Energy Agency-reported values. The values from NRC (1992ae) and the International Atomic Energy Agency (1994aa) are 4.6×10^3 and 5.0×10^4 L/kg [5.5×10^2 to 6.0×10^3 gal/lb], respectively, and are consistent with the geometric mean of 1.6×10^4 L/kg [1.9×10^3 gal/lb] the applicant derived. Because the data sources the applicant used included studies of fish in natural ecosystems that are contaminated with radionuclides, bioaccumulation factors evaluated in those documents would include contributions to fish uptake from contaminated food. Therefore, the NRC staff concludes that the applicant appropriately used these factors to model fish uptake within the model where uncontaminated (nonlocally derived) feed is expected, resulting in an overestimation of dose from fish uptake.

The NRC staff reviewed the applicant's approach to propagating uncertainty in the assumed lognormal distributions of fish bioaccumulation factors and finds that the approach generated ranges that encompass the values reported in the source documents. This is, therefore, considered acceptable for use in TSPA biosphere calculations. DOE assumed a truncated lognormal distribution using the geometric standard deviation computed from the source data and applying a 99 percent confidence interval approach. As an example, the applicant-derived lognormal distribution of fish bioaccumulation factors for technetium ranged from 3.3 to 120 L/kg [0.4 to 14 gal/lb], which encompasses the range of point values {15 to 78 L/kg [1.8 to 9.4 gal/lb]} in the applicant's source documents.

The NRC staff's review of the fish bioaccumulation factors identified an apparent transcription error in the DOE report (BSC, 2004ap). The geometric mean of the fish bioaccumulation factor for carbon was reported differently in two separate tables that should have contained the same values. Specifically BSC (2004ap, Table 6-64), which is the table that initially derives the value from source data, showed a geometric mean fish bioaccumulation factor for carbon of 1.6×10^4 L/kg [1.9×10^3 gal/lb]. BSC (2004ap, Table 6-65) listed a different value of 4.6×10^3 L/kg [5.5×10^2 gal/lb] for this same parameter (a factor of 3.5 lower than the original value computed in Table 6-64). The applicant used the lower value reported in Table 6-65 in the ERMYN calculations, as indicated by SAR Table 2.3.10-10. To evaluate the significance of this discrepancy, the NRC staff considered whether using the higher value would significantly affect the applicant's dose results. This evaluation considered that the applicant's fish consumption dose scales linearly with the bioaccumulation factor. The NRC staff also evaluated the TSPA results for the 10,000-year simulation period in SAR Figure 2.4-20, which shows that the carbon dose contributes approximately 20 percent of the peak mean total dose. SAR Table 2.3.10-11 indicated that the fish pathway contributed 59 percent to the carbon dose. The NRC staff's evaluation of these results of the applicant's TSPA demonstrates that correcting the geometric mean bioaccumulation factor would not significantly change the all-radionuclide total TSPA results.

In summary, on the basis of the information discussed in this subsection, the NRC staff finds that the applicant has accounted for uncertainty and variability in parameter values and has provided the technical bases for parameter ranges, probability distributions, or bounding values used in the fish uptake submodel in its performance assessment. This is in accordance with the requirements in 10 CFR 63.114(a)(2) for the 10,000-year compliance period, carried through the period of geologic stability, as required in 10 CFR 63.114(b).

Air Submodel

The applicant's ERMYN air submodel (SAR Sections 2.3.10.3.1.2 and 2.3.10.3.2.2) models radionuclide concentrations in air from (i) resuspension of contaminated soil or ash, (ii) evaporative cooler aerosols from the use of contaminated groundwater, and (iii) radon gas emanation from contaminated soil or ash. Inhalation of resuspended particulates is the predominant exposure pathway for Pu-239, Pu-240, and Pu-242 in the applicant's performance assessment. Resuspended particulate exposure is modeled in both the groundwater and volcanic ash exposure scenarios. Particulate inhalation also contributes approximately 21 percent to the groundwater dose from Np-237 in the DOE model (SAR Table 2.3.10-11). The other air pathways in the applicant's model contribute less to the performance assessment results than particulate inhalation but are represented in the radionuclides that contribute most to the applicant's results, as summarized in SER Tables 16-1 and 16-2. Therefore, the NRC staff's detailed review of the technical bases for input parameters and ranges in the air submodel discussed in this SER section evaluated those parameters in the applicant's biosphere transport calculations involving particulates, evaporative cooler aerosols, and radon gas.

An important input parameter in the DOE calculation of air concentration of particulates is the mass loading factor {g/m^3 [lb/ft^3]} based on the DOE biosphere dose conversion factor sensitivity analysis results documented in SNL (2007ac, p. 6-150) and SAR Table 2.3.10-17. This is because the radionuclide concentration in air is proportional to the mass loading factor. The mass loading factor computes the concentration of radionuclides in air {Bq/m^3 [Ci/m^3]} from the estimated concentration of radionuclides deposited on the soil surface {Bq/g [Ci/g]}. The NRC staff reviewed the DOE sensitivity analysis methods and concludes that the DOE approach, which includes a statistical correlation analysis of sampled ERMYN model input and biosphere dose conversion factor distribution output, acceptably quantifies the correlation between model input variation and output variation as a means of identifying the sampled input parameters that most influence model results. These methods are acceptable because they are appropriate statistical analysis methods that the NRC and its licensees commonly use to quantify the relationship between individual model input parameter variability and the variability in model output.

In SAR Sections 2.3.10.3.1.2 and 2.3.10.3.2.2 and supporting references, DOE described its derivation of separate mass loading factors for each exposure scenario (i.e., irrigated soil, volcanic ash) on the basis of available literature. DOE derived individual mass loading input parameters for RMEI activity level (active and inactive) and environment (outdoor, indoor, or away from potentially contaminated areas) in its inhalation exposure model.

The NRC staff evaluated the adequacy of the applicant's technical bases and supporting data for the mass loading input parameters used in biosphere exposure scenarios involving groundwater and volcanic ash. The applicant detailed these in BSC (2006ad). In reviewing the applicant's technical bases for groundwater exposure scenario mass loading input parameter values, the NRC staff evaluated the adequacy of the supporting data the applicant used to derive mass loading input parameters against independent NRC estimates derived from available technical information. The NRC staff's review of the supporting data finds that DOE evaluated a range of studies in published, peer-reviewed literature that measured airborne concentrations of total suspended particulate and PM_{10} {small suspended particles that are less than 10 micrometers [3.9×10^{-4} in] in diameter} for a variety of environments and surface-disturbing activities. The NRC staff notes that the applicant based its soil mass loading values on site-specific studies that included measurements of airborne dust in Amargosa Valley applicable to various surface-disturbing activities, including walking, pitching hay, driving, working near construction equipment, and dog walking. The applicant also considered a variety of other studies from sites that the NRC staff concludes are representative or analogous to Yucca Mountain regional conditions. These include arid or semiarid environments and rural agricultural dust-generating activities and exposure conditions. The NRC staff finds that the applicant's supporting information for mass loading values provides a broad base of technical support that addresses the effects of a range of dust-generating activities and site-specific conditions.

As part of the NRC staff's review of the applicant's groundwater exposure scenario mass loading values, the NRC staff evaluated the magnitude of the applicant's selected values. This evaluation involved comparing the results of two of the NRC staff's calculations of the mass of soil the RMEI inhaled (LaPlante, 2010aa). In these calculations, the mass of soil inhaled is the product of constant values for input parameters for mass loading, exposure time, and breathing rate. One of these calculations was based on the applicant's mass loading values, and the other calculation was based on the input parameters the NRC staff derived for the NRC's TSPA 5.1 code by evaluating the available peer-reviewed and other scientific literature (Leslie, et al., 2007aa). Exposure time and breathing rate input parameters in both calculations were set to the same values to isolate the effect of the differences in mass loading inputs on the mass of soil inhaled. Due to the large number of individual mass loading input parameters the applicant used, this calculation efficiently evaluated the combined effect of the applicant's mass loading parameter choices on an intermediate result in the inhalation dose calculation (i.e., mass of soil inhaled). This comparison showed the calculated daily mass of soil resuspended and inhaled based on the DOE mass loading values was 2.5 times larger than the same result computed using the NRC staff's derived mass loading values. This indicates the applicant's selected mass loading values are more conservative than the values that the NRC staff independently derived from available scientific data. Therefore, the NRC staff finds that the magnitude of the applicant's derived values for mass loading produce dust inhalation results that are greater than results based on independently derived mass loading and other applicable input parameters, as identified in Leslie, et al. (2007aa, Table 17-1). Therefore, the NRC staff concludes that the applicant's methods and technical bases for these input parameters are adequate and do not underestimate dose.

For the volcanic ash scenario, the NRC staff recognizes that limited data are applicable to mass loading for a volcanic eruption in the Yucca Mountain region or for analogous conditions elsewhere. DOE reviewed literature that included measured dust levels of volcanic ash resuspended in air for ambient and surface-disturbing conditions at various sites where volcanoes had recently erupted (within 5 years) and also compared the relevance of each analog site (including the Soufrière Hills Volcano in Montserrat, British West Indies, and the Mt. Spurr Volcano in Alaska) to expected conditions in the Yucca Mountain region.

The NRC staff's review of the applicant's technical bases for volcanic ash exposure scenario mass loading input parameter values evaluated the adequacy of the supporting data the applicant used and the methodology used to derive mass loading input parameters. The NRC staff also evaluated the magnitude of the applicant's values against independent NRC estimates derived from available technical information. The NRC staff finds that the applicant's consideration of a range of studies that included dust-level measurements taken during a variety of surface-disturbing conditions at volcanic eruption sites provides adequate technical support for its derived mass loading values.

The NRC staff's review of the magnitude of the applicant's volcanic ash exposure scenario mass loading values involved comparing two NRC staff calculations of the mass of resuspended ash that would be inhaled by the RMEI (LaPlante, 2010aa). In these calculations, the mass of ash inhaled is the product of constant values for input parameters for mass loading, exposure time, and breathing rate. One of these calculations was based on the applicant's mass loading values, and the other calculation was based on the input parameters NRC staff derived for the TSPA 5.1 code by evaluating the available peer-reviewed and other scientific literature, as documented in NRC Table 17-1 (Leslie, et al., 2007aa). Exposure time and breathing rate input parameters in both calculations were set to the same values (Leslie, et al., 2007aa) to isolate the effect of the differences in mass loading inputs on the mass of ash inhaled. Due to the large number of individual mass loading input parameters the applicant used, this calculation efficiently evaluated the combined effect of the applicant's mass loading parameter choices on an intermediate result in the inhalation dose calculation (i.e., mass of ash inhaled). This comparison showed the calculated daily mass of resuspended, inhaled ash based on the DOE mass loading values was consistent with the value computed using the mass loading values the NRC staff independently derived from available scientific data. Therefore, the NRC staff concludes that the magnitude of the applicant's derived values for mass loading, when evaluated in the context of their effect on dose to the RMEI (i.e., using the calculation of the mass of ash inhaled), produces dust inhalation results that are within a reasonable range of results based on independently derived mass loading values and other applicable input parameters, as documented in NRC Table 17-1 (Leslie, et al., 2007aa). This independent verification of the applicant's derived mass loading input parameters further supports the NRC staff's conclusion that the applicant's methods and technical bases for these input parameters are adequate.

The NRC staff reviewed the applicant's treatment of uncertainty and variability in mass loading values for both groundwater and volcanic ash exposure scenarios and concludes that the applicant has accounted for uncertainty and variability in parameter values and has provided the technical bases for parameter ranges and probability distributions used in the air submodel. The NRC staff finds the applicant's approach for developing parameter distributions acceptable because its values are supported by applicable scientific studies. Although supporting data are limited, the applicant reviewed sufficient information to derive input parameter ranges and a mode to characterize simple distributions for use in the TSPA code. The applicant accomplished this by considering the range of values from the available literature and the applicability of each study to the Yucca Mountain exposure scenarios in terms of similar surface-disturbing activities, an arid or semiarid environment, and measurements of total suspended particulates. The applicant then addressed uncertainty and variability in the mass loading parameters [provided in BSC (2006ad, Sections 6.2 and 6.3)] by deriving triangular parameter distributions. These distributions were based on its assessment of the range of applicable literature values and the central tendencies in the data that support selection of a value for the mode of each distribution. The applicant conducted similar literature-based evaluation and selection of mass loading ranges and modes to characterize triangular input distributions for each activity environment and for groundwater and ash exposure scenarios. The resulting input distributions were provided in BSC (2006ad, Table 7-1).

The applicant evaluated personal exposure measurements of total suspended particulates collected during farming activities at 10 farms near Davis and Sacramento, California, that supported a range of 0.30 to 7.93 mg/m^3 [8.1×10^{-6} to 2.1×10^{-4} oz/yd^3]. After evaluating additional data from 7 other studies involving mostly farming activities and 22 sets of measurements taken in Amargosa Valley for various types of activities, a range of 1 to 10 mg/m^3 [2.7×10^{-5} to 2.736×10^{-4} oz/yd^3] {with a mode of 3 mg/m^3 [8.1×10^{-5} oz/yd^3]} was derived for the ERMYN input for the TSPA analyses for mass loading in the active outdoor environment for groundwater-based biosphere dose calculations. The mode of 3 mg/m^3 [8.1×10^{-5} oz/yd^3] was also the mean of the maximum mass loading values measured for 22 surface-disturbing activities in Amargosa Valley, as identified in BSC (2006ad, Section 6.2.1.3).

The NRC staff considers the applicant's mass loading data adequate for inclusion in the biosphere model of Yucca Mountain for three reasons: (i) the data describe activities that are consistent with the characteristics of the Yucca Mountain region (e.g., farming, arid conditions), (ii) the data are based on measurements taken from the Yucca Mountain region, and (iii) the data include breathing zone sampling measurements. The breathing zone sampling data are particularly relevant for supporting an inhalation exposure scenario because the measurements were taken near the head of the individual involved in dust generating activities and, therefore, are more representative than ambient dust measurement of the concentration of resuspended particulates that would be inhaled. On the basis of the NRC staff's review of the applicant's methods for deriving uncertainty distributions in the mass loading values, the NRC staff finds that the applicant has addressed uncertainty and variability in the mass loading parameter values and has provided the technical bases for parameter ranges and probability distributions used in the air submodel in the ERMYN code.

As discussed in SAR Section 2.3.10.3.1.2, the air submodel in the ERMYN code also calculates indoor air radionuclide concentrations from aerosols released from evaporative coolers. This calculation is based on the concentration of radionuclides in groundwater, the rate of water evaporation from coolers, the indoor air exchange rate, and the fraction of radionuclides in water that transfer to air (the water-to-air transfer fraction). DOE identified the water-to-air transfer fraction in the evaporative cooler model as an important input parameter in the evaporative cooler calculation. The NRC staff evaluated the derivation of the water-to-air transfer fraction and concludes that values of the water-to-air fraction were selected conservatively to bound possible values. The applicant assumed a uniform concentration ratio distribution from 0 to 1 for dissolved solids and 1 for gases on the basis of a lack of available studies on contaminant aerosols from evaporative coolers. The NRC staff concludes that this value is conservative because dissolved solids do not evaporate when water evaporates (the same process is used to purify water by distillation). The assumed distribution causes the model to release an average of 50 percent of the dissolved solids (including dissolved radionuclides) that are in the groundwater directly to indoor air. This increases the inhalation dose from aerosols beyond what would be expected under actual conditions. Therefore, the NRC staff concludes that the applicant has addressed uncertainty and variability in parameter values and has provided the technical bases for parameter ranges, probability distributions, or bounding values used in the aerosol release from evaporative cooler calculation within the ERMYN air submodel.

Airborne concentrations of radon gas released from soils irrigated with contaminated water or from contaminated volcanic ash were also calculated in the air submodel (SAR Sections 2.3.10.3.1.2 and 2.3.10.3.2.2; SNL, 2007ac; BSC, 2004ap). Both exposure scenarios consider indoor and outdoor radon concentrations. In the volcanic ash scenario, the outdoor air concentration is also used for the indoor air concentration because the applicant expected the outdoor concentration to be higher than the indoor concentration as a result of the small radon contribution from ash below the RMEI's house. The applicant's groundwater scenario calculates separate radon concentrations for indoor and outdoor environments. The applicant's indoor radon concentration calculations evaluated radon released from soil beneath a hypothetical house built on land that was previously irrigated by contaminated water. In this model, the rate of radon released into the house is a proportion of the outdoor radon flux that accounts for diffusion of radon from underlying soil through the foundation. Indoor radon concentrations in the model were calculated based on (i) the radon flux into the house from soil beneath the house and from outdoors, (ii) the interior air exchange rate, and (iii) the interior volume of the house. The interior air exchange rates account for periods of evaporative cooler use and nonuse based on increased ventilation during cooler operation, which decreases radon concentration in air. The indoor radon diffusion methods are consistent with those used in the RESRAD dose assessment code (Yu, et al., 2001aa) that the EPA developed. Outdoor radon concentrations are based on factors that relate the airborne concentration of Rn-222 to either (i) the Ra-226 concentration in the soil for the groundwater scenario or (ii) the Rn-222 flux density for the volcanic ash scenario.

The NRC staff evaluated the applicant's technical bases and supporting data for input parameters used in the indoor and outdoor radon concentration modeling in the ERMYN code. Input parameters that were reviewed included the fraction of radon flux entering the foundation from soil, and home ventilation rates. DOE chose the concentration fraction of the radon flux from soil underneath the house that would diffuse into the house to be uniformly distributed from 0.1 to 0.25 on the basis of measurements in homes with concrete foundations (SAR Section 2.3.10.3.1.2). The home ventilation rates (for evaporative cooler nonuse periods) were based on minimum ventilation recommendations for manufactured homes, data from a survey of approximately 3,000 U.S. homes, and information from a trade organization representing home ventilation equipment manufacturers (BSC, 2004ap). The home ventilation rates for evaporative cooler use were estimated on the basis of cooler flow rates and the average home interior volume. Uncertainty and variability in the ventilation rates were propagated by deriving a truncated lognormal distribution on the basis of the survey data (for no cooler use) and a uniform distribution for cooler use ventilation rates that spans the estimated range (BSC, 2004ap). The NRC staff finds the applicant's use of diverse information sources, including information applicable to manufactured homes, national survey data of home ventilation rates, and ventilation equipment information to be appropriate for its radon concentration calculation. The NRC staff, therefore, finds that the applicant has provided an adequate technical basis for indoor and outdoor radon concentration input parameters. The NRC staff finds that the approaches used to derive these distributions use common methods that result in distributions that adequately represent the values reported in the referenced information. On the basis of the review of the supporting documentation, the NRC staff finds that the applicant has accounted for uncertainty and variability in parameter values and has provided the technical bases for parameter ranges and probability distributions used in the air submodel radon concentration calculations. The radon concentrations are used for calculating biosphere dose conversion factor input parameters for the TSPA code.

On the basis of information discussed in this subsection, including the NRC staff's review of the data supporting selected input parameters and distributions that the applicant used to model the inhalation of resuspended soil and ash, aerosols from evaporative coolers, and radon released from irrigated soils, the NRC staff finds that the applicant has accounted for uncertainty and variability in parameter values and has provided the technical bases for parameter ranges, probability distributions, or bounding values used in the air submodel in its performance assessment. This follows 10 CFR 63.114(a)(2) requirements to address the 10,000-year compliance period, carried through the period of geologic stability, as required in 10 CFR 63.114(b).

2.2.1.3.14.3.3 Assessment of Human Exposure

DOE calculated human exposures from estimated concentrations of radionuclides in groundwater and soil in three exposure submodels of the ERMYN code (SNL, 2007ac). These submodels are the primary exposure pathways addressed in the DOE exposure scenarios and include the external exposure submodel, the inhalation exposure submodel, and the ingestion exposure submodel (SAR Sections 2.3.10.3.1.7, 2.3.10.3.1.8, and 2.3.10.3.1.9 for the groundwater exposure scenario and SAR Sections 2.3.10.3.2.5, 2.3.10.3.2.6, and 2.3.10.3.2.7 for the volcanic ash exposure scenario). Considering the biosphere pathways that are the primary contributors to dose to the RMEI (SER Tables 16-1 and 16-2), the NRC staff's risk-informed review focused on the inhalation and ingestion exposure submodels. These exposure submodels compute the RMEI's annual intake of radionuclides {e.g., Bq/yr [Ci/yr]} on the basis of the environmental media concentrations (e.g., air, water, livestock products, fish) calculated by the biosphere environmental transport submodels that are depicted in SAR Figures 2.3.10-9 and 2.3.10-10, described in SAR Section 2.3.10.3, and evaluated in SER Sections 2.2.1.3.14.3.2 and 2.2.1.3.12.3. The applicant's exposure models also convert the calculated RMEI radionuclide intakes to dose. The NRC staff's review of the applicant's conversion of intakes into dose is evaluated in SER Section 2.2.1.3.14.3.4.

Inhalation Exposure Model

The applicant's ERMYN inhalation exposure submodel calculates RMEI radionuclide intakes by modeling the inhalation of contaminated air. In the model, airborne contaminants include resuspended soil or ash particulates, aerosols from evaporative coolers, or radon gas emanating from contaminants in soil or ash. DOE calculations showed that the inhalation exposure pathways that are the most risk-significant contributors to the applicant's performance assessment results (SER Tables 16-1 and 16-2) are resuspended particulates from soil and aerosols from evaporative coolers. Inhalation of gaseous emissions of radon from contaminants in soil also contributes to DOE's long-term dose calculation results, but that contribution is less than the contributions from particulates and aerosols.

The applicant's exposure calculations for these three inhalation exposure pathways involve exposure time and breathing rate input parameters that are the focus of the NRC staff's review. The NRC staff focused on these parameters because (i) they directly influence the calculated dose; (ii) they are influenced by a number of other complex variables including the types of human activities, activity durations, and intensity of physical activity; (iii) they are used in all three of the inhalation exposure pathways; and (iv) the applicant's documented technical bases for these inputs are particularly complex relative to the other inhalation exposure submodel inputs such as the fraction of houses with coolers, the evaporative cooler use factor, and the equilibrium factor for radon decay products.

DOE's exposure time input parameters in the ERMYN inhalation exposure submodel apportion the amount of time the RMEI spends in various environments where exposure could occur into three categories: outdoor, indoor, and away from areas potentially contaminated by Yucca Mountain activities. The DOE inhalation exposure submodel also apportions the amount of time spent conducting surface-disturbing activities, which DOE identified as active, to account for increased exposure to radionuclides resuspended from the ground surface by the activity. The applicant identified the amount of time that the RMEI was not conducting surface-disturbing activities as inactive. The applicant grouped the Amargosa Valley population into four population categories: nonworkers, commuters, local outdoor workers, and local indoor workers. The applicant apportioned time spent into five activity–environment categories (by combining the three environment categories with the two activity-level categories): active outdoors, inactive outdoors, active indoors, inactive indoors (sleep), and away from areas potentially contaminated by Yucca Mountain activities. Exposure times were derived from census information on age distribution; employment; commuting characteristics of Amargosa Valley, Nevada, residents; and national survey data on activity times (BSC, 2005ab).

The NRC staff evaluated the applicant's technical bases and supporting data for the exposure times that were used to derive activity–environment categories (BSC, 2005ab). The applicant provided a derivation of exposure times on the basis of data from surveys of the Amargosa Valley and national populations. DOE used Year 2000 census data from the Amargosa Valley census county division (Bureau of the Census, 2002aa) for population distribution by age, work status and hours worked, commute time, and industry of employment. DOE also used detailed national survey data on activity time budgets from Klepeis, et al. (1996aa) and EPA (1997aa) to assign the fraction of time spent inside a residence; outdoors; in a vehicle; and at stores, restaurants, and other indoor locations.

The NRC staff finds that DOE's data adequately support the applicant's derivation of exposure times. This conclusion is based on the applicant's use of local and national data obtained from sources, including the Bureau of the Census and the EPA, to derive exposure times using a data synthesis approach that is transparent, traceable, and technically sound for its intended purpose. The applicant's data synthesis approach used percentages of time spent conducting activities at various locations by age group with Amargosa Valley population information to generate percentages applicable to the Amargosa Valley population. DOE then used additional national survey results (EPA, 1997aa) on time spent outdoors to derive active and inactive outdoor exposure times and apportioned the resulting times spent at various locations to derive its exposure time input parameter values. The applicant's estimates for the fraction of outdoor activity that includes surface-disturbing activities (20 percent of public outdoor time and 50 percent of construction worker outdoor time) were based on EPA survey data and information on local practices. The NRC staff evaluated a sample of values used in the derivation against source documents to verify that the values were incorporated accurately. The NRC staff also conducted simplified calculations of the documented results to verify the data synthesis and found no discrepancies or errors. DOE lognormal distributions of exposure time estimates to propagate variation in ERMYN code were based on the standard errors provided with the survey data and the application of standard lognormal distribution statistical methods. For some parameters, the applicant intentionally assigned standard errors that were larger than those associated with the national survey data to account for uncertainties in applying national data to local conditions. These uncertainties result from potential differences in local human activity practices compared to national patterns and the effect of differences in survey sample sizes on standard errors. Because propagating variation from survey data directly quantifies empirical data variability, the NRC staff concludes that the applicant has accounted for uncertainty and variability in exposure time parameter values and has provided the technical bases for parameter values, ranges, and probability distributions used in the ERMYN inhalation exposure submodel. The NRC staff also concludes that the exposure time input parameters used in the inhalation exposure calculations are consistent with the living styles of current Amargosa Valley residents as required by 10 CFR 63.312(b) because the parameters were accurately derived, as discussed previously, from survey information applicable to the local population.

The applicant derived breathing rates for each population group and for each level of activity within the four potentially contaminated exposure environments (active outdoors, inactive outdoors, active indoors, and inactive indoors). DOE combined breathing rate information for adults by gender and level of physical activity from International Commission on Radiological Protection (1994aa) with census demographic information for Amargosa Valley to derive population gender-weighted breathing rates. DOE then used information from International Commission on Radiological Protection Publication 66 (1994aa) on the fraction of daily time devoted to different levels of activity to derive breathing rates for each exposure environment category. In this manner, the exposure environment categories applied to all population groups considered in the model and were derived, in part, on the basis of surveys of Amargosa Valley, Nevada residents.

The NRC staff evaluated the applicant's technical bases and supporting data for the breathing rates detailed in BSC (2005ab). The gender-weighted breathing rate values that DOE calculated from International Commission on Radiological Protection breathing rates and census data for Amargosa Valley were 0.39, 0.47, 1.38, and 2.86 m^3/hr [0.51, 0.61, 1.81, and 3.74 yd^3/hr] for sleeping, sitting, light exercise, and heavy exercise, respectively. These values are consistent with EPA-recommended adult breathing rate values for use in short-term exposure calculations (EPA, 1997aa). The EPA values are 0.4, 0.5, 1.0, 1.6, and 3.2 m^3/hr [0.5, 0.7, 1.3, 2.1, and 4.2 yd^3/hr] for a similar progression of increasing activity level including rest, sedentary, light, moderate, and heavy activities, respectively. The EPA value for heavy activity {3.2 m^3/hr [4.2 yd^3/hr]} is somewhat higher than the DOE value. However, the EPA values apply to short-term exposures that would have higher breathing rates than values used for the long-term exposure calculations (i.e., annual dose) required by 10 CFR 63.311. Additionally, the EPA-recommended value for outdoor workers for heavy activity {2.5 m^3/hr [3.3 yd^3/hr]} is lower than the DOE value.

The NRC staff also reviewed DOE's methods for deriving the final set of breathing rates for each of the four exposure environments. The NRC staff finds the applicant's methods [BSC (2005ab, Table 6-15)] acceptable for use in long-term exposure calculations in the ERMYN biosphere inhalation exposure model because the weighted sum approach to computing a single breathing rate value applicable to all population groups for each exposure environment is found to be technically sound, transparent, and traceable. The applicant used these breathing rates in ERMYN code calculations as individual fixed input parameters for each exposure environment, and the model propagates breathing rate input parameter uncertainty or variability due to differences in activity level by using different exposure environments. Because the DOE approach accounts for variability and uncertainty in breathing rates based on human activity level and the magnitude of the breathing rates are consistent with other data sources NRC staff identified, as discussed previously, the NRC staff finds that the applicant has accounted for uncertainty and variability in parameter values and has provided the technical bases for the breathing rates used in the performance assessment.

In summary, on the basis of the information discussed in this subsection, the NRC staff finds that the applicant has accounted for uncertainty and variability in parameter values and has provided the technical bases for parameter ranges, probability distributions, or bounding values used in the inhalation exposure submodel in its performance assessment. This complies with 10 CFR 63.114(a)(2) requirements to address the 10,000-year compliance period, carried through the period of geologic stability, as required in 10 CFR 63.114(b). The NRC staff also finds that the exposure time input parameters used in the inhalation exposure calculations are consistent with the living styles of current Amargosa Valley residents, as required by 10 CFR 63.312(b), because the parameters were accurately derived from survey information applicable to the local population.

Ingestion Exposure Model

The applicant's ingestion exposure submodel in ERMYN calculates radionuclide intakes by modeling RMEI consumption of contaminated water, crops, animal products (milk, meat, poultry, eggs, fish), and soil. Ingestion pathways have a more pronounced effect on the applicant's performance assessment results during the 10,000-year compliance period than on the results for the 1-million-year period because of the radionuclides that dominate the dose calculations. Exposure pathways that contribute most to the applicant's performance assessment results (SER Section 2.2.1.3.14.3) include drinking water, fish consumption, and animal product consumption (milk, meat, eggs). The exposure calculations for these ingestion exposure pathways involve consumption rate input parameters for modeled food products the RMEI consumes (i.e., water, fish, and animal food products including milk, meat, and eggs).

DOE derived food consumption rates used in the ERMYN ingestion exposure submodel from a DOE-sponsored survey of Amargosa Valley residents (BSC, 2005ab). The exception is DOE's modeling of drinking water consumption, which the applicant stated was based on the requirements in 10 CFR 63.312(d). The Amargosa Valley survey measured how often residents consumed various locally produced food products. The calculated arithmetic mean [the method specified in 10 CFR 63.312(b)] annual consumption rates for various food types and corresponding standard deviations were then used as input parameters for sampling parameter values in ERMYN, assuming a lognormal distribution, as detailed in BSC (2005ab, Section 6.4.2, Table 6-21).

The NRC staff evaluated the adequacy of the applicant's technical bases and supporting data for the food and water consumption rates as detailed in BSC (2005ab). The requirements of

10 CFR 63.312(b) direct DOE to use projections based upon surveys of Amargosa Valley residents. The NRC staff's review finds that DOE food consumption rates are mean values based on the 1997 survey of Amargosa Valley residents in compliance with this requirement. The NRC staff's review of the survey finds the applicant acceptably generated survey data that provide a technical basis for deriving consumption rate input parameters for the biosphere calculations because the survey (i) included 195 of the reported 872 adult Amargosa residents, or 22 percent of the population; (ii) yielded useful responses from 187 of these contacts and provided explanations for the 8 responses that were eliminated from further analysis; and (iii) obtained information from residents on local food consumption frequency using a satisfactory survey approach.

The NRC staff reviewed the applicant's methods for calculating the annual consumption rates for locally produced food on the basis of the survey data, census information (to incorporate more recent Amargosa Valley population information), and national average daily intakes from the U.S. Department of Agriculture and finds these to be supported by data from credible sources. These methods are acceptable for supporting the dose calculations. The NRC staff finds the DOE values acceptable because they were developed on the basis of surveys of Amargosa Valley residents and are population averages of locally produced food consumption, as required by 10 CFR 63.312(b). The resulting values for annual food consumption rates appear to be low compared to values more commonly used from national food consumption surveys (e.g., NRC, 1992ae). This is because the applicant's values are population averages of individual consumption rates that apply specifically to the consumption of locally produced foods [detailed in BSC (2005ab, Figures 6-3 through 6-12)] rather than all food consumed. Additionally, a large number of Amargosa Valley residents who do not consume locally produced food weight the average consumption in the population to lower values than are found in national average food consumption rates. The low local food consumption rates also reflect the limited capacity of agricultural food production in the Amargosa Valley (LaPlante and Poor, 1997aa). The applicant's drinking water consumption rate of 2 L/d [0.528 gal/d] is consistent with the RMEI water consumption requirement of 10 CFR 63.312(d). The NRC staff concludes that the applicant has accounted for uncertainty and variability in parameter values and has provided technical bases for consumption rate parameter values, ranges, and probability distributions used in the ingestion exposure submodel in its performance assessment.

In summary, on the basis of the information discussed in this subsection, the NRC staff finds that the applicant has accounted for uncertainty and variability in parameter values and has provided the technical bases for parameter ranges, probability distributions, or bounding values used in the ingestion exposure submodel in its performance assessment. This complies with 10 CFR 63.114(a)(2) requirements to address the 10,000-year compliance period, carried through the period of geologic stability, as required by 10 CFR 63.114(b). The NRC staff also finds that the applicant has complied with 10 CFR 63.312(b) and 10 CFR 63.312(d) requirements with regard to the characteristics of the RMEI.

2.2.1.3.14.3.4 Assessment of Dosimetry

The DOE biosphere model uses dose coefficients from the Federal Guidance Report 13 (EPA, 1999aa), which uses tissue-weighting factors recommended in International Commission on Radiological Protection, Publication 60 (1991aa) to calculate effective dose from both internal and external radiation sources. In its TSPA modeling, DOE identified 28 primary radionuclides that were the primary contributors to dose to the RMEI using a radionuclide screening analysis (SNL, 2007ac; SAR Section 2.3.7.4.1.2). DOE then converted radionuclide intake or external exposure to dose using the dose coefficients for the 28 primary radionuclides.

DOE used dose coefficients for external exposure that are defined as the effective dose rate per unit radionuclide concentration in the soil. DOE also used dose coefficients for inhalation and ingestion as the committed effective dose per unit radionuclide intake by inhalation or ingestion.

DOE used dose coefficients for intake of radionuclides in the biosphere model for adults and for a total effective dose equivalent commitment period of 50 years. The biokinetic and dosimetric models used to develop these dose coefficients are based on a hypothetical average adult person with the anatomical and physiological characteristics the International Commission on Radiological Protection (1975aa) defined with further modifications, as described in Federal Guidance Report 13 (EPA, 1999aa). DOE used breathing rates in its biosphere model that are based on the more recent biometric results for adults from the respiratory tract model the International Commission on Radiological Protection (1994aa) developed, as discussed in SER Section 2.2.1.3.14.3.3. The NRC staff finds that the applicant's use of adult dose coefficients and breathing rates is consistent with the requirements in 10 CFR 63.312(e). The NRC staff also finds that DOE's dosimetry model is consistent with the requirements in 10 CFR 63.102(o) because DOE used current and appropriate scientific models and methodologies (recommended by the International Commission on Radiological Protection) to calculate the total effective dose equivalent.

For ingestion and inhalation, DOE chose dose coefficients for the chemical form of the radionuclide that resulted in the highest dose to avoid underestimating dose. But in a few cases, such as C-14, DOE chose dose coefficients that were consistent with the form that was being transported [i.e., the gaseous (carbon dioxide) and solid (particulate) forms have different dose coefficients]. Therefore, the NRC staff finds this approach to be acceptable because using these dose coefficients would not result in underestimating dose to the RMEI.

DOE calculated the total effective dose equivalent to the RMEI as the sum of the effective dose equivalent from external sources plus the committed dose equivalent from internal sources (i.e., sources either inhaled or ingested). The NRC staff finds this approach consistent with International Commission on Radiological Protection recommendations for calculating the total effective dose equivalent and, therefore, meets the applicable requirements in 10 CFR 63.102(o).

The NRC staff finds that the description that DOE provided in the SAR is adequate to fully assess the dosimetry models used in the TSPA. The NRC staff performed a detailed review of the dosimetry data for a selection of radionuclides, including Tc-99, C-14, I-129, Ra-226, Pu-239, Pu-240, Pu-242, and Np-237. The review also included comparing a sample of dose coefficients for other radionuclides that were used in the TSPA. No discrepancies were found between the dose coefficients included in the biosphere model report and those tabulated in Federal Guidance Report 13 (EPA, 1999aa). Therefore, the NRC staff finds that the dose coefficients DOE used meet the applicable requirements of 10 CFR 63.102(o) and 10 CFR 63.312(e) because they incorporate current and appropriate scientific models and methodologies for an adult receptor.

In summary, on the basis of the information discussed in this subsection, the NRC staff finds that the applicant has provided the technical bases for values used in the dosimetry model in its performance assessment in accordance with 10 CFR 63.114(a)(2) requirements to address the period after 10,000 years, carried through the period of geologic stability by the requirement of 10 CFR 63.114(b). The NRC staff finds that the dosimetry method uses current and appropriate scientific models and methodologies (recommended by the International Commission on Radiological Protection), as required by 10 CFR 63.102(o). NRC also finds that the applicant's dosimetry approach complies with the requirements of 10 CFR 63.312(e) because the dose coefficients were developed on the basis of adult metabolic and physiologic data for the RMEI, consistent with present knowledge.

2.2.1.3.14.3.5 Assessment of Integrated Biosphere Modeling Results

DOE biosphere modeling results were provided in SAR Section 2.3.10 and analyzed in greater detail in the Biosphere Model Report (SNL, 2007ac). The exposure pathways found to be the most risk significant in the DOE performance assessment varied depending on the particular radionuclide. The radionuclides and pathways that were most risk significant in the DOE TSPA calculations are summarized in SER Table 16-1 for the 10,000-year simulation period and SER Table 16-2 for the 1-million-year simulation period.

To validate the integrated biosphere model, DOE compared the calculation results for each environmental transport and exposure submodel of the ERMYN code with comparable calculation results from five other biosphere transport and exposure process-level models (SAR Section 2.3.10.5). DOE concluded that the results of the process-level calculations used in those other models were the same, or similar, to the results obtained using the biosphere model (SNL, 2007ac). To verify implementation, DOE compared the results of the biosphere model for representative radionuclides (Pu-239, Ra-226, Th-232, and C-14) with the results of spreadsheet calculations—on the basis of equations used in the biosphere mathematical model—and the results were identical (SNL, 2007ac). On the basis of the information reviewed, the NRC staff finds that the applicant’s model support calculations comply with 10 CFR 63.114(a)(7) because they compare biosphere model results with outputs of detailed process-level models and thereby provide technical bases for the biosphere models used in the performance assessment.

The NRC staff also performed confirmatory calculations to further assess the ERMYN model for a subset of the radionuclides that DOE’s performance assessment identified as the most risk significant (SER Tables 16-1 and 16-2; LaPlante, 2010aa). These confirmatory calculations were performed for the drinking water ingestion pathway of the groundwater exposure scenario. Drinking water ingestion was chosen because (i) it was identified as one of the most risk-significant pathways in the DOE performance assessment, (ii) the model is not complex and therefore could be executed efficiently, and (iii) the results of the confirmatory calculations could be directly compared to DOE results. The results of the confirmatory groundwater calculations were within 2 percent of the mean biosphere dose conversion factor results DOE reported in SAR Table 2.3.10-2, weighted by the DOE drinking water ingestion pathway fractions in SAR Table 2.3.10-11. Given that the DOE results are summary statistics of the model output, a difference of only 2 percent between NRC and DOE values shows them to be consistent. The NRC staff, therefore, finds that the applicant’s model support calculations comply with 10 CFR 63.114(a)(7) by comparing biosphere model results with outputs of detailed process-level models and thereby providing the technical bases for the biosphere models used in the performance assessment.

On the basis of the NRC staff’s review of the applicant’s comparison of results for the transport and exposure submodels of the ERMYN code with results from process-level models, the NRC staff finds the biosphere system description and model integration comply with 10 CFR 63.114. Confirmatory calculations the NRC staff performed demonstrate that the model integration in the TSPA complies with 10 CFR 63.114(a)(7) requirements.

In SAR Section 2.3.10.4.1, DOE describes how their performance assessment considered and evaluated alternative conceptual models for seven processes in the biosphere model abstraction. These processes are (i) radon release from soil, (ii) transfer of radionuclides to air from evaporative cooler operation, (iii) transfer of radionuclides to plants from direct deposition of irrigation water, (iv) transfer of airborne particulates to plants from direct deposition, (v) transfer of radionuclides to animal products from livestock inhalation, (vi) transfer of C-14 to crops, and (vii) environment-specific inhalation exposure. From this evaluation, DOE found that many of these alternative models produced results comparable to the DOE biosphere model. Other alternative conceptual models addressed unique pathways or processes that were not important contributors to dose, were less complex than the DOE biosphere model, or were not applicable to Yucca Mountain. Overall, DOE found the influence of applying the alternative conceptual models to biosphere model results were small and they found these models did not have a significant effect on predicted repository performance or uncertainty. The NRC staff reviewed the applicant's evaluation and finds that the applicant has demonstrated compliance with 10 CFR 63.114(a)(3) because the applicant considered alternative conceptual models and evaluated their effects on repository performance.

2.2.1.3.14.4 Evaluation Findings

The NRC staff has reviewed the SAR and other information submitted in support of the license application, which includes information required by 10 CFR 63.21(c)(1), and finds, with reasonable expectation, that DOE used an acceptable approach for the biosphere characteristics in calculating the dose to the RMEI using acceptable methods. On the basis of the NRC staff's review of the applicant's system description and model integration, the NRC staff finds that the applicant has demonstrated compliance with the requirements in 10 CFR 63.305, 63.311(b), and relevant requirements of 10 CFR 63.114, 63.312, and 63.342 regarding the biosphere characteristics. In particular, the NRC staff finds that

- The applicant provided the technical bases for including biosphere FEPs and has included all biosphere-related FEPs that could significantly change the magnitude or timing of the radiological exposures to the RMEI in the performance assessment in compliance with 10 CFR 63.114(a)(5).

- The included FEPs were consistent with present knowledge of the conditions in the region surrounding the Yucca Mountain site, in compliance with 10 CFR 63.305(a).

- The applicant's groundwater exposure scenario includes a RMEI adult who resides in the accessible environment above the highest concentration of radionuclides in the plume of contamination, in compliance with 10 CFR 63.312(a) and 10 CFR 63.312(e).

- The model includes all potential pathways of radionuclide transport and exposure, in compliance with 10 CFR 63.311(b), because the applicant has included pathways that are consistent with the NRC staff's understanding of the characteristics of the Yucca Mountain region, are commonly included in dose models and assessments, and are based on unique site-specific considerations (e.g., evaporative coolers, fish farming, wild game, and radon).

- The applicant's conceptual model is in compliance with biosphere and RMEI requirements in 10 CFR 63.305(b) because the review found no indication that DOE has projected changes in society, the biosphere (other than climate), human biology, or increases or decreases of human knowledge or technology.

- The applicant is in compliance with 10 CFR 63.305(c) in that it has allowed factors related to the geology, hydrology, and climate to vary based upon cautious but reasonable assumptions of the changes in these factors that could affect the Yucca Mountain disposal system during the period of geologic stability, which is consistent with the requirements for performance assessments specified in 10 CFR 63.342.

- The applicant's model is in compliance with the requirements of 10 CFR 63.305(d) because it includes the resuspension of dry soils by winds and the use of water evaporation to cool residences, making biosphere pathways consistent with arid or semiarid conditions.

- The applicant included in the performance assessment for the period of geologic stability those FEPs used for the initial 10,000-year period, consistent with specific limits, in compliance with 10 CFR 63.342(c).

On the basis of the NRC staff's review of the biosphere transport pathways in the applicant's performance assessment, the NRC staff finds the following for the plant uptake submodel, animal uptake submodel, fish uptake submodel, and air submodel

- The applicant has accounted for uncertainty and variability in parameter values and has provided the technical bases for parameter ranges, probability distributions, or bounding values used in the submodels in its performance assessment in accordance with the requirements in 10 CFR 63.114(a)(2).

- The applicant's performance assessment submodels are in accordance with the requirements in 10 CFR 63.114(b).

On the basis of the NRC staff's review of human exposures in the applicant's performance assessment, the NRC staff finds the following:

- The applicant has accounted for uncertainty and variability in parameter values and has provided the technical bases for parameter ranges, probability distributions, or bounding values used in the inhalation and ingestion exposure submodels in its performance assessment, in accordance with the requirements in 10 CFR 63.114(a)(2).

- The applicant's performance assessment for the period of time after 10,000 years is in accordance with the requirements in 10 CFR 63.114(b).

- Regarding the characteristics of the RMEI, the applicant has demonstrated compliance with the requirements of 10 CFR 63.312(b) and (d). The applicant used projections on the basis of surveys of Amargosa Valley residents. The RMEI has a diet and living style representative of people who now live in Amargosa Valley, Nevada. The RMEI drinks 2 L [0.528 gal] of water per day from wells drilled into groundwater at the location specified in 10 CFR 63.312(a).

For the dosimetry in the applicant's performance assessment, the NRC staff finds the following:

- The dosimetry method the applicant used follows current and appropriate scientific models and methodologies for calculating the total effective dose equivalent (recommended by the International Commission on Radiological Protection), as required by 10 CFR 63.102(o).

- The applicant's dosimetry approach is in compliance with the requirements of 10 CFR 63.312(e) because adult RMEI metabolic and physiologic considerations are consistent with present knowledge.

- The applicant has provided the technical bases for values used in the dosimetry model in its performance assessment in accordance with the requirements in 10 CFR 63.114(a)(2) and 10 CFR 63.114(b).

For the applicant's integrated biosphere modeling results, the NRC staff finds the following:

- The applicant's data are in compliance with 10 CFR 63.114(a)(1) and the uncertainty in the data is propagated through the model in compliance with 10 CFR 63.114(a)(2).

- The applicant considered and evaluated alternative conceptual models in compliance with 10 CFR 63.114(a)(3).

- The applicant's model support calculations are in compliance with 10 CFR 63.114(a)(7) because the applicant compared biosphere model results with outputs of detailed process-level models and thereby provided the technical bases for the biosphere models used in the performance assessment.

- Confirmatory calculations the NRC staff performed demonstrate that the TSPA output is in compliance with the requirements of 10 CFR 63.114(a)(7).

2.2.1.3.14.5 References

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CHAPTER 17

2.2.1.4.1 Demonstration of Compliance With the Postclosure Public Health and Environmental Standards (Individual Protection)

2.2.1.4.1.1 Introduction

By letter dated June 3, 2008, as supplemented on February 19, 2009, the U.S. Department of Energy (DOE) provided in its license application [Safety Analysis Report (SAR) Section 2.4.2 (DOE, 2008ab)] its basis for demonstrating compliance with the individual protection standards for the initial 10,000 years after closure and the period after 10,000 years up to 1 million years. DOE conducted an analysis, through its Total System Performance Assessment (TSPA) computer model, that evaluates the behavior of the high-level waste repository in terms of an annual dose due to potential releases from the repository. The performance assessment provides a method to evaluate the range of features (e.g., geologic rock types, waste package materials), events (e.g., earthquakes, igneous activity), and processes (e.g., corrosion of metal waste packages, sorption of radionuclides onto rock surfaces) that are relevant to the behavior of a repository at Yucca Mountain. Safety Evaluation Report (SER) Section 2.2.1.4.1 provides the U.S. Nuclear Regulatory Commission (NRC) staff's review of the DOE performance assessment used to demonstrate compliance with the individual protection standards.
In particular, the NRC staff's review evaluates whether (i) the performance assessment analysis includes the appropriate scenario classes [a set or combination of features, events, and processes (FEPs) that the performance assessment model uses to represent a class or type of scenario such as seismic activity], (ii) the representation of the scenario classes within the performance assessment is credible (e.g., the performance assessment results are consistent with the models, parameters, and assumptions that make up the performance assessment), and (iii) the annual dose the performance assessment estimates is less than the dose limits set by the regulations for the reasonably maximally exposed individual (RMEI).

The NRC staff's review of DOE's demonstration of compliance with the postclosure public health and environmental standards (Individual Protection) is related to the NRC staff's review of the model abstractions and scenario classes that makeup DOE's TSPA model, which is documented in SER Sections 2.2.1.2 and 2.2.1.3. The review in this section (Section 2.2.1.4.1) provides some discussion and references to the reviews documented in Sections 2.2.1.2 and 2.2.1.3 in support of the review of the performance assessment analyses for compliance with the Individual Protection standards. The NRC Staff's review in Section 2.2.1.4.1 includes references to Sections 2.2.1.2 and 2.2.1.3 to explain the relationship between the TSPA model and the calculation of individual dose.

2.2.1.4.1.2 Regulatory Requirements

The applicant is required to provide information specified in 10 CFR 63.21(c)(11) that relates to 10 CFR 63.113(b). The individual protection standard after permanent closure is specified at 10 CFR 63.311. The regulations also specify constraints for the performance assessment used to demonstrate compliance with the individual protection standard.

The regulations at 10 CFR 63.311 (Individual Protection Standard after Permanent Closure) require that the annual dose must not exceed 0.15 mSv/yr [15 mrem/yr] during the initial 10,000 years following disposal and not exceed 1.0 mSv/yr [100 mrem/yr] after 10,000 years up to 1 million years. The regulations at 10 CFR 63.113(b) specify that a performance assessment must be used to demonstrate compliance with the individual protection dose limit and set forth requirements for the performance assessment analysis at 10 CFR 63.114 and 63.342. The requirements for developing performance assessment analyses (e.g., consideration of FEPs included in the performance assessment, determination of event probabilities, and consideration of uncertainties) are evaluated in SER Sections 2.2.1.2 and 2.2.1.3. These previous sections evaluate DOE development of the analytic models used in the performance assessment analysis. The performance assessment analyses must also use the characteristics of the reference biosphere specified at 10 CFR 63.305 and characteristics of the RMEI specified at 10 CFR 63.312. Sections 2.2.1.3.12 and 2.2.1.3.14 evaluate compliance with the required characteristics of the reference biosphere and RMEI.

The requirements at 10 CFR 63.311 also specify how the performance assessment model is used to estimate the annual dose to the RMEI. In general, DOE is required to use the performance assessment to

- Demonstrate that the arithmetic mean (i.e., average) of the annual dose over the initial 10,000 years following disposal is no greater than 0.15 mSv [15 mrem] per 10 CFR 63.303 and 63.311(a)(1)

- Demonstrate that the arithmetic mean (i.e., average) of the annual dose after 10,000 years up to 1 million years (geological stability) is no greater than 1.0 mSv [100 mrem] per 10 CFR 63.303 and 63.311(a)(2)

The NRC staff's review of the SAR and supporting information follows the guidance in Yucca Mountain Review Plan (YMRP) (NRC, 2003aa) Section 2.2.1.4, as supplemented by additional guidance for the period beyond 10,000 years after permanent closure (NRC, 2009ab), for demonstrating compliance with the postclosure public health and environmental standards. The YMRP acceptance criteria address the following:

- Scenarios used in the calculation of the annual dose as a function of time are adequate

- Total System Performance Assessment provides a credible representation of repository performance

- Annual dose to the RMEI is adequately demonstrated

2.2.1.4.1.3 Technical Review

2.2.1.4.1.3.1 Introduction

The regulations in 10 CFR Part 63 require DOE to use a performance assessment method to demonstrate compliance with the dose limits for individual protection. The DOE performance assessment is implemented through its TSPA code. The TSPA code is used to represent the range of behavior of a Yucca Mountain repository, accounting for uncertainty in the FEPs that could affect the repository evolution over the compliance period. DOE developed its analysis of repository performance using distinct groupings of FEPs—referred to as "scenario" or "event" classes. In very general terms, there are two broad categories of scenario classes: nominal and disruptive. The nominal scenario class comprises those FEPs that are present under "normal" conditions (e.g., infiltration of water, corrosion of the waste package, release of radionuclides, transport of radionuclides in groundwater). The disruptive scenario class includes additional FEPs that account for the effects of specific events (i.e., seismic events, volcanic activity, fault movement) that disrupt or alter the repository performance differently from what the nominal scenario class portrays. In the DOE TSPA model, the nominal scenario class is considered part of the seismic ground motion modeling case so that the combined effects of waste package corrosion, which degrades the mechanical strength of the waste package, and mechanical damage of the waste package due to seismic ground motion are appropriately considered in the post-10,000-year period (SAR p. 2.4-36). (Note: For the initial 10,000 years, the nominal scenario class does not result in any dose, as detailed in SAR p. 2.4-62 and Figure 2.4-22a.) A key aspect of the disruptive scenario classes is the consideration of the probability or likelihood that the disruptive event will occur. By regulation, the annual dose is weighted by the probability of its occurrence.

The NRC staff reviewed SAR Sections 2.4.1 and 2.4.2 and the TSPA model files, including intermediate results provided as part of the license application. Additionally, the NRC staff's review in SER Section 2.2.1.4.1 relies on the NRC staff's findings, presented in SER Sections 2.2.1.1 through 2.2.1.3 of this volume, on the acceptability of the multiple barriers, scenarios, event probabilities, and model abstractions implemented in the DOE TSPA model. Specific SER sections, as applicable, are referenced in SER Section 2.2.1.4.1.

Consistent with the YMRP acceptance criteria, the NRC staff's review entails

- Determining that the probabilities and consequences of each of the scenario classes are appropriately included in the average annual dose (SER Section 2.2.1.4.1.3.2)

- Determining that the results of the performance assessment provide a credible representation of repository performance [e.g., the intermediate results, such as waste package failure, and release rates from the engineered barrier system (EBS), unsaturated, and saturated zones, are consistent with the model abstractions and the average annual dose; confirmatory calculations are consistent with the performance assessment results] (SER Section 2.2.1.4.1.3.3)

- Determining that the calculated average annual dose meets the regulatory limits and is statistically stable [e.g., increasing the number of simulations (statistical sample size) performed with the DOE TSPA is not expected to significantly change the average annual dose] (SER Section 2.2.1.4.1.3.4)

2.2.1.4.1.3.2 Scenarios Used in Calculation of Annual Dose

2.2.1.4.1.3.2.1 Summary of DOE Approach

DOE has identified three distinct event scenario classes (also referred to as event classes or scenario classes) that are included in its TSPA model to demonstrate compliance with the individual protection standard: (i) early failures, (ii) seismic events, and (iii) igneous events. DOE has used two modeling cases within each scenario class to represent specific aspects of the scenario. The early failure scenario class is composed of an early waste package failure modeling case and an early drip shield failure modeling case. The seismic scenario class is composed of a seismic ground motion modeling case and a seismic fault displacement modeling case. The igneous scenario class is composed of an igneous intrusion modeling case and a volcanic eruption modeling case.

The DOE average annual dose curve for individual protection (SAR Figure 2.4-18) is determined by summing the effects of all the scenario classes (i.e., early failure, seismic, and igneous). The annual doses attributed to each of the scenario classes are a direct result of the FEPs used to represent the scenario class and its probability of occurrence.

FEPs included in the scenario classes are reviewed in the NRC staff model abstraction review (SER Section 2.2.1.3). The NRC staff also reviewed and evaluated the FEPs that DOE considered and excluded from the performance assessment (see SER Section 2.2.1.2.1). As previously stated in the SER, the NRC staff finds the DOE approach for identifying the appropriate FEPs used to represent the scenario classes acceptable. The NRC staff's bases for acceptance are addressed in the previously identified SER Sections 2.2.1.2.1 and 2.2.1.3.1-2.2.1.3.14.

Scenario Class Probabilities

The DOE TSPA assessment incorporates the following three distinct event scenario classes: (i) the igneous activity scenario class, which has a very low annual probability [on the order of a 1 in 100 million chance of occurring per year, as outlined in CRWMS M&O (1996aa)]; (ii) the seismic scenario class, which typically results in numerous events occurring over 1 million years (according to SAR Section 2.4.2.1.6, p. 2.4-50, seismic events are expected to occur frequently; however, it is important to evaluate the timing and magnitude of seismic events); and (iii) the early failure scenario class, for which there is a low probability of occurrence for an individual waste package (SAR Section 2.4.2.1.6, p. 2.4-49). These three event scenario classes include the occurrence of nominal processes, whereas, the nominal scenario class represents repository behavior in which no events occur (i.e., no seismic events, no igneous events, and no early failure events; see SAR Section 2.4.2.1.3, pp. 2.4-30-31). The applicant has described how its approach to combine the scenarios to derive aggregated annual dose estimates is appropriate in that it tends to slightly overestimate dose by double counting waste packages potentially affected by different failure modes from the different scenarios (e.g., waste packages failed by a seismic event and an igneous event would be double counted; see SAR Section 2.4.2.1.7).

Igneous Scenario Class

Probability

The igneous scenario class is composed of an igneous intrusion modeling case and a volcanic eruption modeling case. The probability for the igneous intrusion modeling case is described in the DOE model as a Poisson process (a random process in which the events occur independently of one another), and intrusive events are distributed in time with a mean recurrence frequency of 1.7×10^{-8} per year, with a 5^{th} and 95^{th} percentile uncertainty spanning nearly two orders of magnitude, 7.4×10^{-10} to 5.5×10^{-8}, per year. DOE describes the probability of the volcanic eruption modeling case as a subset of the probability used for the igneous intrusion modeling case by using a conditional probability that an igneous intrusive event will also have an eruptive component that ejects waste into the atmosphere. The conditional probability is composed of (i) a conditional probability of 0.28 that an igneous intrusive event could have an eruptive component and (ii) a conditional probability of 0.2968 that the eruptive component of an igneous intrusive event could intersect the waste packages. The combination of these two probabilities results in a net conditional probability of 0.083 that an igneous intrusive event would also manifest a volcanic eruption that intersects waste packages in the repository footprint. Thus, the mean recurrence frequency for the volcanic eruption modeling case is 1.4 × 10^{-9} per year based on the mean recurrence frequency provided previously for the igneous intrusion modeling case of 1.7 × 10^{-8} per year (SAR p. 2.4-49).

Igneous Event Contribution to Average Annual Dose Curve

DOE evaluated the igneous intrusion modeling case and the volcanic eruption modeling case separately, assuming the entire intact or degraded repository inventory is available for release for each modeling case. A decrease in the repository inventory due to the occurrence of other scenarios is not considered. DOE stated this is a conservative assumption in that it does not underestimate the annual dose to the RMEI [SAR Section 2.4.2.1.7.4, p. 2.4-53].

The average annual dose from the igneous scenario class is the sum of the contributions from the igneous intrusion modeling case and the volcanic eruption modeling case. The average annual dose from the igneous intrusion modeling case is more than 99 percent of the average annual dose from the igneous scenario class (intrusion plus volcanic eruption) (SAR Figure 2.4-18).

The igneous intrusion modeling case is the second largest contributor to the overall average annual dose (i.e., summation of average annual dose from all scenario classes) in the 10,000-year period and the largest contributor to the overall average annual dose after 10,000 years (SAR Figure 2.4-18). In the 10,000-year period, Tc-99, Pu-239, Pu-240, and I-129 radioactive elements or radionuclides are the dominant contributors to the average annual dose. After 10,000 years, the dominant radionuclide contributors to the average annual dose are Pu-239, Pu-242, Np-237, and Ra-226 (SAR Figure 2.4-30).

Early Failure Scenario Class

Probability

The early failure scenario class includes two modeling cases: early drip shield failure and early waste package failure. Failure of the drip shield allows seepage water to contact a waste package. Failure of the waste package allows release of radionuclides out of the waste package by diffusion and/or advection. Early failure of either the drip shield or waste package is associated with undetected defects arising from manufacturing processes such as improper heat treatment, base metal selection flaws, improper weld filler material, and emplacement errors. DOE assumed that all of the waste packages under early failed drip shields would also be failed if contacted by seepage water, as described in SNL (2008ag, Section 6.4.1, p. 6.4-4). The probability of having a large number of drip shields and waste packages fail early due to undetected defects is very small. On average, the number of waste packages affected by early failure of drip shields and waste packages is less than 0.02 percent of the total number of waste packages (SAR Section 2.4.2.1.7.2, p. 2.4-52).

Early Failure Contribution to Average Annual Dose Curve

The early failure scenario class contributes on the order of 1 percent or less to the overall average annual dose curve (SAR Figure 2.4-18). In particular, the average annual dose for the early drip shield failure modeling case, which includes contributions from waste packages under drip conditions and unprotected by the drip shields against contact with seepage water, is below 1 × 10^{-5} mSv [0.001 mrem] for all times. The average annual dose for the early waste package failure modeling case is generally 10 times or more greater than the average annual dose for the early drip shield failure modeling case but still on the order of 100 times less than the overall average annual dose curve (SAR Figure 2.4-18).

Seismic Scenario Class

Probability

The seismic scenario class is composed of a seismic ground motion modeling case and a seismic fault displacement modeling case. The DOE model describes seismic ground motion events as a Poisson process, with events distributed in time with a mean recurrence frequency of 4.287×10^{-4} per year {corresponding to the frequency of events with a peak ground velocity exceeding 0.219 m/s [0.72 ft/s] (SAR Section 2.4.2.1.6, p. 2.4-50)}. This is the recurrence frequency of seismic ground motion events that could result in repository damage. Given a recurrence frequency of 4.287×10^{-4} per year, it is expected that seismic ground motion events of that magnitude could occur, on average, every 2,333 years. Thus, because it is expected that multiple seismic events will occur during the compliance period, DOE considered cumulative effects of seismic ground motion from multiple events.

The DOE model describes seismic fault displacement events as a Poisson process with events distributed in time, with a maximum mean recurrence frequency of 2×10^{-7} per year. Because multiple seismic fault displacement events that could affect the repository are expected to be sufficiently rare, DOE evaluated only the effects of single seismic fault displacement.

In both modeling cases for the seismic scenario class, the consequence of events that have a recurrence frequency between the highest recurrence frequency, which could cause repository damage, and the compliance limit of 1×10^{-8} per year are evaluated. The magnitude of an individual seismic event is determined through the use of a probabilistic seismic hazard analysis (PSHA) curve. The PSHA was developed primarily through the use of an expert elicitation process and is documented in CRWMS M&O (1998aa) (see also SER Sections 2.2.1.2.2.3.2 and 2.2.1.3.2 for further details on the NRC staff's evaluation).

Seismic Event Contribution to Average Annual Dose Curve

The average annual dose from the seismic scenario class is the sum of the contributions from the two seismic modeling cases: (i) the seismic ground motion modeling case, which addresses the potential for seismic events to damage waste packages and drip shields due to vibratory ground motion and (ii) the seismic fault displacement modeling case, which addresses the effects of fault displacement on waste packages and drip shields.

Nominal corrosion processes have the potential to alter the susceptibility of the waste package to damage during seismic ground motion events as the corrosion processes gradually weaken the mechanical strength of the waste package. Therefore, the seismic ground motion modeling case also includes both waste package degradation from the nominal processes (e.g., general corrosion) and seismic ground motion.

The average annual dose from the seismic ground motion modeling case is at least 10 times larger than the average annual dose from the seismic fault displacement modeling case over the entire 1-million-year compliance period (see SAR Figure 2.4-18). Although the average annual dose curve from the seismic ground motion modeling case also includes the effects from the nominal scenario class, the nominal scenario class contributes no more than 50 percent to the seismic ground motion modeling case average annual dose curve (compare SAR Figures 2.4-18 and 2.4-22).

The seismic ground motion modeling case is second only to the igneous intrusion modeling case in overall significance to the overall average annual dose curve. The seismic ground motion modeling case is the largest contributor to the overall average annual dose curve for the period after 1,500 years through 20,000 years. The overall average annual dose curve in either the initial 10,000 years or after 10,000 years is dominated by contributions from the seismic ground motion modeling case and the igneous intrusion modeling case (SAR p. 2.4-61).

2.2.1.4.1.3.2.2 NRC Staff's Review of Scenarios Used in Calculation of Annual Dose

The NRC staff has reviewed DOE analytic models and assumptions used in its TSPA analyses, as documented in the SAR and supporting documents, and finds that DOE inclusion of the annual dose from each of the scenario classes into the overall average annual dose curve is acceptable for the following reasons:

- The two modeling cases (i.e., igneous intrusion modeling case and seismic ground motion modeling case) that result in the greatest number of failed waste packages are the largest contributors to the overall average annual dose curve (see SER Section 2.2.1.4.1.3.3.1.1.2).

- DOE overall average annual dose curve appropriately includes the probabilities of occurrence for the scenario classes [the values for probability of igneous and seismic activity are appropriate and reflect the uncertainty for the occurrence of these events (see SER Sections 2.2.1.2.2.3.1 and 2.2.1.2.2.3.2)].

- The value for the probability of early failures reflects the uncertainty in potential defects regarding the drip shield and waste package (see SER Section 2.2.1.2.2.4).

- Model assumptions of the early failure scenario class tend to overestimate release consequences in the sense that no credit is given to early failed waste packages to impede contact of water or moisture with waste forms. Additionally, no credit is assigned to an early failed drip shield as a barrier against seepage, and the corresponding exposed waste packages are also given no credit to contain or impede contact of waste forms with seepage if located under seepage conditions (see SER Section 2.2.1.3.1.3.1.2 for the drip shield and SER Section 2.2.1.3.1.3.2.4 for the waste package).

- The consequences for an eruptive igneous event are appropriate and reflect conservative assumptions in certain key areas of uncertainty such as the amount of waste entrained in the tephra (volcanic ash) (see SER Section 2.2.1.3.13).

- The consequences for an intrusive igneous event are appropriate and reflect conservative assumptions in certain key areas of uncertainty, such as all waste packages and drip shields being rendered ineffective (see SER Section 2.2.1.3.10).

- The magnitude of seismic events leading to waste package damage is appropriate and reflects the material properties of the engineered barriers and design (see SER Section 2.2.1.3.2).

- The number of waste packages that can be potentially affected by fault displacement is appropriate and reflects the geologic setting of Yucca Mountain and the layout of the repository footprint (see SER Section 2.2.1.3.2.4).

2.2.1.4.1.3.3 Credible Representation of Repository Performance

This section of the SER documents the NRC staff's review to determine the credibility of the representation of repository performance in the DOE TSPA. In particular, the NRC staff evaluates the consistency of the characteristics of repository performance in the TSPA (e.g., number of waste packages failed, transport of radionuclides in the geosphere, and scenario probabilities) with the overall dose estimated by the TSPA. The focus of the NRC staff review is on those aspects of repository performance that have the most significance to risk (i.e., the probability weighted dose estimate). The NRC staff's review of individual components of the DOE TSPA (i.e., model abstractions; FEPs included in the TSPA; scenario probabilities; and barrier capabilities) is documented in SER Sections 2.2.1.1 through 2.2.1.3 of this Postclosure Volume of the SER.

The NRC staff reviewed the TSPA documentation in SAR Volume 2 and in the TSPA GoldSim computer model and associated computer files (including intermediate results saved in the GoldSim output files). The NRC staff's review of the TSPA analyses considered how the collection of FEPs that are included in the TSPA model represent a credible characterization of the repository. The NRC staff's review approach includes the NRC staff's confirmatory calculation containing the attributes of the DOE TSPA calculation that were considered important for estimating the annual dose to the RMEI. Identification of the important attributes for performance is based on the NRC staff's review of the capabilities of the barriers important to waste isolation (SER Section 2.2.1.1) and the model abstractions in the TSPA code (SER Section 2.2.1.3) and the NRC staff's independent analysis with its performance assessment model, as outlined in CNWRA and NRC (2008aa) and NRC (2005aa, Appendix D).

DOE has used two modeling cases for each of the three scenario classes [for igneous (intrusive and eruption), for seismic (ground motion and fault displacement), and for early failure (drip shield and waste package)] to estimate overall performance of the Yucca Mountain repository (see Table 17-1). Only one modeling case (volcanic eruptive) releases radionuclides directly to the atmosphere via volcanic ash. The other five modeling cases (seismic ground motion, seismic fault displacement, igneous intrusion, early waste package failure, and early drip shield failure) release radionuclides primarily through groundwater movement. The NRC staff's review of the DOE TSPA calculation related to groundwater releases is provided in SER Section 2.2.1.4.1.3.3.1, and the NRC staff's review of the DOE TSPA calculation for the volcanic eruption modeling case is provided in SER Section 2.2.1.4.1.3.3.2.

Table 17-1. Scenario Classes and Modeling Cases Included in the DOE TSPA

Scenario Class	Modeling Case	Transport Pathway*
Early failure	Drip shield	Groundwater
	Waste package	Groundwater
Seismic	Ground motion	Groundwater
	Fault displacement	Groundwater
Igneous	Intrusive	Groundwater
	Eruption	Atmospheric (volcanic ash)

*Transport pathway indicates the primary pathway for radionuclides to be transported away from the repository to the RMEI location specified in 10 CFR 63.312.

2.2.1.4.1.3.3.1 DOE TSPA Calculation Related to Groundwater Releases

2.2.1.4.1.3.3.1.1 Summary of DOE's Approach in TSPA Related to Groundwater Releases

The modeling cases associated with groundwater releases are described by tracking the water through the system. For example, water could infiltrate the top of the mountain and move downward to the repository; after waste packages are breached and radionuclide releases occur, water could transport radionuclides through the unsaturated zone and then through the saturated zone to the location of the RMEI. In general, the description of the groundwater releases is based on the following repository performance characteristics:

- Seepage (flux of water dripping into drifts)
- Damage to engineered barriers (drip shield and waste package)
- Flux of water entering the waste packages
- Release of radionuclides from the waste package
- Transport of radionuclides in the unsaturated zone
- Transport of radionuclides in the saturated zone
- Annual dose to the RMEI

The volcanic eruption modeling case evaluates the release of radionuclides via volcanic ash deposited on the ground. The volcanic eruption modeling case is evaluated separately (see "Description and Understanding of TSPA Calculation Related to Releases from a Volcanic Eruption Event" later in this section) from the modeling cases that involve radionuclide release through the groundwater pathway.

2.2.1.4.1.3.3.1.1.1 Summary of DOE's TSPA for Seepage of Water Into Drifts

The flux of water reaching the drifts (i.e., drift seepage) is originally derived from rainfall over the mountain. Two important metrics for performance for the upper natural barrier are seepage flux (the amount of liquid water entering the repository drifts) and seepage fraction (number of waste package locations with dripping water). The latter is the fraction of the repository area where seepage occurs (the seeping environment); the remainder of the area would not receive seepage. Seepage, or dripping water, has the potential to fall onto the drip shields and later contact the waste packages after the drip shields degrade sufficiently to allow water to pass through the drip shield.

Precipitation to Deep Percolation

DOE divided the first 10,000 years into three periods: present day (0–600 years), monsoonal (600–2,000 years), and glacial transition (2,000–10,000 years) climates (SAR Tables 2.3-1–2.3-4; DOE, 2008ab). The glacial transition climate spans 80 percent of the first 10,000 years and has the most significant impact on the performance of the repository over this initial period. For the glacial transition period, the applicant calculated an average precipitation of 296.7 mm/yr [11.7 in/yr] in the repository footprint, as described in DOE (2010ai, Enclosure 1, Table 1). Processes including runoff of water from hillsides, evaporation, and lateral diversion of water caused by the Paintbrush Tuff nonwelded rock layer alter the amount of rainfall that eventually ends up as deep percolation (the amount of water reaching the repository level). DOE estimated an average deep percolation of 21.74 mm/yr [0.86 in/yr] at the repository horizon, as detailed in DOE (2010ai, Enclosure 1, Table 1) at the repository footprint for the initial 10,000 years. Thus, approximately 7 percent of the rainfall ends up as deep percolation at the repository footprint. (SER Sections 2.2.1.3.5 and 2.2.1.3.6 provide further details on climate and infiltration.)

For the post-10,000-year period, 10 CFR 63.342(c)(2) specifies a time-independent flux for deep percolation as a range of 10 to 100 mm/yr [0.39 and 3.9 in/yr] using a truncated lognormal distribution, which results in an arithmetic mean of 37 mm/yr [1.5 in/yr] for the deep percolation. In the license application, DOE used a log-uniform distribution with an arithmetic mean of 32 mm/yr [1.3 in/yr] and a range between 13 and 64 mm/yr [0.51 and 2.5 in/yr] (SAR Section 2.3.2.3.5.1), on the basis of the proposed regulation. The final regulation specifies a 16 percent higher average deep percolation than the value DOE used in the SAR (see SER Section 2.2.1.3.6 for further discussion on the reason for the different values used for deep percolation). Using this higher deep percolation for a bounding calculation, DOE estimated a corresponding increase in the 1-million-year RMEI average annual dose from 0.02 mSv/yr to 0.023 mSv/yr [2.0 mrem/yr to 2.3 mrem/yr], which remains below the regulatory limit of 1 mSv/yr [100 mrem/yr], as detailed in DOE (2009cb, Enclosure 6).

Deep Percolation to Seepage

The TSPA model includes a number of factors, such as focusing or diverging of flow at the repository footprint, vapor barrier surrounding a drift when the drift temperature is above the boiling point of water, different waste package types, capillary diversion, drift degradation (prevalent in the seismic cases) in estimating seepage fraction, and drift seepage. Seepage is set to zero if the drift wall temperature exceeds 100 °C [212 °F] (SAR Section 2.3.3.4.1.1). The period of time when drift wall temperatures exceed 100 °C [212 °F] is generally limited to the first 2,000 years, as the heat generated by radionuclide decay decreases.

The longevity of the drip shield and waste package limits the significance of the seepage at early times, especially during the initial 10,000 years. The uncertainty in estimating the longevity of the drip shield and waste package are evaluated in SER Sections 2.2.1.3.1 and 2.2.1.3.2.

Over the repository footprint, seepage flux is approximately 10 percent of the deep percolation for intact drifts and can increase up to 49 percent for degraded drifts as the applicant predicted through the seismic ground motion modeling case, outlined in DOE (2010ai, Enclosure 5). For the igneous intrusive scenario, all of the percolating flux enters the drift at all locations, as described in DOE (2009ct, Enclosure 7). Table 17-2 provides the DOE values for the seepage fraction and the average seepage rate over the repository footprint (i.e., averaged over both seeping and nonseeping environments). As the seepage rate increases, the number of locations where dripping occurs over the repository footprint (seepage fraction) also increases.

Table 17-2. DOE Mean Values for the Repository Average Seepage Rate Into Drifts*

Time Period	Nominal/Early Failure	Seismic Ground Motion	Igneous Intrusive
Seepage from 2,000 to 10,000 years	2.0 (mm/yr)	2.3 (mm/yr)	21.7 (mm/yr)
Seepage fraction from 2,000 to 10,000 years	31%	31%	100%
Seepage after 10,000 years	3.4 (mm/yr)	15.5 (mm/yr)	31.8 (mm/yr)
Seepage fraction after 10,000 years	40%	69%	100%

*See SER Section 2.2.1.3.6 for further information.

2.2.1.4.1.3.3.1.1.2 Summary of DOE's TSPA for Damage to Engineered Barriers (Drip Shield and Waste Package)

The drip shield and the waste package are two important components of the EBS (SER Sections 2.2.1.3.1 and 2.2.1.3.2 evaluate behavior of the drip shield and waste package). The drip shield degrades gradually over time (from general corrosion) or at specific times from large seismic events or igneous events. From the distributions considered in the TSPA model to represent corrosion rates of Titanium Grade 7, time to failure by general corrosion of the drip shield was computed to range from 260,000 to 340,000 years (SAR Figures 2.1-8 and 2.4-24). Seismic ground motion can collapse the drip shield by mechanical failure due to static and dynamic loads caused by rockfall and ground motion. As illustrated in SAR Figure 2.1-11, drip shield collapse due to seismic ground motion occurs primarily between 25,000 and 350,000 years, with the vast majority of the failures occurring between 200,000 and 300,000 years (see SAR Figure 2.4-24). Drip shields are assumed to fail whenever an intrusive igneous event occurs, when a fault displacement event breaches the waste package, or when significant general corrosion occurs. Once the drip shield is failed, seepage water that enters the drifts can contact the surface of the waste package.

The damage mechanisms leading to waste package failure are "crack" and "patch" (holes) failure. Within the DOE TSPA model, crack and patch failures of the waste package are treated separately because of differences in how water may enter a breached waste package. DOE assumed that seepage water cannot freely flow through cracks on the waste package because of the small size of the cracks. Because of processes such as general corrosion, or ruptures and punctures of the waste package, patch failures represent significantly larger openings and seepage water is assumed to enter the waste package through these holes or openings. The waste packages with large or patch openings could allow release of radionuclides carried by flowing water (i.e., advective release) and diffusion, while those with crack openings could only allow release by diffusion.

The five modeling cases associated with groundwater releases have distinct characteristics for the timing and extent of waste package failure. Assumed to occur at the time of closure, the early failure scenario class consists of two modeling cases: drip shield early failure and waste package early failure. The early failure scenario has, in probabilistic terms, on average, less than one waste package and one drip shield failing (see SER Section 2.2.1.3.1.3.2.4 for early waste package failure and SER Section 2.2.1.3.1.3.1.2 for drip shield failure). The igneous intrusion modeling case assumes all waste packages fail at the time of the event and lead to release to the water pathway (see SER Section 2.2.1.3.10 for further details). The seismic scenario class consists of a ground motion modeling case, where the waste packages are damaged by seismic ground motion, and a fault displacement modeling case, where displacement along a fault may damage the waste packages that lie along the fault. The seismic fault displacement modeling case has, on average, tens of waste packages failing (see SER Section 2.2.1.3.2 for further details). The seismic ground motion modeling case contains a range of waste package failures—typically initial damage is primarily due to cracks in the waste package from ground motion, and, later in time (e.g., after 100,000 years), general corrosion, ruptures and punctures, and further cracking damages the waste packages.

Table 17-3 provides the cumulative number of waste package failures accounted for in the seismic ground motion modeling case for selected times (i.e., 10,000; 100,000; 400,000; and 800,000 years) to provide some perspective on the time-dependent nature of waste package failure. The values in Table 17-3 for the seismic ground motion modeling case were taken from DOE (DOE, 2009bj, Enclosure 1, Figures 9, 10, 13, and 14) and SAR Figures 2.1-12 a and c.

Table 17-3. Cumulative Number of CSNF* and CDSP† Waste Packages Breached for the Seismic Ground Motion and Igneous Intrusion Modeling Cases in DOE's TSPA					
Process		**10,000 Years**	**100,000 Years**	**400,000 Years**	**800,000 Years**
Seismic Ground Motion	All failure types (cracks and patches)	CSNF 1.6 CDSP 34.2	CSNF 20.5 CDSP 1,024.8	CSNF 739.2 CDSP 1,366.4	CSNF 3,531.6 CDSP 2,049.6
	Ruptures and punctures (patches)	CSNF 0 CDSP 0.3	CSNF 0 CDSP 0.7	CSNF 24.6 CDSP 13.7	CSNF 82.1 CDSP 34.2
	General corrosion (patches)	CSNF 0 CDSP 0	CSNF 0 CDSP 0	CSNF 0.3 CDSP 4.8	CSNF 328.5 CDSP 239.1
Igneous Intrusion‡	All failures are patch failures	CSNF 1.4 CDSP 0.58	CSNF 14 CDSP 5.8	CSNF 55.9 CDSP 23.2	CSNF 111.7 CDSP 46.5
*Repository contains 8,213 commercial spent nuclear fuel (CSNF) waste packages. †Repository contains 3,416 codisposal (CDSP) waste packages. ‡Igneous intrusion values calculated assuming all waste packages failed weighted by the probability of occurrence of an igneous event on or before the given time (i.e., annual probability for the event, of 1.7×10^{-8} per year, multiplied by the time period).					

These values are weighted by the probability for the seismic events to occur. In the first 10,000 years, a small number of waste packages fail in the seismic ground motion modeling case due to rare but potentially damaging earthquakes. The majority of the failed waste packages are codisposal packages (CDSP). These packages are not equivalent to the more robustly designed transportation, aging, and disposal canisters that contain the commercial spent nuclear fuel (CSNF) waste packages. Most of the initial damage is attributed to cracks that are small enough to prevent seepage water from entering the waste package. At 100,000 years, the number of failed codisposal waste packages due to cracks is 1,025 (30 percent of the 3,416 CDSP waste packages in the repository) and the number of failed commercial spent nuclear fuel waste packages is 20 (less than 1 percent of the 8,213 CSNF waste packages in the repository). At 800,000 years, approximately 40 percent of the CSNF and 50 percent of the CDSP waste packages fail due to cracks. Some of these cracks are from seismically induced stress corrosion cracks on the waste package surface. Others are due to stress corrosion cracks in the closure welds—considered a general corrosion process. Waste packages start to fail by general corrosion patches at around 500,000 years, and 239 (approximately 7 percent) of the CDSP waste packages and 328 (approximately 4 percent) of the commercial spent nuclear fuel waste packages have at least one failed patch due to general corrosion by 800,000 years, as described in DOE (2009bj, Enclosure 1, Response Number 1, Figures 9 and 10). Waste package failure due to ruptures and punctures is limited: approximately 1 percent of the CSNF and CDSP waste packages failed after 800,000 years, as shown in DOE (2009bj, Enclosure 1, Response Number 1, Figures 13 and 14).

The majority of the waste package failures are associated with the seismic ground motion modeling case, which includes the nominal processes such as general corrosion, and the igneous intrusion modeling case. In the DOE TSPA model, the seismic ground motion and igneous intrusion modeling cases contribute most to the overall average annual dose curve and are generally more than a factor of 10 greater than the other modeling cases (SAR Figure 2.4-18). The NRC staff evaluation in SER Sections 2.2.1.3.1 and 2.2.1.3.2 finds DOE representation for the timing and extent of waste package failures acceptable. Therefore, the NRC staff's detailed review of DOE groundwater releases focuses on these two modeling cases, which are the dominant contributors to the overall average annual dose curve (SAR Figure 2.4-18).

2.2.1.4.1.3.3.1.1.3 Summary of DOE's TSPA for Seepage of Water Into Waste Packages

In the TSPA model, two conditions are required for seepage water to enter a waste package: (i) drip shield failure must occur to allow water to contact the waste package outer barrier and (ii) the waste package outer barrier must be breached by patches (it is assumed that seepage or dripping water cannot flow into waste package cracks due to the small opening of cracks; therefore, seepage water enters the waste package only through patch failures). When these required conditions are met, water flow through the waste package is modeled as a quasi-steady state, where water flux into the waste package is equal to water flux out.

The waste package surface is divided into a large number of patches with each patch having distinct properties that can affect the corrosion rate of the patch. The extent of waste package degradation (i.e., number of patch failures on the waste package) determines the quantity of water entering the waste package. The waste package outer barrier is considered unable to divert water when a mean value of approximately 62 patches fail (62 patches comprises approximately 4 percent of the total surface area of the waste package), at which point water flow through the waste package equals the incoming seepage rate. When fewer than 62 patches fail, water flux through the waste package is linearly related to the number of patches that fail. Generally, a single patch failure allows 1/62 of the seepage flux to pass through the waste package. Because of uncertainty incorporated into the submodel, this value can range from 0 to 2.4 times the 1/62 value.

The waste package patch failures in the seismic ground motion modeling case are of limited extent (i.e., the waste package surface area has a limited number of failed patches). Approximately 1 percent of the waste package surface area is breached by general corrosion after 1 million years, as described in DOE (2009bj, Enclosure 1, Response Number 1, Figures 11 and 12). Patch failure by ruptures and punctures compromises approximately 0.2 percent of the waste package surface by 1 million years, as outlined in DOE (2009bj, Enclosure 1, Response Number 1, Figures 15 and 16). Thus, the compromised area remains limited for the 1-million-year period after patch failure occurs due to general corrosion, ruptures, or punctures. Additionally, the number of waste package patch failures is limited over the 1-million-year time period: 10 percent of waste package failures are from general corrosion and approximately 3 percent of waste package failures from ruptures and punctures after 1 million years. These failures occur primarily at long times [e.g., after 400,000 years, as shown in DOE (2009bj, Enclosure 1, Response Number 1, Figures 9, 10, 13, and 14)].

For the igneous intrusion modeling case, the drip shield and waste package are assumed to be ineffective barriers against seepage in the TSPA model (i.e., no credit to decrease seepage is given to the drip shield or waste package after the time at which the event occurs). Thus, DOE assumes all drip shields and waste packages are failed after the igneous intrusion event occurs [SAR Volume 2, page 2.4-42 and DOE (2009ct, Enclosure 8, p. 13)].

2.2.1.4.1.3.3.1.1.4 Summary of DOE's TSPA for Release of Radionuclides From the Waste Package

Waste form degradation and subsequent radionuclide release cannot occur prior to waste package breach and/or failure. Assuming waste package failure, radionuclide release may be advective if seepage water enters the waste package; otherwise, the release will be a diffusive release.

DOE explained that seepage water does not flow through crack failures; thus, the radionuclide release from cracks in the waste package is controlled by radionuclide diffusion in an assumed continuum of aqueous pathways through the cracks. DOE described patch failure as a more extensive damage mechanism of the waste package surface due to general corrosion or waste package ruptures and punctures driven by seismic events and mechanical interactions with drip shields or other waste packages. Radionuclide release through damaged patches is assumed to be diffusive if seepage does not contact the waste package, which could occur in the DOE model if the drip shield is not breached, or if the waste package is under non-drip conditions. If, on the other hand, the waste package is under drip conditions, the drip shield is failed, and the waste package is breached by patches (by corrosion or processes driven by seismic events), then radionuclides are released from the waste package by flowing water (i.e., advective release) and diffusion. Advective release is effective also in the igneous intrusion case, in which all waste packages fail completely and seepage is assumed to contact all waste packages (SAR Section 2.3.7.12).

In general, diffusive and advective release of radionuclides from a waste package will be affected by the size of the openings, degradation rate of the waste, solubility limits, sorption onto corrosion products, and the presence of colloids (SAR Section 2.3.7.13). The significance of these features and processes can vary for specific radionuclides, as described next.

Size of the Openings

The overall surface area of the crack and patch openings directly affects radionuclide diffusion out of the waste package (more surface area results in more release). Additionally, the overall surface area of the patch openings can affect the amount of water entering the waste package and, thus, the amount of dissolved radionuclides released from the waste package in the advective or flowing water. Once an average of 4 percent of the waste package surface is failed, due to patches, DOE assumes that the waste package no longer limits the amount of seepage water that enters the waste package. Thus, all seepage water is assumed to enter the waste package when 4 percent or more of the waste package surface area is failed (SAR page 2.4-156).

Degradation Rate of Waste

Radionuclides cannot leave the waste package faster than the waste degrades. Generally, the degradation rates used in TSPA for CSNF result in somewhat short times (e.g., hundreds to thousands of years) for the waste form to significantly degrade, as outlined in DOE (2009an, Enclosure 5, Table 1.1-1); therefore, the degradation rate only affects those radionuclides that are not limited by other release constraints inside the waste package (e.g., Tc-99 and I-129 are not solubility limited and are not sorbed or attached onto corrosion products). For CDSP waste packages, the glass waste form can degrade much slower than CSNF (e.g., thousands to millions of years for the glass waste form to significantly degrade versus hundreds to thousands of years for CSNF); however, the defense spent nuclear fuel waste form is assumed to instantly degrade, as described in DOE (2009an, Enclosure 5, p. 6).

Solubility Limit

Some radionuclides have a solubility limit—a function of the properties of the radionuclide and the water chemistry inside the waste package—that controls the amount that can be dissolved in water. Radionuclides such as plutonium (e.g., Pu-242) and neptunium (e.g., Np-237) have the potential for low release rates due to solubility limits (see SER Section 2.2.1.3.4.3.3).

Corrosion Products

The TSPA includes a process by which certain radionuclides attach onto corrosion products within the waste package, and, thus, release from the waste package is delayed. This is especially effective for a radionuclide such as Np-237 that is somewhat soluble and attaches onto corrosion products, as described in DOE (2009an, Enclosure 5, p. 22).

Colloids

Colloids are tiny particles that remain suspended in water and are thus able to move with the water and facilitate the transport of certain radionuclides. Colloids can facilitate release of radionuclides out of the waste package; radionuclides sorbed or attached onto irreversible colloids are not affected by solubility limits and stationary corrosion products. Irreversible colloids can also sequester radionuclides by becoming unstable and settle out (see SER Section 2.2.1.3.4.3.4). DOE stated that the contribution to annual dose from irreversible colloids is small [i.e., contribution from irreversible colloids never exceeds 30 percent, as shown in DOE (2009an, Enclosure 5, pp. 24–25)]. The contribution to annual dose from a radionuclide such as Pu-242 will be mainly from aqueous releases, which are composed of both dissolved radionuclides and reversible colloids (e.g., SAR p. 2.4-93 and SAR Figure 2.4-73).

Release rates of radionuclides from an individual waste package are dependent on the type of radionuclide. High mobility characterizes the soluble, nonsorbing radionuclides (e.g., Tc-99, I-129, Cl-36, Se-79), which may result in nearly complete release of the inventory of the high-mobility radionuclides from the EBS over the 1-million-year compliance period (SAR Figure 2.1-24). Much lower mobility characterizes the relatively insoluble, sorbing nuclides (e.g., Np-237, Pu-242). For the concentration-limited radionuclides (e.g., Pu-242 and Np-237), DOE explained that releases will be significantly lower than the release rates for soluble radionuclides (e.g., Tc-99) and will increase as water flow into the waste packages increases. For example, as corrosion patch area increases in size over time, more water may enter the waste package (SAR p. 2.4-63). At the end of the 1-million-year compliance period, approximately 0.1 percent of the Np-237 inventory has been released from the EBS with the majority of the release occurring over the later portion of the compliance period (SAR Figure 2.1-25).

2.2.1.4.1.3.3.1.1.5 Summary of DOE's TSPA for Transport of Radionuclides in the Unsaturated and Saturated Zones

Transport of radionuclides through the unsaturated zone is affected by the following processes: (i) relatively fast fracture flow versus slow flow in the porous rock matrix, (ii) radionuclides that sorb onto mineral surfaces, and (iii) colloid-facilitated transport of radionuclides.

Transport of radionuclides can depend on whether water flow occurs principally in fractures or in porous media. Flow in fractures is conceptualized as being relatively fast because the effective porosity is relatively small [average estimated value of 0.001 (SAR Table 2.3.9-4)]. Conversely, flow in porous media is conceptualized as relatively slow because the effective flow porosity is relatively high [average estimated value of 0.18 (SAR Table 2.3.9-4)]. Additionally, given the limited surface area for fracture surfaces as compared to rock pores, radionuclides can be significantly delayed by sorption to mineral surfaces.

Colloids can decrease the transport times of strongly sorbing radionuclides, such as plutonium and americium, by permanently attaching onto colloids that increase the transport velocity relative to sorbing radionuclides transported as a dissolved species in water. This colloid attachment can also occur reversibly when radionuclides temporarily attach to or detach from colloids as they move through the system. Generally, most of the release of Pu-242 from the unsaturated and saturated zone is via dissolved plutonium and plutonium reversibly associated with colloids (referred to as “aqueous” release in SAR Figure 2.4-108). Limited release of Pu-242 is associated with irreversible colloids, whereby Pu-242 permanently attaches onto the colloid.

Transport of Radionuclides in the Unsaturated Zone

Transport of radionuclides in the unsaturated zone depends to some extent on the location from which they are released. In the northern area of the repository, water is expected to move principally within fractures. Average travel times from the repository to the saturated zone from the northern area of the repository are on the order of 5 to 100 years for nonsorbing solutes without decay or matrix diffusion (SAR Figure 2.3.8-36). Conversely, in the southern repository area, the Calico Hills nonwelded tuff unit has higher matrix permeability that can accommodate flow almost entirely within the rock matrix (porous flow). Average travel times from the southern repository area to the saturated zone are on the order of 500 to 5,000 years (SAR Figure 2.3.8-36). DOE travel times were determined using the Glacial Transition 10^{th} percentile infiltration map, which is the most likely infiltration map of the four maps considered for the Glacial Transition period (see SAR p. 2.3.2-70).

For sorbing radionuclides, travel times depend on the radionuclide-specific sorption coefficient. More strongly sorbing aqueous species, such as Pu-242, have transport times on the order of hundreds of thousands of years and longer in the southern area. Some radionuclides that are dominant contributors to the total inventory are significantly delayed before reaching the water table due to sorption of radionuclides onto the rock matrix that exists in the southern area (e.g., Cs-137, Sr-90, Pu-239, Pu-240, Am-241, Am-243) [SAR Section 2.3.8].

Table 17-4 shows how the combined processes affect flow of different radionuclides through the unsaturated zone by providing representative transport times for nonsorbing Tc-99, moderately sorbing Np-237, strongly sorbing Pu-242, and Pu-242 attached to colloids.

Transport of Radionuclides in the Saturated Zone

In the DOE TSPA, radionuclides released from a Yucca Mountain repository would eventually enter the saturated zone within the fractured volcanic tuffs of the Crater Flat group. Transport away from the repository area would occur through permeable flowing fracture networks in the volcanic aquifer system for more than 10 km [6.2 mi] and transition to a valley fill alluvial flow system for the last few kilometers before reaching the regulatory compliance boundary approximately 18 km [11.2 mi] from the southern boundary of the repository footprint. The exact location of the volcanic rock–alluvium contact is uncertain and is treated stochastically in the

Table 17-4. Radionuclide Transport Times in the Unsaturated Zone for the Northern and Southern Repository Areas From DOE Breakthrough Curves		
	Transport Time* for Release in Northern Repository Area	**Transport Time* for Release in Southern Repository Area**
Tc-99	10 years	1,000 years
Np-237	10 years	10,000 years
Pu-242	30 years	>1 million years
Pu-242 irreversible colloids	100 years	1,000 years
*Transport times reflect approximate arrival for 50 percent of peak concentration for a model case with point releases at representative locations in northern and southern model areas, representative parameter values, and Glacial Transition 10th percentile infiltration map. Estimated from SAR Figures 2.3.8-43 for Tc-99, 2.3.8-44(b) for Np-237, 2.3.8-47(a) for Pu-242, and 2.3.8-48 for Pu-242 irreversible colloids.		

saturated zone transport abstraction model using an alluvium uncertainty zone. The fracture flow path for the volcanic tuff is conceptualized as being relatively fast because the effective porosity is relatively small [average estimated value of 0.001 (SAR Table 2.3.9-4)]. Flow in the alluvial portion of the flow system is conceptualized as relatively slow because the effective flow porosity is relatively high [average estimated value of 0.18 (SAR p. 2.3.9-59)]. Overall, the transport time for nonsorbing radionuclides ranges from about 10 years to several thousand years (SAR p. 2.3.9-9). Sorbing radionuclides can be significantly delayed by sorption to alluvium mineral grains in which case transport times for strongly sorbing radionuclides generally exceed 10,000 years (SAR p. 2.3.9-9). Table 17-5 provides transport times for select radionuclides representing a range of sorption behavior.

Table 17-5. Summary of DOE Median Transport Times in the Saturated Zone Under Glacial-Transition Climate State		
Species	**Range of Median Transport Times (Years)**	**Median Transport Time Among All Realizations**
C-14, Tc-99, I-129 (aqueous, nonsorbing)	10 to 20,000	200
Reversible colloids: plutonium	2,000 to 1 million	100,000
Neptunium	100 to 600,000	4,000
Irreversible colloids: plutonium	100 to 600,000	5,000
Estimated from SAR Figures 2.3.9-16 for Tc-99; 2.3.9-45 for Np-237; 2.3.9-46 for Pu reversibly attached on colloids; and 2.3.9-47 for Pu irreversibly attached on colloids.		

2.2.1.4.1.3.3.1.1.6 Summary of DOE's TSPA for Annual Dose to the RMEI

Following postclosure EBS release and groundwater radionuclide transport, the DOE TSPA model executes the biosphere model abstraction to calculate biosphere radionuclide transport and the annual dose to the RMEI. The exposure scenarios implemented in the DOE TSPA model (i.e., groundwater, volcanic ash) calculate annual dose to an individual adult member of a hypothetical farming community located 18 km [11.2 mi] south of the potential repository along the path of groundwater flow. Exposure pathways in the DOE biosphere model are based on assumptions about residential and agricultural uses of the water and indoor and outdoor activities. These pathways include ingestion, inhalation, and direct exposure to radionuclides deposited to soil from irrigation. Ingestion pathways include drinking contaminated water, eating crops irrigated with contaminated water, eating food products produced from livestock raised on contaminated feed and water, eating farmed fish raised in contaminated water, and inadvertently ingesting soil. Inhalation pathways include breathing resuspended soil, aerosols from evaporative coolers, and radon gas and its decay products resulting from the high-level radioactive waste (SAR Section 2.3.10.1).

DOE biosphere model results are quantified by the Biosphere Dose Conversion Factors (BDCFs). A biosphere dose conversion factor is the calculated annual dose to the RMEI from all potential exposure pathways as a result of a unit concentration of a radionuclide in groundwater or surface soil mixed with volcanic ash (SAR Section 2.3.10.1). Mean groundwater exposure scenario biosphere dose conversion factors and primary exposure pathways (from SAR Tables 2.3.10-11 and 2.3.10-12) for radionuclides that are important contributors to the DOE TSPA annual dose results (SAR Figure 2.4-26 a and b) are provided in Table 17-6. (Note: The volcanic ash exposure scenario for the igneous eruptive modeling case is discussed in the next section.)

Table 17-6. DOE Groundwater BDCFs*

Radionuclide	Mean BDCF Sv/yr per Bq/m^3 [mrem/yr per pCi/L]	Primary Pathways
Tc-99	1.1×10^{-9} [4.1×10^{-3}]	42% drinking water 37% animal product 17% crop
Np-237	2.7×10^{-7} [1.0]	56% inhalation 29% drinking water
Pu-242	9.1×10^{-7} [3.4]	75% inhalation 19% drinking water

*Biosphere dose conversion factors

The average annual doses are largest for the seismic ground motion and igneous intrusive modeling cases (generally a factor of 10 or more larger than the other modeling cases; see SAR Figure 2.4-18). Tc-99 (a nonsorbing radionuclide) is the largest contributor to the average annual dose in the initial 10,000 years. Tc-99 accounts for approximately 0.001 mSv/yr [0.1 mrem/yr] of the peak of the overall average annual dose of approximately 0.003 mSv/yr [0.3 mrem/yr] (SAR Figure 2.4-20a). After 10,000 years and up to 1 million years, the peak of the overall average annual dose occurs at 1 million years, with Pu-242 and Np-237 being the largest contributors to the peak of the overall average annual dose. Pu-242 and Np-237 account for approximately 0.01 mSv/yr [1.0 mrem/yr] of the peak of the overall average annual dose of approximately 0.02 mSv/yr [2.0 mrem/yr] at 1 million years (SAR Figure 2.4-20b).

2.2.1.4.1.3.3.1.2 NRC Staff's Review of DOE's TSPA Calculation Related to Groundwater Releases

The NRC staff conducted confirmatory calculations to assist in its review of the DOE TSPA results. The confirmatory calculations provide both a quantitative understanding of the attributes of the performance assessment and an understanding of whether there is a general consistency between submodels of the performance assessment and the overall results, including uncertainty (e.g., whether the timing and extent of breaching of the waste package are consistent with the timing and magnitude of the average annual dose). The confirmatory calculations were performed for selected time periods (i.e., 10,000; 100,000; 400,000; and 800,000 years) to provide perspective on the time-dependent nature of waste package failure, associated radioactive decay, and release of specific radionuclides. Detailed documentation of the NRC staff's confirmatory calculation is provided in NRC and CNWRA (2014aa).

The confirmatory calculations are based on the NRC staff's understanding of the TSPA calculation obtained from its SAR review, including the TSPA models and the code's intermediate outputs. Thus, the confirmatory calculations address key quantitative attributes of the repository system to help evaluate overall performance. This approach provides a straightforward method for determining whether the TSPA results provide a credible representation of the repository performance (i.e., the average annual dose curve is consistent with the model abstractions, probabilities, and treatment of uncertainties, which have been reviewed in SER Sections 2.2.1.2 to 2.2.1.3.14).

In assessing the credibility of the DOE TSPA average annual dose curve resulting from the groundwater pathway, separate confirmatory calculations were conducted for (i) the amount of water entering failed waste packages, (ii) the release of radionuclides from the waste packages, (iii) transport of radionuclides through the saturated and unsaturated zones, and (iv) annual dose to the RMEI. The calculations were performed using representative values (e.g., mean and median values); however, solubility limits made use of a high and low value because these limits had the greater potential to affect the dose estimates for some radionuclides considered in the analysis. Simplifying assumptions used in performing the calculations were intended to not underestimate peak dose); for example, decrease in the ash deposit due to erosion and weathering was not considered.

Amount of Water Entering Failed Waste Packages

The NRC staff determined a representative quantity of seepage water entering the waste package using average values from the SAR for the seepage rates and seepage fraction. Specifically, the NRC staff divided the seepage rates by the seepage fraction (Table 17-2) to obtain the seepage rate at dripping locations in the drifts and multiplied this rate by the cross-sectional area of an emplacement drift {27.5 m^2 [296 ft^2]} to determine the seepage volume entering the drift. The amount of seepage water entering the drift that then enters the waste package is determined by the extent of the damage to the surface of the waste package (i.e., the size of the opening in the waste package from patch openings). The DOE TSPA assumes when 4 percent or more of the surface is failed, all seepage enters the waste package; otherwise, the amount of seepage entering the waste package linearly decreases below 4 percent to 0 percent. The NRC staff's approach closely approximates the value DOE used (see SER Section 2.2.1.3.3.3.3 for the NRC staff evaluation of the quantity of water entering the waste package). The average damage to the surface of the waste package was obtained from DOE (2009bj, Enclosure 1, Response Number 1, Figures 11, 12, 15, and 16). Table 17-7 presents the average volume of seepage that might enter a waste package that has patch failure (assuming seepage is present and the drip shield has failed) at specific times over the 1-million-year period (i.e., 10,000; 100,000; 400,000; and 800,000 years). One particularly important aspect of the NRC staff's calculated values in Table 17-7 is the average amount of seepage water entering patches due to ruptures and punctures decreases between 400,000 years and 800,000 years. This counterintuitive result is due to (i) ruptures, which can damage a larger surface area than punctures, occurring at early times and dominating the early average value for damage and (ii) punctures, occurring primarily at later times (when drifts are degraded and drip shields and waste packages are weakened by corrosion) and dominating the later average values. General corrosion of the waste package after very long time periods results in the largest amount of water contacting waste for the seismic ground motion modeling case because there are approximately 5 times more waste packages breached by general corrosion than by ruptures and punctures after 800,000 years (see Table 17-3). The igneous intrusion modeling case, which assumes the waste package and drip shield no longer prevent water from contacting waste, represents the maximum amount, on average, of water that can

Table 17-7. NRC Staff Confirmatory Calculation Results for the Average Volume of Seepage Water Entering Patch Failures in a Single Waste Package for Seismic Ground Motion (Ruptures, Punctures, and General Corrosion) and Igneous Intrusion Modeling Cases for CSNF* and CDSP† Waste Packages				
	Volume (in liters/yr)			
Process	10,000 Years	100,000 Years	400,000 Years	800,000 Years
Ruptures and punctures	CSNF 0 CDSP 208	CSNF 0 CDSP 630	CSNF 63 CDSP 94	CSNF 47 CDSP 47
General corrosion	CSNF 0 CDSP 0	CSNF 0 CDSP 0	CSNF 11 CDSP 13	CSNF 47 CDSP 63
Igneous intrusion	CSNF 609 CDSP 609	CSNF 892 CDSP 892	CSNF 892 CDSP 892	CSNF 892 CDSP 892

*Commercial spent nuclear fuel
†Codisposal

Note: Values of zero indicate no failed packages for the indicated failure type and time

enter a failed waste package and contact waste. Larger amounts of water generally result in a larger release of solubility limited radionuclides.

Release of Radionuclides From Waste Packages

Release rates of radionuclides from an individual waste package are dependent primarily on the type of radionuclide. High mobility characterizes the soluble, nonsorbing radionuclides (e.g., Tc-99, I-129, Cl-36, Se-79) that result in relatively rapid waste package depletion after failure, whereby repository release is controlled by the amount present in the waste package or inventory and the degradation rate of the waste form. The overall release rate from the repository is dependent on the releases from the individual waste packages and how the releases from all the failed waste packages add together (or overlap at a particular time) to produce an overall release rate for the repository. For example, if all the packages failed at the same time, then the releases from all the waste packages would be occurring at the same time and would combine to produce the repository release rate. If, however, waste packages fail at different times, the potential for the releases to overlap in time will depend on the length of time between the failed packages and the time it takes a waste package to release the inventory of a particular radionuclide. When releases from a waste package are somewhat rapid, occurring over hundreds to thousands of years, as is the case for the high-mobility radionuclides, the potential for releases from all the waste packages to overlap in time is reduced unless all the waste packages fail within the same time period over which the rapid release occurs. High release rates will persist for short periods of time (e.g., hundreds to thousands of years); thus, the overlap period for high waste package release rates will be short (a smaller number of waste package releases could potentially overlap in time). In contrast, low release rates may persist for hundreds of thousands of years and longer and the overlap time period would be much longer and include the potential for a larger number of failed packages to contribute to the overall repository release rate.

As part of the NRC staff confirmatory calculation (NRC and CNWRA, 2014aa), a simplified approach was used to estimate releases from the waste package for three radionuclides important to the annual dose: Tc-99 (a soluble, nonsorbing radionuclide) and Np-237 and Pu-242 (relatively insoluble, sorbing radionuclides). The release rate from the repository engineered barriers is determined by multiplying the release rate from a single waste package times the number of waste package failures. This simple approach has the potential to calculate releases from the repository that would exceed, over the time period of the waste package failures, the amount of material that was present in the failed waste packages when release rates are high. NRC staff calculated a bounding value for the release from a single waste package to ensure the releases from the repository would not exceed the amount of material present in the failed waste packages [bounding value is calculated as the inventory of a single waste package times the waste package failure rate; see NRC and CNWRA (2014aa) for further details]. The release rate from the engineered barriers for the repository were determined for selected time periods (i.e., 10,000; 100,000; 400,000; and 800,000 years) to provide some perspective on the time-dependent nature of waste package failure, the effect of radioactive decay, and the release rate for a specific radionuclide.

For soluble, nonsorbing radionuclides (e.g., Tc-99), the release rate from the waste packages is assumed to be relatively rapid including release by diffusion through small cracks. Because diffusional release of soluble, nonsorbing radionuclides can be significant, all waste package breach types (e.g., rupture and puncture, general corrosion patches, stress corrosion cracks) are significant to estimating releases from the repository for the soluble nonsorbing radionuclides. Consistent with a rapid release time for soluble, nonsorbing radionuclides, DOE has stated that the release of Tc-99 tracks the waste package failure (SAR p. 2.4-63). The NRC staff used the bounding value for the release rate of Tc-99 in its confirmatory calculation because a relatively rapid release rate for a soluble, nonsorbing radionuclide can result in most, if not all, of the inventory being released over time periods that are short (e.g., thousands of years) relative to the time periods of the confirmatory calculation. The results of this simple, bounding calculation for Tc-99 showing the peak values at various times for the seismic ground motion modeling case and the igneous intrusion modeling case appear in Table 17-8. (Note: The waste package failure rate for the intrusion modeling case also includes a multiplicative factor to account for the annual probability for the intrusive event to occur.) Given that the majority of waste package failures in the seismic ground motion modeling case are due to crack failures (see Table 17-3) where the only release will be from the diffusion process, the diffusional releases are significant for the release of soluble, nonsorbing radionuclides (e.g., Tc-99) in the ground motion modeling case.

Table 17-8. NRC Staff Confirmatory Calculation Results for the Average Release Rates for Tc-99 (Seismic Ground Motion Modeling Case) for CSNF* and CDSP† Waste Packages

	10,000 Years	**100,000 Years**	**400,000 Years**	**800,000 Years**
Waste package failure rate‡ (waste packages/yr)	CSNF 1.6×10^{-4} CDSP 3.42×10^{-3}	CSNF 2.05×10^{-4} CDSP 0.0102	CSNF 1.85×10^{-3} CDSP 3.42×10^{-3}	CSNF 4.41×10^{-3} CDSP 2.56×10^{-3}
Inventory§ (grams)	CSNF 7,405 CDSP 1,131	CSNF 5,526 CDSP 844	CSNF 2,082 CDSP 318	CSNF 567 CDSP 86
Average release rate (grams/yr)	CSNF 1.2 CDSP 3.9	CSNF 1.1 CDSP 8.6	CSNF 3.8 CDSP 1.1	CSNF 2.5 CDSP 0.22

*Commercial spent nuclear fuel
†Codisposal
‡Determined using all waste package failures types (i.e., cracks and patches) from Table 17-3 divided by the time .
§Inventory is for one waste package at the specific time (Tc-99 half-life is 213,000 years); initial inventory from SAR Volume 2 Table 2.3.7-5.

Much lower mobility characterizes the relatively insoluble, sorbing nuclides (e.g., Np-237, Pu-242). Advective releases of these radionuclides are limited by maximum limits on their concentrations in water (from either precipitation or sorption to stationary corrosion products) and water flow through the waste package. Because releases of the lower mobility nuclides can take tens of thousands of years and longer, the potential for releases from individual waste packages to overlap in time is greater. For the concentration-limited radionuclides (e.g., Pu-242 and Np-237), DOE has explained that releases will be significantly lower than the release rates for soluble radionuclides (e.g., Tc-99) and will increase as water flow into the waste packages increases (e.g., as corrosion patch area increases in size, over time more water may enter the waste package; see SAR p. 2.4-63).

For the relatively insoluble, sorbing radionuclides (e.g., Np-237, Pu-242), the approximation for the repository-wide EBS release rate accounts for key aspects of repository performance that affect the release rate from the waste package (i.e., solubility limits, volume of water flux moving through a breached waste package, corrosion products, and radioactive decay and production). The release rate from the repository engineered barriers is determined by multiplying the release rate from a single waste package times the number of waste package failures, subject to the previously described constraint that the releases from the repository cannot exceed the amount of material present in the failed waste packages. Generally, the constraint on the release rate is necessary when the inventory is small and/or when the solubility limits and water flow rates are high. As the length of time for the inventory of a specific radionuclide to be completely released from a waste package increases, the potential for releases from waste packages, which failed at different times, to overlap or combine and result in larger releases from the repository increases. For example, if the time to release the entire inventory of a given radionuclide was 100,000 years, then the releases from all waste packages that failed within 100,000 years would all overlap by the end of the 100,000-year period. The NRC staff confirmatory calculation accounted for the overlap of waste package releases, which can vary with time due to radioactive decay and increasing water flow through the waste package, attributed to an increased damaged area of waste package patch breaches, which generally occurs at later times [e.g., increased general corrosion patch area at longer times, as shown in DOE (2009bj, Enclosure 1, Response Number 1, Figures 11 and 12)].

Generally, for the insoluble radionuclides, the larger repository release rates occur at later times (e.g., hundreds of thousands of years) when a larger number of waste packages are breached by patch failures, which allow seepage water to enter the waste package and release radionuclides by the advective flow of water out of the waste package. The advective releases, when present, tend to be significantly larger than the diffusive releases [see NRC staff analysis in NRC and CNWRA (2014aa)]. Therefore, the NRC staff's confirmatory calculation considers only advective releases out of the waste package for the insoluble radionuclides, which occurs only for failed packages in the dripping environment after the drip shield fails. The NRC staff's calculation assumes that the drip shield is always failed when the waste package is breached by a patch failure (i.e., an opening sufficiently large that dripping water can enter the waste package). Advective releases are calculated by estimating (i) the failure rate for waste packages breached by patches, such as by general corrosion and ruptures and punctures from the seismic ground motion modeling case; (ii) the amount of seepage that enters and flows through the waste package, which is dependent on the size of the waste package holes or patches, the seepage fraction and seepage rates, and for the NRC staff's calculation, an assumption that the drip shield is not functioning as a barrier to seepage; and (iii) the effects of solubility limits and corrosion products on the concentration of radionuclides in the water flowing through the waste package. The results of the NRC staff's simplified calculations for Np-237 and Pu-242 for the ground motion modeling case are shown in Table 17-9. Uncertainty was explicitly accounted for in the release of these radionuclides. Values for the solubility limits used upper and lower values representative of the uncertainty range DOE used in its TSPA analysis (see SER Section 2.2.1.3.4).

Table 17-9. NRC Staff Confirmatory Calculation Results for the Average Release Rates for Np-237 and Pu-242 in the Seismic Ground Motion Modeling Case for CSNF* and CDSP† Waste Packages				
	10,000 Years	**100,000 Years**	**400,000 Years**	**800,000 Years**
General corrosion cumulative failures ‡ (waste packages)	CSNF 0 CDSP 0	CSNF 0 CDSP 0	CSNF 0.3 CDSP 4.8	CSNF 328.5 CDSP 239.1
Water flux through general corrosion§ (liters/yr/waste package)	CSNF 0 CDSP 0	CSNF 0 CDSP 0	CSNF 11 CDSP 13	CSNF 47 CDSP 63
Rupture and puncture cumulative failures‡ (waste packages)	CSNF 0 CDSP 0.3	CSNF 0 CDSP 0.7	CSNF 24.6 CDSP 13.7	CSNF 82.1 CDSP 34.2
Water flux through ruptures and punctures§ (liters/yr/waste package)	CSNF 0 CDSP 208	CSNF 0 CDSP 630	CSNF 63 CDSP 94	CSNF 47 CDSP 47
Np-237 inventory‖ (grams/waste package)	CSNF 15,530 CDSP 504	CSNF 15,084 CDSP 490	CSNF 13,687 CDSP 444	CSNF 12,024 CDSP 390
Np-237 release rate¶ (grams/yr)	CSNF 0 CDSP 2.9×10^{-4} to 4.7×10^{-3}	CSNF 0 CDSP 2.4×10^{-3}	CSNF 0.016 to 0.32 CDSP 0.011 to 0.014	CSNF 0.20 to 4.0 CDSP 0.092
Pu-242 inventory‖ (grams/waste package)	CSNF 5,360 CDSP 39	CSNF 4,542 CDSP 33	CSNF 2,614 CDSP 19	CSNF 1,251 CDSP 9
Pu-242 release rate¶ (grams/yr)	CSNF 0 CDSP 8.1×10^{-5} to 3.6×10^{-4}	CSNF 0 CDSP 1.6×10^{-4}	CSNF 4.5×10^{-3} to 0.11 CDSP 6.1×10^{-4}	CSNF 0.056 to 0.44 CDSP 2.2×10^{-3}

*Commercial spent nuclear fuel
†Codisposal
‡From Table 17-3 patch failures (i.e., ruptures, punctures, and general corrosion).
§Water flux taken from Table 17-7.
‖Inventory is for one waste package at the specific time (Np-237 half-life is 2.14 million years, Pu -242 half-life is 376,300 years); initial inventory from SAR Volume 2 Table 2.3.7-5 (Np-237 inventory includes complete decay of Am-241 into Np-237).
¶Release rate determined with solubility limits of 0.3 to 6 mg/L for Np-237 and 0.006 to 0.5 mg/L for Pu-242 and corrosion product factor of 0.05 for Np-237 and 0.7 for Pu-242 (NRC and CNWRA. 2014aa).

The release rates from the CSNF waste packages overall are larger than the releases from the CDSP fuel packages due to the larger inventory for these radionuclides present in CSNF and the larger number of CSNF waste packages in the repository. The NRC staff's calculation does not account for releases of radionuclides from the waste packages that are associated with colloids. This is because releases of radionuclides that are not associated with colloids (i.e., radionuclides that are dissolved in water) are larger than releases of radionuclides associated with colloids (see the NRC staff's review in SER Section 2.2.1.3.4.3.4).

The NRC staff performed confirmatory calculations for the igneous intrusive modeling case similar to those performed for the seismic ground motion modeling case. The igneous intrusive modeling case is somewhat simpler in that it is assumed that all waste packages are completely failed (i.e., no diversion of seepage water) once the event occurs and the seepage fraction is 100 percent (i.e., all packages experience dripping water). The NRC staff accounted for the probability of the event occurring (i.e., mean annual frequency of an igneous event intersecting the repository is 1.7×10^{-8} per year; SAR p. 2.3.11-9) by incorporating the event probability into the determination of the number of waste package failures (see Table 17-3). The number of waste package failures in the seismic ground motion modeling case reported in Table 17-3 also represents values that incorporate the probability of the seismic events occurring. The releases for the igneous intrusive modeling case are presented in Table 17-10 for the soluble radionuclide Tc-99 and in Table 17-11 for the insoluble radionuclides Np-237 and Pu-242.

Transport of Radionuclides through the Unsaturated and Saturated Zones

As part of the NRC staff's confirmatory calculation, the NRC staff developed multiplicative factors to account for the effect of transport in the DOE TSPA evaluation for the unsaturated and saturated zones on the releases of radionuclides to the RMEI location. The effectiveness of the travel times in the geosphere is related to the time at which the annual dose occurs because the travel time effectively delays radionuclide transport to the RMEI location and thereby delays the dose. Thus, the NRC staff developed factors for representing the effects of the unsaturated and saturated zones for reducing radionuclide releases at specific time periods in the confirmatory calculation (see Table 17-12). The nonsorbing radionuclide Tc-99 is typified by short delay time, and thus, the releases are unaffected (i.e., no reduction in the release rates resulted in a reduction factor of 1) by either the unsaturated or saturated zones. As noted in Table 17-4, DOE representation for the unsaturated zone is split with slower releases for sorbing radionuclides for the southern half of the repository. For the NRC staff's confirmatory calculation, the unsaturated zone reduces the releases for the sorbed radionuclides (Np-237 and Pu-242) up to 50 percent (i.e., a reduction factor of 0.5) due to delay times on the order of thousands of years and longer in the southern portion of the unsaturated zone. Sorption of radionuclides causing the delay or slowing of radionuclide travel times for thousands of years will be less noticeable at longer times (e.g., 100,000 years). After 10,000 years, only the more strongly sorbed radionuclide Pu-242 continues to be reduced by the effectiveness of the unsaturated zone. (Note: a small quantity of Pu-242 is associated with irreversible colloids that would not be reduced by sorption; however, this amount is small in the DOE TSPA analysis, as explained in SER Section 2.2.1.4.1.3.3.1.1.4, and is not considered in the NRC staff's confirmatory calculation.) The saturated zone provides somewhat more effectiveness for Pu-242 (reversible colloids and dissolved Pu-242) at longer times and is assumed to significantly reduce releases (i.e., 97 percent reduction resulting in a reduction factor of 0.03) over the 10,000-year time period due to the median transport time of 95,000 years (see Table 17-5). The saturated zone is less effective for Np-237 because median transport time is 3,700 years (see Table 17-5). The confirmatory calculation (i) considered radionuclides that are released to the RMEI location representing the more significant contributors to the TSPA-calculated average annual dose and (ii) did not consider releases of a variety of other radionuclides that were reduced to much lower levels due to radioactive decay when the waste package is intact and during transport in the unsaturated and saturated zones (e.g., reversible colloids for americium and thorium are

Table 17-10. NRC Staff Confirmatory Calculation Results for the Average Release Rates for Tc-99 in the Igneous Intrusive Modeling Case for CSNF* and CDSP† Waste Packages				
	10,000 Years	**100,000 Years**	**400,000 Years**	**800,000 Years**
Waste package failure rate‡ (waste packages/yr)	CSNF 1.4×10^{-4} CDSP 5.8×10^{-5}	CSNF 1.4×10^{-4} CDSP 5.8×10^{-5}	CSNF 1.4×10^{-4} CDSP 5.8×10^{-5}	CSNF 1.4×10^{-4} CDSP 5.8×10^{-5}
Inventory§ (grams)	CSNF 7,405 CDSP 1,131	CSNF 5,526 CDSP 844	CSNF 2,082 CDSP 318	CSNF 567 CDSP 86
Average release rate (grams/yr)	CSNF 1.0 CDSP 0.068	CSNF 0.77 CDSP 0.049	CSNF 0.29 CDSP 0.018	CSNF 0.079 CDSP 5.0×10^{-3}

*Commercial Spent Nuclear Fuel
†Codisposal
‡determined using values in Table 17-3 divided by the time.
§Inventory is for one waste package at the specific time (Tc-99 half-life is 213,000 years); initial inventory from SAR Volume 2 Table 2.3.7-5.

Table 17-11. NRC Staff Confirmatory Calculation Results for the Average Release Rates for Np-237 and Pu-242 in the Igneous Intrusive Modeling Case for CSNF* and CDSP† Waste Packages				
	10,000 Years	**100,000 Years**	**400,000 Years**	**800,000 Years**
Waste package cumulative failures‡ (packages)	CSNF 1.4 CDSP 0.58	CSNF 14.0 CDSP 5.8	CSNF 55.9 CDSP 23.2	CSNF 111.7 CDSP 46.5
Water flux§ (liters/yr/waste package)	CSNF 609 CDSP 609	CSNF 892 CDSP 892	CSNF 892 CDSP 892	CSNF 892 CDSP 892
Np-237 inventory‖ (grams/waste package)	CSNF 15,530 CDSP 504	CSNF 15,084 CDSP 490	CSNF 13,687 CDSP 444	CSNF 12,024 CDSP 390
Np-237 release rate¶ (grams/yr)	CSNF 0.013 to 0.26 CDSP 5.3×10^{-3} to 0.029	CSNF 0.19 to 2.1 CDSP 0.028	CSNF 0.75 to 1.9 CDSP 0.026	CSNF 1.5 to 1.7 CDSP 0.023
Pu-242 inventory‖ (grams/waste package)	CSNF 5,360 CDSP 39	CSNF 4,542 CDSP 33	CSNF 2,614 CDSP 19	CSNF 1,251 CDSP 9
Pu-242 release rate¶ (grams/yr)	CSNF 3.6×10^{-3} to 0.30 CDSP 1.5×10^{-3} to 2.3×10^{-3}	CSNF 0.052 to 0.64 CDSP 1.9×10^{-3}	CSNF 0.21 to 0.37 CDSP 1.1×10^{-3}	CSNF 0.17 CDSP 5.2×10^{-4}

*Commercial Spent Nuclear Fuel
†Codisposal
‡From Table 17-3.
§Water flux taken from Table 17-7.
‖ Inventory is for one waste package at the specific time (Np-237 half-life is 2.14 million years, Pu -242 half-life is 376,300 years); initial inventory from SAR Volume 2 Table 2.3.7-5 (Np-237 inventory includes complete decay of Am-241 into Np-237).
¶Release rate determined with solubility limits of 0.3 to 6 mg/L for Np-237 and 0.006 to 0.5 mg/L for Pu-242 and corrosion product factor of 0.05 for Np-237 and 0.7 for Pu-242 (NRC and CNWRA. 2014aa).

Table 17-12. NRC Staff Confirmatory Calculation Values for the Effectiveness (factor of 1.0 results in no reduction in releases) of the Unsaturated and Saturated Zones for Reducing Release Rates for Specific Radionuclides*

	0 to 10,000 Years	10,000 to 200,000 Years	200,000 to 600,000 Years	600,000 to 1 million Years
Tc-99 (nonsorbing)	UZ 1.0 SZ 1.0	UZ 1.0 SZ 1.0	UZ 1.0 SZ 1.0	UZ 1.% SZ 1.0
Np-237 (moderately sorbing)	UZ 0.55 SZ 0.60	UZ 0.95 SZ 0.90	UZ 1.0 SZ 0.95	UZ 1.0 SZ 1.0
Pu-242 (aqueous) (strongly sorbing)	UZ 0.50 SZ 0.03	UZ 0.50 SZ 0.40	UZ 0.50 SZ 0.65	UZ 0.50 SZ 0.75

*For more details regarding NRC staff determination of the effectiveness factor, see NRC and CNWRA, 2014aa.

delayed more than 1 million years in the saturated zone; see Table 17-5), because the contributions to the average annual dose would have been so small.

Annual Dose to the RMEI

The NRC staff's confirmatory calculation for the annual dose to the RMEI is completed by multiplying the Biosphere Dose Conversion Factors (BDCFs) in Table 17-6 with the NRC staff's estimated saturated zone releases to estimate an annual dose for comparison with DOE TSPA results. Tables 17-13 through 17-15 compare the confirmatory calculation and the TSPA results. Overall, the annual doses from the confirmatory calculation are in general agreement with the TSPA results (e.g., a majority of either single value comparisons are within a factor of two or single values for the TSPA results fall between the upper and lower values when the confirmatory calculation provides both an upper and lower value). The igneous intrusive modeling case, which is already somewhat simplified in the TSPA model by assuming all waste packages fail when the event occurs, tends to exhibit the best fit between the confirmatory calculation and the TSPA results. The fit for Tc-99 also exhibits a better fit regardless of the modeling case because the representation of Tc-99 in the repository is less complex: high solubility and mobility for Tc-99 limits the factors affecting release and transport of Tc-99. Although the ground motion modeling case is a bit more complicated due to the variety and timing of waste package breaches (e.g., cracks, ruptures, and patches), the results of the NRC staff's confirmatory calculation are in general agreement with the TSPA results. There is no precise agreement between the NRC staff's confirmatory calculation and the results of the DOE TSPA results due to the simplifying assumptions made in the confirmatory calculation [see NRC and CNWRA (2014aa) for further details on assumptions for the NRC staff confirmatory calculation]. The NRC staff's confirmatory calculation was used to confirm the NRC staff's understanding of the key attributes of the repository performance in the DOE TSPA analyses and to confirm that those attributes are consistent with DOE dose results. The confirmatory calculation considered the effect on dose by (i) the number of waste packages and the extent of waste package damage; (ii) the drift seepage; (iii) solubility limits for individual radionuclides (including the effect of corrosion products); (iv) the inventory of specific radionuclides; (v) sorption in the geosphere; and (vi) the probability of disruptive events. Consistency between the confirmatory calculation and DOE TSPA results provides further confidence that DOE TSPA analysis is consistent with the model abstractions described in the license application and reviewed in SER Section 2.2.1.3.

Table 17-13. NRC Staff Confirmatory Calculation Results for the Average Dose Estimates for Tc-99 for the Seismic Ground Motion Modeling Case and Igneous Intrusive Modeling Case				
	10,000 Years	**100,000 Years**	**400,000 Years**	**800,000 Years**
Annual release from repository engineered barrier system (grams/year) (from Tables 17-8 and 17-10)	Seismic 5.1 Igneous 1.1	Seismic 9.7 Igneous 0.82	Seismic 4.9 Igneous 0.31	Seismic 2.7 Igneous 0.084
Reduction factor for unsaturated and saturated zone transport (from Table 17-12)	1	1	1	1
BDCF* (mrem/yr per pCi/l) (from Table 17-6)	4.1×10^{-3}	4.1×10^{-3}	4.1×10^{-3}	4.1×10^{-3}
NRC confirmatory calculation of annual dose† (mrem/yr)	Seismic 0.096 Igneous 0.021	Seismic 0.18 Igneous 0.015	Seismic 0.092 Igneous 5.8×10^{-3}	Seismic 0.057 Igneous 1.6×10^{-3}
DOE TSPA average annual dose‡ (mrem/year)	Seismic 0.10 Igneous 0.017	Seismic 0.16 Igneous 0.013	Seismic 0.13 Igneous 7.0×10^{-3}	Seismic 0.090 Igneous 1.6×10^{-3}
* Biosphere dose conversion factor †Annual dose calculation based on an annual water demand of 3,000 acre-feet per 10 CFR 63.312(c). ‡TSPA results are approximate based on SAR Figures 2.4-26 and 2.4-30.				

Table 17-14. NRC Staff Confirmatory Calculation Results for the Annual Dose for Np-237 for the Seismic Ground Motion Modeling Case and Igneous Intrusive Modeling Case				
	10,000 Years	**100,000 Years**	**400,000 Years**	**800,000 Years**
Annual release from repository engineered barrier system (grams/yr) (from Tables 17-9 and 17-11)	Seismic 2.9×10^{-4} to 4.7×10^{-3} Igneous 0.018 to 0.29	Seismic 2.4×10^{-3} Igneous 0.22 to 2.1	Seismic 0.027 to 0.34 Igneous 0.78 to 1.9	Seismic 0.29 to 4.1 Igneous 1.5 to 1.7
Reduction factor for unsaturated and saturated zone transport (from Table 17-12)	0.33	0.855	0.95	1
BDCF* (mrem/yr per pCi/l) (from Table 17-6)	1	1	1	1
NRC confirmatory calculation of annual dose† (mrem/yr)	Seismic 1.8×10^{-5} to 3.0×10^{-4} Igneous 1.1×10^{-3} to 0.018	Seismic 4.0×10^{-4} Igneous 0.036 to 0.34	Seismic 5.0×10^{-3} to 0.061 Igneous 0.14 to 0.34	Seismic 0.055 to 0.78 Igneous 0.29 to 0.32
DOE TSPA average annual dose‡ (mrem/yr)	Seismic 1.5×10^{-6} Igneous 3.0×10^{-3}	Seismic 2×10^{-4} Igneous 0.05	Seismic 2×10^{-3} Igneous 0.13	Seismic 0.04 Igneous 0.22
*Biosphere dose conversion factor †Annual dose calculation based on an annual water demand of 3,000 acre-feet per 10 CFR 63.312(c). ‡TSPA results are approximate based on SAR Figures 2.4-26 and 2.4-30.				

Table 17-15. NRC Staff Confirmatory Calculation Results for the Annual Dose for Pu-242 for the Seismic Ground Motion Modeling Case and Igneous Intrusive Modeling Case				
	10,000 Years	**100,000 Years**	**400,000 Years**	**800,000 Years**
Annual release from repository engineered barrier system (grams/yr) (from Tables 17-9 and 17-11)	Seismic 8.1×10^{-5} to 3.6×10^{-4} Igneous 5.1×10^{-3} to 0.3	Seismic 1.6×10^{-4} Igneous 0.054 to 0.64	Seismic 5.1×10^{-3} to 0.11 Igneous 0.21 to 0.37	Seismic 0.058 to 0.44 Igneous 0.17
Reduction factor for unsaturated and saturated zone transport (from Table 17-12)	0.015	0.20	0.325	0.375
BDCF* (mrem/yr per pCi/l) (from Table 17-6)	3.4	3.4	3.4	3.4
NRC confirmatory calculation of annual dose† (mrem/yr)	Seismic 4.3×10^{-6} to 2.0×10^{-5} Igneous 2.8×10^{-4} to 0.016	Seismic 1.2×10^{-4} Igneous 0.040 to 0.47	Seismic 6.2×10^{-3} to 0.13 Igneous 0.25 to 0.43	Seismic 0.080 to 0.58 Igneous 0.23
DOE TSPA average annual dose‡ (mrem/year)	Seismic 0 Igneous 0	Seismic 4.0×10^{-4} Igneous 0.05	Seismic 0.013 Igneous 0.23	Seismic 0.15 Igneous 0.23
*Biosphere dose conversation factor †Annual dose calculation based on an annual water demand of 3,000 acre-feet per 10 CFR 63.312(c). ‡TSPA results are approximate based on SAR Figures 2.4-26 and 2.4-30.				

Summary of NRC Staff's Review of DOE TSPA Calculation Related to Groundwater Releases

The NRC staff conducted its confirmatory calculation to assist in its review of the DOE TSPA models and calculations. The confirmatory calculation provides both a quantitative understanding of the attributes of the performance assessment and an understanding of whether there is a general consistency between submodels of the performance assessment and the overall results (e.g., whether the timing and extent of breaching of the waste package is consistent with the timing and magnitude of the average annual dose) including uncertainty. On the basis of its confirmatory calculation, the NRC staff makes the following findings:

- The NRC staff's confirmatory calculation follows the water and key radionuclides through the repository system (seepage to release to transport to annual dose) for the dominant modeling cases (i.e., seismic ground motion and igneous intrusion events have the largest number of waste package failures and are the largest contributors to annual dose). The NRC staff finds credible the DOE TSPA calculated average annual dose curve in that the average annual doses are consistent with the intermediate model results (e.g., seepage, waste package failures, release of radionuclides from the engineered barrier system, and transport in the geosphere). The NRC staff's review of the intermediate models of the DOE TSPA is described under the NRC staff's model abstraction review in SER Section 2.2.1.3.

- DOE description of the barriers important to waste isolation is fully consistent with the key attributes of the repository in the NRC staff's confirmatory calculation. In particular,

(i) The waste package prevents significant releases for long periods of time (e.g., less than 1 percent of CSNF waste packages are breached after 100,000 years; see Table 17-3)

(ii) Once breached, the releases from the waste package are limited due to the manner in which the waste package fails (i.e., majority of waste package breaches are due to cracks that do not allow seepage water to enter the waste packages), which limits the amount of water that enters the waste package; solubility limits and corrosion products within the waste package that reduce the release of many radionuclides from the waste package; and the limited amount of seepage that is present due to the upper natural barrier (i.e., rock layers above the repository)

(iii) After release from the waste package, a variety of radionuclides sorb onto rock surfaces and are delayed for thousands of years in portions of both the unsaturated zone and the saturated zone (see Table 17-12)

- The confirmatory calculation focused on certain long-lived radionuclides, on the order of 100,000 years or longer, that could eventually be released at the RMEI location in sufficient quantities to be important for dose calculations. Both a nonsorbing radionuclide (Tc-99) and sorbing radionuclides (Np-237 and Pu-242) were considered in the NRC staff's calculations. The initial inventory of high-level waste is composed of a large quantity (in terms of curies) of radionuclides (e.g., Cs-137, Sr-90, Am-241) with shorter half-lives, on the order of 1,000 years and less. For the short-lived radionuclides, DOE described in DOE (2009an, Enclosure 7, Response Number 7, Table 1-3) that delay in the geosphere has the capability to reduce releases nearly 100 percent, thus, short-lived radionuclides were not included in the NRC staff's confirmatory calculation for radionuclide releases to groundwater.

- The effect of uncertainties on the DOE average annual dose curve is limited primarily because a number of important aspects of repository performance are near maximum values. For example, after 10,000 years nearly all the waste packages are dripped on (seepage fraction of 69 percent for seismic ground motion modeling case and 100 percent for igneous intrusion; see Table 17-2) and, given the 1-million-year period, a variety of long-lived radionuclides can eventually make it to the RMEI location (see Tables 17-4 and 17-5). Releases from the waste package will be affected by the failure rate for the waste package, including the areal extent of the waste package breaches, solubility limits, and the effect of corrosion products. The confirmatory calculation considered the low and high values of the solubility limits to provide some insight on how uncertainty in release from the waste package might impact the annual dose. Use of the highest solubility limit, as expected, increases the annual dose in the NRC staff's confirmatory calculation. The estimated peak dose in the NRC staff's confirmatory calculation at 10,000 years is 0.0014 mSv/yr [0.14 mrem/yr] compared to the regulatory limit of 0.15 mSv/yr [15 mrem/yr] for the initial 10,000 years and an estimated peak dose of 0.025 mSv/yr [2.5 mrem/yr] at 800,000 years compared to the regulatory limit of 1.0 mSv/yr [100 mrem/yr] for the period after 10,000 years.

On the basis of the DOE license application description of the TSPA models and its results, the NRC staff's confirmatory calculation provides further confidence that the DOE average annual dose curve is consistent with the model abstractions, scenario probabilities, and the capabilities of the barriers important to waste isolation and meets the regulatory limits up to 10,000 years and after 10,000 years up to the period of geologic stability (i.e., 1 million years).

2.2.1.4.1.3.3.2 DOE TSPA Calculation for the Volcanic Eruption Modeling Case

2.2.1.4.1.3.3.2.1 Summary of DOE's Approach in the TSPA for the Volcanic Eruption Modeling Case

An eruptive volcanic event at the repository involves the intersection of ascending magma and a drift and eruption at the surface (see SER Section 2.2.1.3.10). Radioactive material entrained in tephra can be transported downwind and deposited on the ground surface where potential exposures can occur from (i) inhalation of radionuclides due to high-level waste entrained in ash particles, which are suspended in the air, including the breathing of radon gas and its daughter products from high-level waste entrained in the ash deposited on the ground surface; (ii) ingestion of radionuclides from locally produced crops and animal products that are assumed to be contaminated from direct (e.g., crops grown in soil containing contaminated tephra) and indirect (e.g., animals raised on feed that has been grown in soil containing contaminated tephra) contact with contaminated tephra; and (iii) external exposure to radionuclide-containing soils and tephra. Estimating the consequences of such an event is dependent on the concentration of radionuclides in tephra and the amount of ash persisting at the RMEI location (from both the direct deposition of tephra during the event and redistribution of tephra after the event due to water and wind action over time).

On average, the volcanic eruption modeling case impacts four (a range of one to seven) waste packages and entrains all of the waste into magma. Of this magma and waste, 30 percent is considered to form tephra; thus, 30 percent of the waste in the waste packages hit by the event is, on average, contained in the tephra (range of 10 to 50 percent). Once radioactively contaminated volcanic tephra is present at the RMEI location, potential exposures are estimated for three specific pathways (external exposure; ingestion; and inhalation, which includes radon exposure) using Biosphere Dose Conversion Factors (BDCFs). DOE BDCFs are provided in Tables 17-16 and 17-17 for some of the radionuclides included in the DOE TSPA relevant to the volcanic eruption modeling case.

The volcanic eruption modeling case average annual dose curve is one of the lowest curves of all the modeling cases resulting in an average dose that is less than 10^{-5} mSv/yr [0.001 mrem/yr] over the 1-million-year period (SAR Figure 2.4-18). During the initial 10,000 years, the annual dose is dominated by Pu-239, Pu-240, and Am-241 (SAR Figure 2.4-32). For these three radionuclides, the inhalation exposure accounts for more than 98 percent of the average annual dose for the volcanic eruption modeling case, as shown in SAR Table 2.3.10-15. At very early times (i.e., the initial 500 years), there is some contribution from Sr-90 and Cs-137 (primarily from external exposure). At very long times (i.e., after 100,000 years), the annual dose is dominated by Ra-226 (SAR Figure 2.4-32). These results are partially due to the half-lives for these radionuclides. Sr-90 and Cs-137 have half-lives less than 100 years, and Am-241 has a half-life of 432 years; thus, the hazard is somewhat short lived. The longer term hazard is with Pu-239 (half-life of 24,000 years), Pu-240 (half-life of 24,000 years), and Ra-226 (half-life of 1,600 years), which is a daughter product in the Uranium series (SAR Figure 2.4-21) containing long-lived U-234 (half-life of 240,000 years) and U-238 (half-life of 4.47 billion years).

Table 17-16. DOE Volcanic Eruption Modeling Case Short-Term and Long-Term Inhalation BDCFs*		
Radionuclide	**BDCF Sv/yr per Bq/kg† [mrem/yr per pCi/g]**	**Primary Pathways**
Pu-239	4.0×10^{-7} [1.5] short term‡ 6.1×10^{-7} [2.3] long term	98% of Pu-239 eruptive dose is inhalation: 39% short term 60% long term
Am-241	3.2×10^{-7} [1.2] short term‡ 5.0×10^{-7} [1.8] long term	94% of Am-241 eruptive dose is inhalation: 37% short term 57% long term

*Biosphere dose conversion factors
†Sources: SAR Table 2.3.10-14 and SNL, 2007, "Biosphere Model Report," MDL–MGR–MD–000001, Rev. 02, Tables 6.12-2 and 6.12-3, Las Vegas, Nevada: Sandia National Laboratories.
‡The short-term inhalation exposure is applicable only for the initial year of the eruption.

Table 17-17. DOE Volcanic Eruption Modeling Case Combined Ingestion, Radon, and External BDCFs*		
Radionuclide	**BDCF Sv/yr per Bq/m²† [mrem/yr per pCi/m²]**	**Primary Pathways**
Sr-90	1.8×10^{-9} [6.7×10^{-6}]	79% external exposure
Cs-137	7.2×10^{-9} [2.7×10^{-5}]	99% external exposure
Ra-226	3.3×10^{-8} [1.2×10^{-4}]	65% external exposure 33% radon decay products

*Biosphere dose conversion factors
†Sources: SAR Table 2.3.10-14 and SNL, 2007, "Biosphere Model Report," MDL–MGR–MD–000001, Rev. 02, Table 6.12-4, Las Vegas, Nevada: Sandia National Laboratories.

2.2.1.4.1.3.3.2.2 NRC Staff's Review of DOE TSPA Calculation for the Volcanic Eruption Modeling Case

The NRC staff reviewed the TSPA and SAR with respect to representing an extrusive igneous event, including the treatment of uncertainty, and finds the DOE approach acceptable. In particular, the NRC staff evaluated and finds acceptable (i) the probability for an extrusive event to intersect the repository and hit waste packages (SER Section 2.2.1.2.2), (ii) the model abstractions for disruption of the waste package by an extrusive igneous event (SER Section 2.2.1.3.10), (iii) the model abstractions for airborne transport and deposition of radionuclides expelled by a potential future volcanic eruption following igneous disruption of waste packages and the redistribution of those radionuclides in soil (SER Section 2.2.1.3.13), and (iv) the volcanic exposure scenario for the RMEI (SER Section 2.2.1.3.14).

The largest contribution to annual dose for the volcanic eruption modeling case is from the inhalation and external exposure pathways (see SAR Table 2.3.10-15). Annual dose from each of these pathways is dependent on the amount of radionuclides that are transported with the tephra (ash), and the inhalation pathway is also dependent on the amount of this material that is potentially available for inhalation by the RMEI (i.e., the mass loading of tephra and radioactive material in the air). In particular, the NRC staff finds that

- The amount of radionuclides entrained in tephra (ash) is reasonably conservative relative to entrainment of rock fragments in other volcanic conduits within similar and different geologic environments to Yucca Mountain (SER Section 2.2.1.3.10)

- Values for mass loading are appropriate and consistent with available information and are generally consistent with the NRC staff's studies and analyses (SER Section 2.2.1.3.14)

The NRC staff also performed a confirmatory calculation to help evaluate the reasonableness of the DOE TSPA average annual dose curve for this modeling case [detailed documentation of the NRC staff's confirmatory calculation is provided in NRC and CNWRA (2014aa)]. Because the average annual dose for the volcanic eruption modeling case is more than 100,000 times less than the regulatory dose limit, a simplified calculation was performed for a few key radionuclides that would tend to bound some of the assumptions for this modeling case. The confirmatory calculation looked at radionuclides that include both the inhalation and direct exposure pathways and assumed that waste entrained in tephra (ash) was deposited directly at the RMEI location and persists without erosion removal at that location over the time period of the estimated dose (i.e., only radioactive decay reduces the level of radionuclides at the RMEI location after the time of the initial deposition of the tephra). Additionally, the confirmatory calculation assumes tephra from the eruptive event is always deposited directly to the RMEI location (i.e., during the volcanic eruption, the wind always blows in the direction of the RMEI such that ashfall occurs at the RMEI location). To provide a perspective on how radioactive decay and probability of the event affects the annual dose, two time periods were selected for the calculation: a time equal to the half-life of the radionuclide and three times the half-life. Radioactive decay will tend to decrease the annual dose to an individual over time. Tables 17-18 through 17-20 present the calculations for a few of the radionuclides that are most significant to performance or radiation exposure to the RMEI in the volcanic eruption modeling case.

The resulting annual doses from the NRC staff's confirmatory calculation are larger than the average annual doses calculated by the DOE detailed TSPA model. This is not unexpected, because the NRC staff's confirmatory calculation, which assumes direct deposits persist without erosion and the wind always blows in the direction of the RMEI, is more conservative than the DOE TSPA, which accounts for erosion of the tephra deposit and the wind direction can vary (See SER Section 2.2.1.3.13). For the volcanic eruption modeling case (Tables 17-18 through 17-20), the annual dose from the confirmatory calculation is significantly below the compliance measures (more than 10,000 times) and the annual dose in the DOE TSPA is also significantly below the compliance measures (more than 100,000 times). Consistency between the NRC staff's confirmatory calculation for the volcanic eruption modeling case (Tables 17-18 through 17-20) and DOE representation of the volcanic eruption modeling case provides further confidence that the DOE TSPA analysis is credible and the representation consistent with the assumptions and models described in the SAR and reviewed in SER Sections 2.2.1.2.2, 2.2.1.3.10, 2.2.1.3.13, and 2.2.1.3.14.

Table 17-18. NRC Staff Confirmatory Calculation Results for Pu-239 and Am-241 Annual Doses for the Volcanic Eruption Modeling Case (Inhalation Pathway)				
	Pu-239 (Half-Life 24,065 Years)		Am-241 (Half-Life 432 Years)	
Performance Aspect	24,065 Years	72,195 Years	432 Years	1,296 Years
Number of waste packages (SAR Figure 2.3.11-12)	4	4	4	4
Fraction of waste entrained in ash (SAR p. 2.3.11-51)	0.3	0.3	0.3	0.3
Tephra volume (km^3)	0.038	0.038	0.038	0.038
Inventory per CSNF* waste package (with mixed oxide added) [Ci] (SAR Table 2.3.7-5)	1,370	343	17,554	4,390
Concentration in ash† (pCi/g)	43	11	550	140
NRC Staff Weighted annual dose‡ (mrem/yr)	3.3×10^{-3}	2.6×10^{-3}	6.0×10^{-4}	4.6×10^{-4}
DOE TSPA weighted average annual dose (mrem/yr) (SAR Figure 2.4-32)	8.0×10^{-5}	6.0×10^{-5}	3.0×10^{-5}	2.0×10^{-5}

*Commercial spent nuclear fuel
†Concentration in ash calculated by dividing quantity of radionuclide released in tephra by tephra volume times the tephra density (tephra density is 1 gram/cc; SAR p. 2.3.11-61).
‡Weighted annual dose calculated by multiplying concentration with the long term BDCF (Table 17-16), annual probability (1.4×10^{-9}), and the time.

Table 17-19. NRC Staff Confirmatory Calculation of Sr-90 and Cs-137 Annual Doses for the Volcanic Eruption Modeling Case (External Pathway)				
	Sr-90 (Half-Life 29 Years)		Cs-137 (Half-Life 30 Years)	
Performance Aspect	29 Years	87 Years	30 Years	90 Years
Number of waste packages (SAR Figure 2.3.11-12)	4	4	4	4
Fraction of waste entrained in ash (SAR p. 2.3.11-51)	0.3	0.3	0.3	0.3
Tephra volume (km^3)	0.038	0.038	0.038	0.038
Inventory per CSNF† waste package (with MOX* added) [Ci] (SAR Table 2.3.7-5)	52,011	13,009	81,518	20,388
Ash areal concentration‡ (pCi/m^2)	1.6×10^{7}	4.1×10^{6}	2.6×10^{7}	6.4×10^{6}
NRC Staff Weighted annual dose§ (mrem/yr)	4.4×10^{-6}	3.4×10^{-6}	2.9×10^{-5}	2.2×10^{-5}
DOE TSPA weighted average annual dose (mrem/yr) (SAR Figure 2.4-32)	2.0×10^{-6}	1.8×10^{-6}	2.0×10^{-5}	1.0×10^{-5}

*mixed oxide fuel
†Commercial spent nuclear fuel
‡Ash areal concentration calculated by dividing quantity of radionuclide released in tephra by tephra volume times the tephra density and assuming a 1-cm [0.39 in]-thick deposit (tephra density is 1 gram/cc; SAR p. 2.3.11-61).
§Weighted annual dose calculated by multiplying concentration with BDCF (Table 17-17), annual probability (1.4×10^{-9}), and the time.

Table 17-20. NRC Staff Confirmatory Calculation of Ra-226 Annual Dose for the Volcanic Eruption Modeling Case (External Pathway)		
	Ra-226 (Half-Life 1,600 Years)	
Performance Aspect	**240,000 Years***	**720,000 Years***
Number of waste packages entrained in magma (SAR Figure 2.3.11-12)	4	4
Fraction of waste entrained in ash (SAR p. 2.3.11-51)	0.3	0.3
Tephra volume (km^3)	0.038	0.038
Inventory per CSNF‡ waste package (with mixed oxide added) [Ci]	8.4§	4.1§
Ash areal concentration‖ (pCi/m^2)	2,650	1,295
NRC Staff Weighted annual dose¶ (mrem/yr)	1.1×10^{-4}	1.6×10^{-4}
DOE TSPA weighted average annual dose (mrem/yr) (SAR Figure 2.4-32)	5.0×10^{-5}	5.0×10^{-5}

*time periods selected based on U-234 (half-life of 240,000 years).
‡Commercial spent nuclear fuel
§Ra-226 inventory assumes Ra-226 activity in secular equilibrium with U-234; (initial U234 and U238 inventories from SAR Table 2.3.7-5).
‖Ash areal concentration calculated by dividing quantity of radionuclide released in tephra by tephra volume times the tephra density and assuming a 1-cm [0.39 in]-thick deposit (tephra density is 1 gram/cc; SAR p. 2.3.11-61).
¶Weighted annual dose calculated by multiplying concentration with BDCF (Table 17-17), annual probability (1.4×10^{-9}), and the time.

2.2.1.4.1.3.4 Statistical Stability of Average Annual Dose Limits

2.2.1.4.1.3.4.1 Summary of DOE's Approach for Statistical Stability

Stability of Average Annual Dose Estimates

DOE addressed the question of the stability of the average annual dose estimates in SAR Section 2.4.2.2.2. The term *stability* refers to the numerical reproducibility of statistics (e.g., average annual dose) or their level of convergence as a function of model features such as size of the statistical sample and numerical approximations. Variation in the TSPA results is a function of a particular combination of uncertain and variable parameters (DOE described its treatment of epistemic and aleatory uncertainty in SAR Section 2.4.2.1.1). DOE identified aleatory parameters as those parameters with uncertainty irreducible by additional experiments or site characterization. Examples of aleatory parameters are the time of seismic and igneous events, the extent of waste package damage during a seismic or faulting event, the location of the compromised or breached waste package in the repository, and the type of waste package [e.g., CSNF or CDSP waste packages] compromised after a disruptive event. The stability of the average annual dose will, in part, be a function of the size of the discrete sample of aleatory parameter values. DOE analyzed the effect of the size of these discrete samples by increasing the number of aleatory realizations from 30 to 90 (SAR pg. 2.4-85), considering more waste package damage fractions for the seismic and faulting modeling case, and accounting for more event times (e.g., doubling the number of event times for the seismic ground motion modeling case, increasing the number of event times from 10 to 50 for the igneous intrusion modeling case) and determined that these types of changes would have a minor effect on the magnitude of the overall average annual dose curve. DOE compared annual dose curves for a set of five realizations for all modeling cases in SAR Figures 2.4-55 to 61 and concluded, in qualitative terms, that the annual dose curve for the analyzed realizations was stable with respect to aleatory uncertainty.

DOE also examined the stability of the average annual dose curve to the treatment of the epistemic uncertainty. The epistemic parameters are generally inputs to specific submodels used to represent the repository and its components that are considered to be fixed or deterministic parameters (e.g., the mean value of a fracture permeability distribution; the unsaturated zone fracture frequency; the temperature dependency of general corrosion of Alloy 22). DOE used a statistical sample size of 300 realizations for each modeling case in the SAR. To examine the stability of the annual dose curve with respect to the treatment of the epistemic uncertainty, DOE estimated dose statistics (mean, median, 5th and 95th percentiles) for the nominal modeling case considering 1,000 realizations and compared those statistics to corresponding 300-realization statistics (SAR Figure 2.4-38). DOE showed the 300-realization and 1,000-realization annual dose statistics (mean, median, 0.05 and 0.95 percentiles) were comparable (SAR Figure 2-4-38).

For all of the modeling cases, DOE considered three replicates with 300 realizations and compared statistics (mean, median, 0.05 and 0.95 percentiles) among the replicates (SAR Figures 2.4-37 to 2.4-52). Each replicate sample had the same number of realizations; however, the combination of sampled parameter values was different for each replicate. DOE qualitatively concluded that the statistics were similar for the three replicates. Also, DOE estimated 95 percent confidence bounds for the average annual dose using information from the replicates and a t-distribution with 2 degrees of freedom, as described in SNL (2008ag, Section J4.10). In all model cases, the 0.95 percentile in the average annual dose was relatively close {e.g., largest difference of 0.01 mSv/yr [1 mrem/yr] between the three replicates and generally much less for the vast majority of the 1-million-year period, as shown in SNL [2008ag, Figure J5-5(a)]} to the overall average annual dose. DOE concluded the overall average annual dose, computed using 300 realizations, to be statistically stable, as described in SAR Section 2.4.2.2.2 and SNL (2008ag, Section 7.3.2).

DOE updated its model from TSPA Model v5.000 to v5.005, with most validation and model stability analyses performed with TSPA Model v5.000, but the annual doses reported in the SAR are based on TSPA Model v5.005 (SAR p. 2.4-76 to 78). DOE compared the effect of the change from version v5.000 to v5.005 and documented those analyses in SNL [2008ag, Figures 7.3.1-17(a) to 7.3.3-13(a)]. Although the stability analyses were not repeated, the comparisons indicate a similar numerical behavior of versions v5.000 to v5.005, and thus, the applicant stated that the same conclusions, with regard to stability, apply to version v5.005. DOE computed a range for the overall average annual dose using bootstrap analyses, compared the results of these analyses in SAR Figures 2.4-53 and 54, and concluded that 300 epistemic realizations were sufficient to estimate the average annual dose and that the results of TSPA Model v5.005 are statistically stable (SAR Section 2.4.2.2.2.2, p. 2.4-82).

Comparison with Annual Dose Standard

DOE presented the overall average annual dose curve over the entire compliance period in SAR Figure 2.4-10. The peak of the overall average annual dose curve is approximately 0.003 mSv/yr [0.3 mrem/yr] over the 10,000-year time period {dose limit of 0.15 mSv/yr [15 mrem/yr] for this period} and is approximately 0.02 mSv/yr [2 mrem/yr] over the 1-million-year period {dose limit of 1.0 mSv/yr [100 mrem/yr] for this period}.

2.2.1.4.1.3.4.2 NRC Staff's Review of Statistical Stability of Average Annual Dose

The NRC staff reviewed the DOE TSPA model and analytic results, as well as the SAR, and finds the calculated overall average annual dose curve statistically stable and below the regulatory dose limit for the 10,000-year and 1-million-year periods because

- The overall average annual dose curve is reasonably stable with respect to the different approaches for representing epistemic and aleatory uncertainties (i.e., the average annual dose does not significantly change under the different approaches, for example see SAR Figures 2.4-38, 2.4-37 to 52, 2.4-55 to 61)

- The overall average annual dose curve is well below the regulatory limits (i.e., the estimated peak dose is approximately 50 times less than the regulatory limit for the initial 10,000 years and the period after 10,000 years in SAR Figure 2.4-10)

- Model updates from TSPA Model v5.000 to v5.005 caused only a moderate change in the magnitude of the overall average annual dose; the same conclusions with respect to average annual dose stability are expected to apply to both versions v5.000 and v5.005 {i.e.; the model updates do not cause different numerical model behavior in regard to statistical stability; for example, see SNL [2008ag, Figures 7.3.1-17(a) to 7.3.3-13(a)}

Finally, the NRC staff's confirmatory calculation provided further confidence that the DOE TSPA results were consistent with the model abstractions and capabilities of the barriers important to waste isolation described in the SAR. The NRC staff reviewed and found acceptable the model abstractions, including uncertainties, scenario probabilities, the technical basis for excluding FEPs, and the description of the capabilities of the barriers important to waste isolation (see SER Sections 2.2.1.1–2.2.1.3). DOE repository performance calculations showed significant margins to the regulatory dose limits. Further, the NRC staff's confirmatory calculations found that only a limited number of performance attributes (e.g., failure rate of the waste package, seepage flux into the waste package, solubility limits, and retardation in the saturated and unsaturated zones) had the potential to significantly alter the resulting average annual dose. As described in SER Section 2.2.1.4.3.3.1.2, the effect of uncertainties on the DOE average annual dose curve is limited. As addressed previously, DOE incorporated the model uncertainties into the analyses and the analyses were shown to converge to a stable solution. The NRC staff found the analytic models reviewed and approved in SER Sections 2.2.1.1 through 2.2.1.3 of this volume acceptable in that they were technically sound and provide an acceptable representation of repository performance (i.e., the radiological consequences for the Yucca Mountain facility would not be significantly underestimated).

2.2.1.4.1.4 Evaluation Findings

The NRC staff has reviewed the SAR and the other information submitted in support of the license application, which includes the information required by 10 CFR 63.21(c)(11), and finds, with reasonable expectation, that the requirements of 10 CFR 63.113(b) are satisfied. In particular

- The engineered barrier system is designed so that, working in combination with the natural barriers, the annual dose to the RMEI is below the postclosure individual protection standards during the first 10,000 years after permanent closure and for the period after the initial 10,000 years up to the period of geologic stability, consistent with the requirements at 10 CFR 63.303 and 63.311

- The ability of the geologic repository to limit radiological exposures demonstrated, through a performance assessment analysis, meets the requirements of 10 CFR 63.114 and 10 CFR 63.342, uses the reference biosphere defined in 10 CFR 63.305(a)–(d), uses the RMEI as defined in 10 CFR 63.312(a)–(e), excludes the effects of human intrusion, and constrains the performance assessment consistent with the requirements at 10 CFR 63.342

The NRC staff has reviewed the SAR and the other information submitted in support of the license application and finds that the requirements of 10 CFR 63.114 and the constraints on the performance assessment requirements at 10 CFR 63.342, including use of the reference biosphere [10 CFR 63.305 and the RMEI (10 CFR 63.312)] are satisfied. The technical requirements for conducting a performance assessment and the constraints for the performance assessment have been met, as documented in SER Sections 2.2.1.2 and 2.2.1.3. In particular

- Appropriate data from the site and surrounding region, uncertainties and variabilities in parameter values, and alternate conceptual models have been used in the analyses, in compliance with 10 CFR 63.114(a)(1–3)

- DOE considered only FEPs consistent with the limits on performance assessment specified at 10 CFR 63.342, in compliance with 10 CFR 63.114(a)(4)

- Specific FEPs have been included in the analyses, and appropriate technical bases have been provided for inclusion or exclusion in the performance assessment for the initial 10,000 years, in compliance with 10 CFR 63.114(a)(5)

- Specific degradation, deterioration, and alteration processes have been included in the analyses, taking into consideration their effects on annual dose for the first 10,000 years, and appropriate technical bases have been provided for inclusion or exclusion, in compliance with 10 CFR 63.114(a)(6)

- Adequate technical bases are provided for models used in the performance assessment for the first 10,000 years, as required by 10 CFR 63.114(a)(7)

Specific FEPs included in the performance assessment for the first 10,000 years have been included in the performance assessment for the period after 10,000 years subject to the following constraints:

(a) DOE has adequately considered the requirements in 10 CFR 63.342(c)(2), which allows climate change in the post-10,000-year period to be represented by a constant in time value for deep percolation, based on a lognormal distribution, truncated to vary between 10 and 100 mm/yr [0.39 and 3.9 in/yr] {arithmetic mean of 41 mm/yr [1.6 in/yr] and standard deviation of 33 mm/yr [1.3 in/yr]}. DOE used the percolation distribution from the draft rule [70 FR 53,313 (Sept. 8, 2005)] (SAR Section 2.3.2.4.1.2.4.2) because the final rule [74 FR 10,811 (Mar. 13, 2009)] was not promulgated until a few months before DOE submitted the license application. The NRC staff found the DOE approach for climate change in the post-10,000-year period to be acceptable (see SER Section 2.2.1.3.6).

(b) DOE has limited the seismic analyses to damage to the drifts in the repository, failure of the waste packages, and changes to the height of the water table below Yucca Mountain, as required by 10 CFR 63.342(c)(1).

(c) DOE has limited the analysis of igneous activity to those effects that damage the waste package directly, causing releases of radionuclides to the biosphere, atmosphere, or groundwater, as required by 10 CFR 63.342(c)(1).

(d) DOE included general corrosion in the performance assessment, as required by 10 CFR 63.342(c)(3).

DOE has used the characteristics of the reference biosphere and the RMEI specified at 10 CFR 63.305 and 10 CFR 63.312.

2.2.1.4.1.5 References

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CHAPTER 18

2.2.1.4.2 Demonstration of Compliance With the Human Intrusion Standard

2.2.1.4.2.1 Introduction

Safety Evaluation Report (SER) Section 2.2.1.4.2 provides the U.S. Nuclear Regulatory Commission (NRC) staff's review of the U.S. Department of Energy's (DOE's) Total System Performance Assessment (TSPA) calculation used to demonstrate compliance with the human intrusion standard, as described in DOE's Safety Analysis Report (SAR) Section 2.4.3 (DOE, 2008ab). The geologic record provides a basis for evaluating the likelihood of geologic processes and events, but there is no similar record of extended duration that can be used to constrain either the probability that human intrusion could occur or the characteristics of such intrusion. Regulations specify that the potential effects of human intrusion on waste isolation must be considered when evaluating repository performance. The NRC staff's review evaluates whether the repository can adequately perform if its barriers are breached by a human intrusion.

2.2.1.4.2.2 Regulatory Requirements

The applicant is required to provide information specified in 10 CFR 63.21(c)(13) to demonstrate compliance with 10 CFR 63.113(d). To evaluate human intrusion, the regulations establish dose limit requirements [10 CFR 63.321(b)], requirements specific to the human intrusion scenario [10 CFR 63.321(a) and 63.322], and requirements for conducting the performance assessment [10 CFR 63.113(d), 63.114, 63.303, and 63.342]. Accordingly, the U.S. Department of Energy (DOE or the applicant) must evaluate when a human intrusion might occur and the consequences of the human intrusion, in accordance with the previously noted regulatory requirements. In particular, the individual protection standard for human intrusion requires the applicant to

- Determine the earliest time after disposal that the waste package would degrade sufficiently that a human intrusion could occur without the drillers recognizing it [10 CFR 63.321(a)]

- Assume for the human intrusion scenario that (i) there is a single human intrusion as a result of exploratory drilling for groundwater, (ii) the intruders drill a borehole directly through a degraded waste package into the uppermost aquifer underlying the Yucca Mountain repository, (iii) the drillers use the common techniques and practices that are currently employed in exploratory drilling for groundwater in the region surrounding Yucca Mountain, (iv) careful sealing of the borehole does not occur—instead, natural degradation processes gradually modify the borehole, (v) no particulate waste material falls into the borehole, (vi) the exposure scenario includes only those radionuclides transported to the saturated zone by water (e.g., water enters the waste package, releases radionuclides, and transports radionuclides by way of the borehole to the saturated zone), and (vii) no releases are included that are caused by unlikely natural processes and events [10 CFR 63.322]

- Demonstrate that there is a reasonable expectation that the reasonably maximally exposed individual (RMEI) receives, as a result of the human intrusion, no more than the following annual dose: 0.15 mSv [15 mrem] for 10,000 years following disposal and 1.0 mSv [100 mrem] after 10,000 years but within the period of geologic stability [10 CFR 63.321(b)]

The NRC staff’s review followed the guidance in the Yucca Mountain Review Plan (YMRP) (NRC, 2003aa, Section 2.2.1.4.2) for the demonstration of compliance with the human intrusion standard. The acceptance criteria in the YMRP address the timing of an intrusion event, the representation of the human intrusion event in the total system performance assessment, and the annual dose to the reasonably maximally exposed individual.

In addition, the NRC staff has reviewed the applicant’s description of the human intrusion event as part of its review of events that were included in the performance assessment found in SER Section 2.2.1.2.2.3. The NRC staff reviewed the DOE definition of the human intrusion event to determine whether the event definition is unambiguous and consistent with regulatory requirements and, as described in SER Section 2.2.1.2.2.3, concludes DOE has adequately defined the human intrusion event.

2.2.1.4.2.3 Technical Review

The regulations require DOE to use a performance assessment to demonstrate compliance with the dose limits for human intrusion [10 CFR 63.342]; however, the human intrusion calculation is subject to specific requirements regarding the determination of the timing of the human intrusion and assumptions with respect to the nature and extent of the intrusion scenario. Accordingly, the performance assessment for the human intrusion scenario is somewhat different than the performance assessment used to demonstrate compliance with the individual protection standard. The NRC staff expects the two performance assessments to differ because the performance assessment used to evaluate the human intrusion scenario includes disruption of the repository due to a postulated human intrusion event as prescribed in 10 CFR 63.322. However, the NRC staff expects those portions of the performance assessment not affected by the regulatory specifications for the human intrusion scenario to be the same as the performance assessment used for demonstrating compliance with the individual protection standard (e.g., transport of radionuclides in the saturated alluvium and characteristics of the biosphere are not affected by the postulated human intrusion event). In the YMRP (NRC, 2003aa, p. 2.2-138, Acceptance Criterion 2), the NRC staff’s review includes a determination that the TSPA for human intrusion is identical to the TSPA for individual protection, except that it assumes the occurrence of the postulated human intrusion scenario prescribed by regulation. As a result, the NRC staff’s review of the applicant’s performance assessment for the human intrusion scenario evaluates (i) whether or not the performance assessment used for the human intrusion scenario is the same as the performance assessment used for individual protection (i.e., except for the required representation of the human intrusion scenario, there are no differences between the performance assessment used for demonstrating compliance with the individual protection dose limits and the performance assessment used for demonstrating compliance with the human intrusion dose limits that would result in a significant underestimation of the peak dose for the human intrusion scenario) and (ii) whether or not those portions of the performance assessment for the human intrusion scenario are adequately represented in the performance assessment, consistent with the specifications in 10 CFR 63.322. Those portions of the TSPA for human intrusion that are identical to the TSPA for individual protection (e.g., biosphere) are evaluated as part of the review of the TSPA for individual protection (SER Section 2.2.1.4.1) and are not duplicated in this section.

The NRC staff's evaluation involves reviewing DOE's SAR, Total System Performance Assessment Analysis Model Report, and the TSPA model files including intermediate results provided as part of the license application.

The NRC staff's review entails determining whether

- DOE's selection of the earliest time for the human intrusion to occur is adequately supported (SER Section 2.2.1.4.2.3.1)

- The performance assessment for the human intrusion calculation provides a credible representation of the human intrusion scenario (SER Section 2.2.1.4.2.3.2)

- Dose limits are met and statistically stable [e.g., increasing the number of simulations (statistical sample size) performed with DOE's TSPA model is not expected to significantly change the calculated average dose] (SER Section 2.2.1.4.2.3.3)

2.2.1.4.2.3.1 Timing of Human Intrusion Event

Description of DOE Approach

As specified in 10 CFR 63.321(a), DOE must determine the earliest time at which a driller would penetrate a waste package without recognition (e.g., that a metal object had been contacted rather than rock), which is referred to as the human intrusion event. In SAR Section 2.4.3.2 the applicant identified general corrosion as the process that, given sufficient time, could cause significant degradation of the drip shield and waste package such that drilling performance would most likely not be affected by the presence of the drip shield and waste package. The applicant determined that there is only a 0.0001 percent chance that the drip shield will fail by corrosion before approximately 230,000 years under the nominal scenario class, which represents "normal" conditions (see SER Section 2.2.1.4.1 for discussion of the nominal scenario). The applicant also determined that the waste package has only a 5 percent chance of failure (i.e., significant degradation or thinning of the walls of the waste package) from general corrosion prior to 600,000 years. On the basis of these results, the applicant selected 200,000 years as the earliest time the waste package would degrade sufficiently that a human intrusion could occur without the drillers recognizing it. The applicant considered this a conservative approach because the waste package is estimated to have experienced limited degradation due to corrosion (i.e., waste package to be substantially intact) by that time.

The applicant also evaluated other events that might affect the timing of the human intrusion event. As specified in 10 CFR 63.322(g), the applicant need not consider unlikely natural processes and events (i.e., those events with less than 1 chance in 100,000 per year of occurring) in the evaluation of human intrusion. The applicant evaluated the likelihood of early undetected defects, igneous events, and seismic events. For early undetected defects and igneous disruptive events, the applicant determined that the likelihood was less than the limit for likely events. For seismic events, the applicant determined that damage to either the drip shield or waste package that might compromise the structural integrity of the drip shield or waste package (e.g., rupture or framework buckling of the drip shield, punctures and ruptures of the waste package) is also less than the limit for likely events (SAR Section 2.4.3.2.2, pp. 2.4-303 and 304). For seismic damage that is considered likely to occur [i.e., stress corrosion cracking (SCC) of the drip shield or waste package], the applicant asserted such damage would not be sufficient to prevent the driller from recognizing that a metal object had been contacted (SAR Section 2.4.3.2.2, pp. 2.4-303 and 304).

NRC Staff's Evaluation

The NRC staff has evaluated the applicant's technical basis supporting the selection of the time of occurrence of human intrusion and finds it acceptable for the following reasons:

- General corrosion, which uniformly thins the entire surface of a material, represents a degradation process that could eventually thin or reduce the thickness of the outer barrier to the extent that a driller may not recognize the presence of a waste package. DOE has shown that (i) after 1 million years, approximately 10 percent of the waste packages, on average, are failed due to general corrosion of the waste package and (ii) prior to 400,000 years, less than 0.01 percent of commercial spent nuclear fuel waste packages and approximately 0.1 percent of codisposal waste packages are breached, as described in DOE (2009bj, Enclosure 1, Figures 9 and 10). Additionally, after 1 million years, approximately 1 percent of the surface area of the waste package is breached. The NRC staff's review of the DOE model for general corrosion of the waste package outer barrier that was implemented in the Total System Performance Assessment–License Application code found that the DOE model for general corrosion of the waste package outer barrier is acceptable and concluded that DOE provided adequate technical support for its calculations of the timing and magnitude of waste package breach by general corrosion (see SER Section 2.2.1.3.1.3.2.1).

- Stress corrosion cracking generally refers to a process whereby cracks form in metals or alloys in a corrosive environment and under sustained tensile stresses. Using the TSPA code, DOE calculated that even if there is sufficient stress to initiate and propagate cracks, the breached area of the waste package will be limited by the small crack size and density. The seismic ground motion modeling case, which included unlikely seismic events, resulted in less than 0.1 percent of the surface area damaged due to SCC over the 1-million-year period, as identified in DOE (2009bj, Enclosure 1, Figures 3, 4, 7, and 8). The NRC staff's review of the DOE models for stress corrosion cracking of the waste package outer barrier that were implemented in the TSPA code found that DOE acceptably accounts for stress corrosion cracking of the waste package outer barrier in the TSPA code (see SER Section 2.2.1.3.1.3.2.3). Given the limited surface area of the waste package affected by stress corrosion cracking, the NRC staff finds acceptable the assumption that such degradation would not prevent the driller from recognizing the presence of the waste package (i.e., general corrosion of the waste package rather than stress corrosion cracking could prevent the driller from doing so).

- DOE determined that the igneous intrusion is an unlikely event (i.e., estimated to have less than 1 chance in 100,000 per year of occurring and at least 1 chance in 100 million per year of occurring) and, therefore, is excluded from the analysis of human intrusion by 10 CFR 63.322(g). On the basis of its review of the license application, the NRC staff finds DOE's determination that igneous intrusion is an unlikely event acceptable because (i) the preponderance of information indicates that the mean annual probability for igneous disruption of the proposed repository by a basaltic dike (intrusive case) is on the order of 1 in 100 million per year to 1 in 10 million per year and (ii) mean probability values significantly higher (i.e., 1 in 1 million per year) or lower (i.e., 1 in 1 billion per year) than this range are not consistent with past patterns of activity in the Yucca Mountain region and therefore are not considered credible (see SER Section 2.2.1.2.2.3.1).

- Seismic damage to the waste package due to tensile tearing (i.e., rupture and puncture of the waste package that represent relatively large openings in the waste package) could damage significantly more surface area of the waste package than seismically induced stress corrosion cracking, which represents relatively small openings in the waste package (see SER Sections 2.2.1.4.1.3.3.1.1.2 and 2.2.1.4.1.3.3.1.1.3). However, DOE estimated that the probability of seismic events of sufficient magnitude to cause such damage is less than 1 chance in 100,000 per year and therefore designated seismic damage as an unlikely event that can be excluded from the human intrusion scenario. On the basis of its review of the license application, the NRC staff concluded that DOE's exclusion of waste package rupture was appropriately supported by kinematic analyses that considered the mechanical properties of the waste package, impact velocities of the waste package during seismic events, and degradation of drifts and drip shields (see SER Section 2.2.1.3.2.7).

- Early failure of the drip shield and waste package could be a factor affecting the timing of an intrusion event. DOE's estimate of the probability for early failure of a drip shield is less than 1 chance in 100,000, and therefore DOE designated it as an unlikely event that can be excluded from the human intrusion scenario. On the basis of its review of the license application, the NRC staff found acceptable the probability for drip shield failure (i.e., mean value of approximately 2 chances in 1 million) based on the methodology used to estimate the probability of damage to the drip shield and data from industrial analogues for fabrication and handling of the drip shield (see SER Section 2.2.1.2.2.4). DOE assumed the time of the human intrusion event occurs once the drip shield fails and did not take credit for any delay in the time for the human intrusion event due to the waste package being intact. The NRC staff finds this approach for consideration of early waste package failure acceptable because the human intrusion is conservatively assumed to penetrate the waste package at the same time the drip shield fails, and thus the waste package does not cause delay of the intrusion event.

2.2.1.4.2.3.2 Representation of Intrusion Event

Description of DOE Approach

The applicant developed a separate performance assessment to evaluate the consequences of a postulated human intrusion event assumed to occur 200,000 years after permanent closure. The key elements of the postulated human intrusion event are the effects of the borehole on seepage into the waste package, release of radionuclides from the waste package, and transport of radionuclides through the unsaturated zone to the saturated zone.

The performance assessment for individual protection is used for the human intrusion analysis, including the identical sampling approach for treating uncertainty (i.e., Latin hypercube sampling). The applicant modified its performance assessment for individual protection to represent human intrusion in a manner consistent with the regulatory requirements for the human intrusion scenario in 10 CFR 63.322. Specifically, the performance assessment for human intrusion

- Does not include unlikely events (e.g., igneous activity or faulting)

- Assumes that damage to a single waste package occurs at 200,000 years and is the result of drill bit penetration with a cross-sectional area of 0.0324 m^2 [0.349 ft^2]; the area is based on the cross section of a borehole with a diameter of 20.3 cm [8 in]

- Assumes that seepage water enters the waste package through the borehole

- Assumes the borehole is degraded and filled with rock debris

- Assumes that releases from the waste package are passed into a fracture pathway that is assumed to exist in the borehole all the way to the saturated zone (SAR p. 2.4-296)

- Assumes radionuclides move with the flowing water down the borehole fracture to the saturated zone and are slowed only due to matrix diffusion of dissolved radionuclides from the fracture into the rock matrix

- Considers only radionuclides transported by water from the waste package to the saturated zone by way of the borehole in the exposure scenario

The quantity of water that enters the waste package and matrix diffusion in the borehole are key aspects of the representation of the human intrusion event that affect the estimated doses (e.g., infiltration was identified as an important parameter affecting the expected dose in SAR Figure 2.4-173). The quantity of water that enters the waste package through the borehole affects the release of radionuclides that are solubility limited (e.g., the release of solubility-limited radionuclides such as Np-237 will commonly be proportional to the amount of water leaving the waste package). The applicant described how the amount of water that enters the waste package through the borehole is limited to the seepage entering the borehole (deep percolation is assumed to pass directly into the borehole opening). Other processes (e.g., drift seepage water splashing on the waste package surface and entering the waste package through the hole created by the borehole) were evaluated and determined to not significantly add to the quantity of water entering the borehole, as described in DOE (2009bj, Enclosure 4, Section 1.1). The applicant also described the basis for the process of matrix diffusion in the borehole, which can potentially delay both sorbing and nonsorbing radionuclide transport by providing a means for radionuclides to move from the relatively fast-flowing water in the borehole fracture into the slower moving water in the porous matrix of the rubble in the borehole. Although water in the borehole is estimated to take approximately 3 years to move through the unsaturated zone to the saturated zone, nonsorbing (I-129) and sorbing (Np-237) radionuclides are estimated to be delayed approximately 1,250 and 64,000 years, respectively, as outlined in DOE (2009bj, Enclosure 5, p. 8). The applicant described in DOE (2009cp, p. 2) that this effect is due to the large effective surface area for communication between the fracture and the matrix in the degraded borehole and along the borehole.

DOE also evaluated the potential effect on repository performance if the borehole penetrated a perched water zone (i.e., groundwater separated from an underlying body of groundwater by an unsaturated zone) below the repository. If radionuclides were present in a perched water zone, the borehole penetration of a perched water zone could potentially affect the transport of radionuclides from within the perched zone to the saturated zone. The applicant stated that the effect of the borehole would be limited because (i) perched water zones below the repository are isolated and have limited volume, as described in DOE (2009bj, Enclosure 6, p. 2); (ii) the significance of an equivalent 20.3-cm [8-in]-diameter borehole to capture and divert any lateral flow associated with the perched water is expected to be small because the area associated with fractures in the rock is more than 10,000 times greater than the area associated with the borehole, as identified in DOE (2009bj, Enclosure 6, p. 5); and (iii) the performance assessment already includes fast transport times in fault zones, which would not be significantly influenced by another fast pathway—namely, the borehole—as outlined in DOE (2009bj, Enclosure 6, p. 6).

NRC Staff's Evaluation

The NRC staff has evaluated the applicant's technical basis supporting its separate performance assessment for the postulated human intrusion event. The NRC staff concludes that the applicant's separate performance assessment for the human intrusion event provides a credible representation of the human intrusion event for the following reasons:

- **Assumptions for the method of transport from the waste package are acceptable**.

  The applicant's approach in SAR p. 2.4-295 provided for radionuclide release from the waste package directly following the assumed time of the human intrusion event. The releases from the waste package pass directly into the borehole and travel down the borehole via water in an assumed continuous fracture all the way to the saturated zone {i.e., via the flow of water in a single small fracture with an approximate fracture aperture of 3 mm [0.12 in]}, which was estimated to have an average water travel time through the unsaturated zone to the saturated zone of a few years, as described in DOE (2009bj, Enclosure 5, p. 6). The NRC staff finds the DOE's approach for the transport of radionuclides from a breached waste package acceptable because the assumed continuous fracture path provides (i) a consistent method of transport among the models of the TSPA that connects the releases from the waste package to the saturated zone and (ii) water travel times in this continuous fracture path on the order of a few years is a conservative estimate for travel from the waste package through the unsaturated zone to the saturated zone compared to the 100-year travel times the applicant has presented for fault zones, which are also characterized as a continuous "fracturelike" path through the unsaturated zone to the saturated zone (DOE, 2009bj, Enclosure 6, p. 6). Further, travel times on the order of years represent no significant delay for water down the borehole; thus, this value would not result in the dose being underestimated.

  Additionally, DOE determined the amount of seepage water that enters the waste package from the borehole, which provides the water flux for advective transport of radionuclides out of the waste package using the same deep percolation values from the performance assessment for individual protection and assuming that this seepage enters the cross-sectional area of the borehole, as outlined in SAR p. 2.4-317. The NRC staff concludes this approach, which assumes all water flowing downward in the borehole enters the waste package, is acceptable because it is consistent with the movement of deep percolation vertically downward toward the repository horizon. Additionally, as the applicant evaluated in DOE (2009bj, Enclosure 4, pp. 2–3), the potential for other adjacent seepage water entering the drift to enter the waste package is limited due, in part, to the limited distance that such seepage water "splashing" on the corroded waste package surface could travel and enter the borehole opening into the waste package.

- **Physical processes associated with the postulated human intrusion have been verified**.

  Matrix diffusion in the borehole is the primary means for delay of radionuclides transported from the waste package through the unsaturated zone to the saturated zone. The NRC staff finds that the applicant provided adequate support to verify its approach for matrix diffusion within the borehole by providing a comparison with an analytical solution, which had essentially an identical match between the human intrusion approach in the performance assessment and the analytical solution (DOE, 2009bj, Enclosure 5, pp. 8–11). The applicant also explained differences between matrix diffusion within the borehole and its approach for representing matrix diffusion within the unsaturated zone in the performance assessment for individual protection using the Active Fracture Model. The applicant explained in DOE (2009cp, Enclosure 1, p. 1) that the Active Fracture Model was developed for fracture networks rather than a single fracture (as in the borehole) and, therefore, is not appropriate for the human intrusion borehole pathway. The NRC staff finds DOE's approach of representing the borehole as a single fracture acceptable because of the limited diameter of the borehole and, as described under the previous bullet of this evaluation, the water travel time in the single fracture is on the order of a few years, representing no significant delay for water moving down the borehole.

- **The uncertainty in the results is consistent with the postulated intrusion event**.

  The results of the treatment of uncertainty, as displayed in the spread of dose curves in SAR Figures 2.4-11 and 2.4-159, are consistent with (i) the radionuclide inventory of the intruded waste package and (ii) the release and transport characteristics of the soluble, nonsorbing radionuclides (e.g., Tc-99 and I-129 are the main contributors to the peak dose shortly after the human intrusion) and less soluble, sorbing radionuclides (i.e., Pu-242 and Np-237, which are main contributors to dose long after the human intrusion occurs).

  Additionally, the NRC staff performed a confirmatory calculation to understand the magnitude of releases to the location of the RMEI due to the human intrusion event (NRC and CNWRA, 2014aa). Using the average dose curve and the average biosphere dose conversion factors from the applicant's performance assessment, the NRC staff calculated the magnitude of the releases for radionuclides most relevant to the dose calculation. For those radionuclides that do not sorb onto rock surfaces or corrosion products, especially Tc-99 and I-129, a very large fraction of the inventory for these radionuclides (on the order of 1 percent over a 100-year period) must be released to the RMEI location to produce the peak dose. Conversely, radionuclides that do sorb onto rock surfaces, especially Pu-242 and Np-237, release a much smaller fraction than 0.01 percent over a 100-year period to sustain the peak dose for these radionuclides. The results of this confirmatory calculation are consistent with the postulated human intrusion event and the relevant aspects of the performance assessment, including uncertainties, which the applicant used to demonstrate compliance with the individual protection requirements (see SER Section 2.2.1.4.1 for further details on the performance assessment used for individual protection).

- **The sampling method ensures the range for uncertain parameters are sampled**.

  The applicant used the Latin hypercube sampling (stratified Monte Carlo technique) for sampling uncertain parameters, which also was used for the performance assessment for individual protection. The NRC staff finds the DOE's approach to sampling adequate to ensure the sampled parameters were sampled across their range of uncertainty because (i) Latin hypercube sampling (stratified Monte Carlo technique) is a common sampling approach used in analyses involving uncertain parameters such as waste disposal, (ii) DOE considered alternative sampling combinations (called "replicates" in SAR Section 2.4.3.3.3) that resulted in nearly identical dose curves, and (iii) scatter plots (SAR Figures 2.4-174 and 2.4-176) showed that DOE's sampling approach produced sampled values over the ranges of uncertainty considered for the parameters [the scatter plots presented sampled values for an uncertain parameter (x-axis) versus the resulting dose for the TSPA simulation that used the specific value for the parameter (y-axis)].

  Although not part of the required assumptions of the human intrusion scenario in 10 CFR 63.322, the applicant evaluated the potential effects on performance from a borehole penetrating a perched water zone. The NRC staff evaluated the applicant-provided information to understand how a borehole might affect the transport of radionuclides from a perched water zone. The NRC staff finds acceptable the DOE's explanation that the potential for a borehole penetrating a perched water zone to significantly impact repository performance is limited because (i) a single, 20.3-cm [8-in]-diameter borehole (cross-sectional area less than 1/10 of a square meter or approximately 1/3 square foot) filled with rubble does not represent a significant feature that could divert significant water flow relative to the unsaturated zone flow already occurring in faults and fractures beneath the repository footprint of approximately 5.7 million m^2 [61 million ft^2] (SAR p. 2.3.1-85) [e.g., DOE estimated in DOE (2009bj, Enclosure 6, p. 5) that 30 percent of the total water flux below the repository reaches the water table via faults]; (ii) given the borehole would primarily affect water flow in the matrix (e.g., water flow in fractures and faults would not be that dissimilar from the continuous fracture path assumed for the borehole), the impact is limited due to the small amount of water that reaches the water table below Yucca Mountain via matrix flow [i.e., less than 20 percent of the flux at the water table is from matrix flow with the remaining flux coming from fractures and faults, as identified in DOE (2009bj, Enclosure 6, Table 1)]; and (iii) perched water is of somewhat limited areal extent {i.e., DOE estimated an equivalent area of 900 m^2 [9,700 ft^2] for the radius of a perched zone, as described in DOE (2009bj, Enclosure 6, p. 4), which occurs mostly in the northern part of the repository where fracture flow is more prevalent, as outlined in DOE (2009bj, Enclosure 6, p. 2)}.

2.2.1.4.2.3.3 Annual Dose to RMEI

Description of DOE's Approach

DOE presented the dose curve for the human intrusion scenario in SAR Figure 2.4-11. The peak of the mean dose curve is approximately 0.0001 mSv/yr [0.01 mrem/yr] shortly after the time of the intrusion (i.e., 200,000 years). DOE's estimated dose is on the order of 10,000 times less than the dose limit of 1 mSv/yr [100 mrem/yr] for the period after 10,000 years.

The applicant performed tests to determine the computational stability of the average dose curve used to demonstrate compliance with the dose limit for human intrusion (SAR Section 2.4.3.3.3). The tests (i) computed three replicates to allow for different combinations of sampled values over their parameter ranges (SAR Figure 2.4-160); (ii) increased the number of aleatory samples from 30 to 90 (SAR Figure 2.4-161), and (iii) refined the timestep scheme, as shown in SNL {2008ag, Figures 7.3.3-10[a] and 7.3.3-11[a]}. The applicant concluded that expected doses were relatively unaffected (i.e., stable) by changes in values of sampled parameters, sample size, and timestepping.

NRC Staff's Evaluation

The NRC staff finds that DOE's estimated peak dose of 0.0001 mSv/yr [0.01 mrem/yr] for the human intrusion scenario is acceptable for the following reasons:

- **The TSPA for the human intrusion scenario is performed separately for the TSPA for individual protection and meets the requirements of 10 CFR 63.114**.

  DOE developed a separate dose curve (SAR Figure 2.4-11) for the human intrusion scenario using the separate TSPA model described in SAR Section 2.4.3. The NRC staff finds the TSPA for the human intrusion scenario meets the requirements for performance assessments, specified at 10 CFR 63.114, relevant to the stylized human intrusion scenario because DOE (i) provided the technical basis for those attributes of the human intrusion scenario not specified by regulation (i.e., water entering the waste package through the borehole and transport with the borehole), and these were different from the performance assessment for individual protection; (ii) accounted for uncertainties in the representation of the human intrusion (see SER Section 2.2.1.4.2.3.2, NRC Evaluation, Item 2); (iii) excluded unlikely features, events, and processes (see SER Section 2.2.1.4.2.3.1, NRC Evaluation, Items 3, 4, and 5); and (iv) provided a comparison with an alternative model [i.e., the analytical solution of Sudicky and Frind described in DOE (2009bj, Enclosure 5, Figure 2)] to support the approach for matrix diffusion in the borehole.

- **The TSPA model for the human intrusion scenario assumes the characteristics specified at 10 CFR 63.322**.

  Those portions of the performance assessment that the applicant used to represent the human intrusion event incorporated the required specifications for the human intrusion scenario at 10 CFR 63.322 in that (i) there is a single exploratory borehole that intersects the waste package providing a conduit to the saturated zone, and as a result, water enters the waste package and transports radionuclides from the intersected waste package to the saturated zone (SAR pp. 2.4-293 to 2.4-298); (ii) the borehole is depicted as a 20.3-cm [8-in]-diameter borehole, which results in a cross-sectional area of 0.0324 m^2 [0.349 ft^2], based on current drilling practices, that is assumed to be filled with rubble of collapsed host rock and not carefully sealed, and particulate waste material does not fall into the borehole (SAR pp. 2.4-293 to 2.4-298); and (iii) unlikely features, events, and processes were excluded (see SER Section 2.2.1.4.2.3.1 for the review of the excluded events).

  Those portions of the performance assessment for human intrusion that have not been modified to specifically represent the human intrusion event are consistent with the performance assessment used to demonstrate compliance with individual protection.

The TSPA model for the human intrusion scenario and the performance assessment for individual protection are the same (i.e., any differences would not result in a significant underestimation of the peak dose for the human intrusion scenario) regarding the following relevant portions of the performance assessment used for individual protection: (i) releases from the waste package, where releases are affected by solubility limits; (ii) degradation rates for the waste forms and sorption onto corrosion products; (iii) radionuclide transport in the saturated zone; and (iv) the representation of the characteristics of the biosphere and RMEI, which includes biosphere dose conversion factors.

- **The estimate of the mean dose is statistically stable.**

  The NRC staff reviewed SAR Section 2.4.3 as well as the relevant information in SNL (2008ag, Section 7.3). The NRC staff finds that the applicant acceptably demonstrated the statistical stability of the expected dose because DOE considered a range of tests (i.e., different combinations of sampled values, increased aleatory sample size, and reduced timesteps), all of which resulted in dose curves that do not change the overall result that the peak dose is on the order of 0.0001 mSv/yr [0.01 mrem/yr] {see SAR Figures 2.4-160(a) and 2.4-161 and SNL [2008ag, Figures 7.3.3-10(a) and 7.3.3-11(a)]}.

- **The dose estimate is consistent with the overall repository performance and the assumed characteristics of the human intrusion scenario**.

  The human intrusion scenario occurs at 200,000 years and intercepts a single waste package. Thus, the NRC staff expects that any radionuclide with a radioactive half-life on the order of 20,000 years and less, without a long-lived parent radionuclide, would decay sufficiently prior to the intrusion event to significantly limit the contribution to dose. The DOE's TSPA results for the human intrusion scenario are consistent with the concept that radionuclides with long half-lives are expected to be the largest contributors to dose [i.e., significant contributors to peak dose are Tc-99 (half-life of 213,000 years), I-129 (half-life of 15.7 million years), Pu-242 (half-life of 376,300 years), Se-79 (half-life of 1.13 million years), Cs-135 (half-life of 2.3 million years), and Np-237 (half-life 2.14 million years), as shown in SAR Figure 2.4-159]. Additionally, both the human intrusion scenario (SAR Figure 2.4-159) and the waste package early failure modeling case [SAR Figure 2.4-18(b)] result in peak doses over the million-year period on the order of .0001 mSv/yr [0.01 mrem/yr], consistent with the characteristic that the human intrusion scenario considers one waste package and the waste package early failure modeling case considers approximately one failed waste package (see SER Section 2.2.1.4.1.3.2). Based on this information, the NRC staff finds that DOE's estimated dose for the human intrusion scenario is reasonable and consistent with the overall repository performance and the assumed characteristics of the postulated human intrusion scenario.

- **The annual dose curve meets the performance objectives at 10 CFR 63.321**.

  DOE estimated a peak dose of approximately 0.0001 mSv/yr [0.01 mrem/yr] (SAR Figure 2.4.11) near 200,000 years after repository closure, which is approximately 10,000 times less than the regulatory limit for the period after 10,000 years of 1.0 mSv/yr [100 mrem/yr]. The NRC staff finds that DOE's annual dose curve for the human intrusion scenario meets the regulatory dose limit at 10 CFR 63.321.

2.2.1.4.2.4 Evaluation Findings

The NRC staff has reviewed the SAR and the other information submitted in support of the license application, which includes the information required by 10 CFR 63.21(c)(13). The NRC staff finds with reasonable expectation that the requirements of 10 CFR 63.113(d) are satisfied. The requirements for demonstrating repository performance, in the event of limited human intrusion, have been met. In particular, the NRC staff finds, with reasonable expectation, the following:

- DOE's specification for the timing of the human intrusion event to occur at 200,000 years after permanent closure of the repository is acceptable.

- DOE has demonstrated that the performance assessment used to estimate the annual dose curve meets the requirements for the postulated human intrusion event, the results are statistically stable, and the peak of the dose curve is below the dose limit of 1 mSv/yr [100 mrem/yr] as required by10 CFR 63.303 and 63.321. In particular, DOE estimates the peak dose to be approximately 0.0001 mSv/yr [0.01 mrem/yr], which is nearly 10,000 times less than the regulatory limit (SAR Figure 2.4-11).

- DOE has demonstrated that the performance assessment meets the relevant requirements in 10 CFR 63.114 and 63.342 and acceptably represents the human intrusion scenario as required by 10 CFR 63.322.

2.2.1.4.2.5 References

DOE. 2009bj. "Yucca Mountain—Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.4.4) Safety Evaluation Report Vol. 3, Chapters 2.2.1.4.1, 2.2.1.4.2, and 2.2.1.4.3, Set 1." Letter (July 29) J.R. Williams to J.H. Sulima (NRC). ML092110472. Washington, DC: DOE, Office of Technical Management.

DOE. 2009cp. "Yucca Mountain—Supplemental Response to Request for Additional Information Regarding License Application (Safety Analysis Report Section 2.4.3), Safety Evaluation Report Vol. 3, Chapter 2.2.1.4.2, Set 1." Letter (October 20) J.R. Williams to J.H. Sulima (NRC). ML092940188. Washington, DC: DOE, Office of Technical Management.

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CHAPTER 19

2.2.1.4.3 Demonstration of Compliance With Separate Groundwater Protection Standards

2.2.1.4.3.1 Introduction

Safety Evaluation Report (SER) Section 2.2.1.4.3 provides the U.S. Nuclear Regulatory Commission (NRC) staff's evaluation of the U.S. Department of Energy's ("DOE" or "applicant") calculation used to demonstrate compliance with the separate standards for protection of groundwater—an important source of drinking water. The NRC's regulations provide separate standards to protect the groundwater resources in the vicinity of Yucca Mountain and specify the approach to be taken to estimate the concentration of radionuclides in groundwater. This approach for groundwater protection (10 CFR 63.331) is similar to that used in estimating dose to the reasonably maximally exposed individual (RMEI) (10 CFR 63.311, 63.312). There are three distinct groups of radionuclides evaluated under groundwater protection: (i) radionuclides that are characterized as alpha emitters (e.g., Np-237) (this group explicitly excludes radon and uranium); (ii) radionuclides that are characterized as beta- and photon-emitting radionuclides (e.g., I-129, Tc-99); and (iii) the combined concentration of Ra-226 and Ra-228 released from the repository and the natural background levels of Ra-226 and Ra-228 in the groundwater. There are a number of similarities in the performance assessment used for demonstrating compliance with the individual protection standard and the performance assessment used to demonstrate compliance with the separate groundwater protection standards, including weighting the results by the probability of occurrence.

2.2.1.4.3.2 Regulatory Requirements

The applicant is required to provide information specified in 10 CFR 63.21(c)(12) to demonstrate compliance with 10 CFR 63.113(c). Separate groundwater protection standards for the initial 10,000 years after closure of the repository are in 10 CFR 63.331. The regulations also specify constraints for the performance assessment used to demonstrate compliance with the groundwater protection standards at 10 CFR 63.332 and requirements for conducting a performance assessment in 10 CFR 63.113(c), 63.114, 63.303, and 63.342.

Specific regulatory requirements related to DOE's demonstration of compliance with separate groundwater protection standards follow:

Performance Assessment for Groundwater Protection

- The performance assessment must be conducted in accordance with the general requirements for the performance assessment covering the initial 10,000 years specified in 10 CFR 63.114, 63.303, and 63.342.

Representative Volume

- The representative volume is the volume of groundwater that would be withdrawn annually from an aquifer containing less than 10,000 mg of total dissolved solids per liter of water [10 CFR 63.332(a)].

- DOE must determine the concentration of radionuclides that will be released from the Yucca Mountain repository that will be in the representative volume of groundwater for comparison with the separate groundwater protection standards [10 CFR 63.332(a)].

- DOE must determine the position and dimensions of the representative volume using average hydrologic characteristics, which must include the highest concentration level in the plume of contamination in the accessible environment [10 CFR 63.332(a)(1 and 2)].

- The representative volume contains 3,000 acre-ft [3.7 X 10^9 L] of water [10 CFR 63.332(a)(3)].

Separate Standards for Groundwater Protection (10 CFR 63.331, Table 1)

- The combined concentration of Ra-226 and Ra-228 from repository releases cannot exceed 5 pCi/L (including natural background radiation presently in groundwater at Yucca Mountain).

- For gross alpha activity (including Ra-226 but excluding radon and uranium), the combined concentration from repository releases and natural background radiation presently in groundwater at Yucca Mountain must be less than 15 pCi/L. (Np-237 is an example of an alpha-emitting radionuclide.)

- The combined concentration of beta- and photon-emitting radionuclides from repository releases cannot exceed 0.04 mSV [4 mrem] per year to the whole body or any organ {on the basis of drinking 2 L [0.53 gal] of water per day from the representative volume}. (Tc-99 and I-129 are examples of beta- and photon-emitting radionuclides.)

- DOE must determine background concentrations of specific radionuclides in groundwater as identified previously for Ra-226, Ra-228, and gross alpha activity.

The performance assessment for compliance with the individual protection standard is consistent with the performance assessment used to evaluate compliance with the groundwater protection standards (i.e., differences are due to the regulatory requirement that unlikely events are not to be included in the performance assessment used for groundwater protection). As a result, the NRC staff review of the applicant's groundwater protection analysis focused on DOE's determination of the representative volume and compliance with the separate limits specified for groundwater protection. The performance assessment for individual protection is reviewed in SER Section 2.2.1.4.1.

The NRC staff's review of the SAR and supporting information follows the guidance in the Yucca Mountain Review Plan (YMRP), NUREG–1804 (NRC, 2003aa), Section 2.2.1.4.3, Analysis of Repository Performance that Demonstrates Compliance with the Separate Ground-Water Protection Standards, as supplemented by additional guidance by NRC (2009ab). The YMRP acceptance criteria that provide guidance for the NRC staff's review are

1. An Adequate Demonstration is Provided That the Expected Concentration of Combined Radium-226 and Radium-228, Expected Concentration of Specified Alpha-Emitting Radionuclides, and Expected Whole Body or Organ-Specific Doses from any Photon- or Beta-Emitting Radionuclides at Any Year During the Compliance Period Do Not Exceed the Separate Ground-Water Protection Standards.

2. The Methods and Assumptions Used to Determine the Position of the Representative Volume of Ground Water are Credible and Consistent, and the Representative Volume of Ground Water Includes the Highest Concentration Level in the Plume of Contamination in the Accessible Environment.

3. The Methods and Assumptions Used to Calculate the Physical Dimensions of the Representative Volume of Ground Water are Credible and Consistent.

In its review of the SAR and supporting information, the NRC staff used a risk-informed approach, and guidance provided by the YMRP, as supplemented by NRC (2009ab), to evaluate DOE's compliance with the ground-water protection standards. The NRC staff considered all three YMRP acceptance criteria in its review of information provided by DOE. In the context of these criteria, only those aspects of the model that substantively affect the performance assessment results, as determined by the NRC staff, are discussed in detail. The NRC staff's determination is based both on risk information provided by DOE and on the NRC staff's knowledge gained through experience and independent analyses. Further, because DOE assumed that all radionuclides which reach the accessible environment in a given year are included in the annual water demand of 3,000 acre-ft [3.7×10^9 L], the NRC staff conducted a simplified review consistent with YMRP Section 2.2.1.4.3.1.

2.2.1.4.3.3 Technical Review

The NRC staff review of DOE's demonstration of compliance with the separate standards for groundwater protection focused on those portions of the analysis that are distinct to the groundwater protection analysis. Specifically, the NRC staff review focused on DOE's approach for including the highest concentration level of the plume in the representative volume, the dimensions of the representative volume, and comparison of the performance assessment results with the separate standards for groundwater protection.

2.2.1.4.3.3.1 Representative Volume Location

Description of DOE Approach

DOE used the same performance assessment model for evaluating compliance with the separate groundwater protection standards as it used for the individual protection standards in the sense that the model abstractions for flow paths in the saturated zone and radionuclide transport in the saturated zone are the same. However, DOE excluded the consideration of unlikely FEPs from the performance assessment used for groundwater protection (i.e., igneous activity and low probability seismic events are excluded). The location of the representative volume of groundwater was consistent with the approach used for determining the pathway for radionuclide transport to the location of the RMEI which is approximately 18 km [11 mi] south of the repository, as identified in the Safety Analysis Report (SAR) (DOE, 2008ab, Volume 2, p. 2.1-1). Additionally, DOE used the same approach for determining the concentration of radionuclides in groundwater for demonstrating compliance with both the separate groundwater protection and individual protection standards [i.e., the annual average radionuclide concentration, due to releases from the repository, was determined by assuming that all radionuclides that reach the compliance location in a given year are included in 3,000 acre-ft [3.7×10^9 L], which is the annual water demand for the individual protection standard and the representative volume for the separate groundwater standard (SAR Section 2.4.4)].

NRC Staff's Evaluation

The NRC staff has reviewed SAR Section 2.4.4 and concludes that the applicant's approach for determining the location of the representative volume and including the highest concentration within the plume in the accessible environment is acceptable and consistent with the YMRP. Specifically, the NRC staff finds the following.

- The location of the representative volume is approximately 18 km [11 mi] south of the repository. This location is based on DOE's specification for the postclosure controlled area, which extends the southern boundary of the controlled area to 36°40'13.6661" north latitude, consistent with the definition of the controlled area specified in 10 CFR 63.302 (SAR Volume 2, p. 2.4-7). Thus the location of the representative volume is in the accessible environment (defined in 10 CFR 63.302 as being outside the controlled area) immediately outside the controlled area.

- The location of the representative volume has been determined consistent with the radionuclide transport paths in the performance assessment used to demonstrate compliance with the individual protection standard because the same radionuclide transport paths were used in the performance assessment for individual protection and the performance assessment for groundwater protection (SAR Volume 2, p. 2.4-337).

- The location of the representative volume ensures that all radionuclides released to the accessible environment are considered in the assessment because DOE's Total System Performance Assessment (TSPA) assumes all radionuclides are captured in the representative volume (i.e., "total radionuclide capture," SAR Volume 2, p. 2.4-337).

- The highest concentration level of radionuclides in the plume of contamination in the accessible environment is included in the representative volume because all radionuclides (i.e., the entire plume of contamination, which includes the highest concentration level of radionuclides) released into the accessible environment are included in the representative volume that is annually withdrawn (SAR Volume 2, p. 2.4-337).

2.2.1.4.3.3.2 Representative Volume Dimensions

Description of DOE Approach

DOE estimated the dimensions of the representative volume using saturated zone models and assumptions that were also used for determining compliance with the individual protection standards. DOE estimated, using the slice of the plume method, that dimensions of a width of 3,000 m [9,842 ft], a depth of 200 m [656 ft], and a length of 30 m [98 ft] in the direction of groundwater flow would include all the simulated flow paths of radionuclides crossing the compliance location into the accessible environment (SAR Volume 2, p. 2.4-337). DOE estimated these dimensions using average properties for hydrologic parameters such as groundwater flow rate and alluvium flow porosity (SAR Volume 2, p. 2.4-337), and these dimensions were used to calculate the physical dimensions of the aquifer necessary to contain the representative volume of groundwater.

DOE also presented a more detailed depiction of the cross section of the plume at the compliance location to further support the dimensions of the representative volume. The more detailed analysis was based on numerous particle tracks, provided in

DOE (2009bj, Enclosure 7, Figure 1), representing potential release points for repository releases to the saturated zone using the saturated zone site-scale flow model. Although the cross section of the plume, based on the particle traces, is not a rectangular shape, DOE estimated that a rectangular shape of approximately 3,300 m [10,827 ft] in width (horizontally) by 220 m [722 ft] in depth (vertically) would enclose the horizontal and vertical extent of the plume cross section {an area of 726,000 m^2 [7.8 million ft^2]}. DOE estimated that approximately 40 percent of this rectangular shape did not contain any significant portion of the plume; thus, DOE estimated a cross-sectional area of 435,000 m^2 [4.7 million ft^2] given the irregularities of the shape produced by the particle traces depicted in DOE (2009bj, Enclosure 7, Figure 1). DOE's simple rectangular approximation {i.e., 3,000 by 200 m [9,842 by 656 ft]} results in a cross-sectional area of 600,000 m^2 [6.4 million ft^2], which provides a value between the two values calculated from the particle tracks {one a rectangular shape of 726,000 m^2 [7.8 million ft^2] and the other an irregular shape of 435,000 m^2 [4.7 million ft^2]}. The third dimension of the representative volume was selected to obtain the volume of 3,000 acre-ft [3.7×10^9 L], as specified in 10 CFR 63.332(a)(3). DOE calculated the third dimension, or the length parallel to the flow direction (i.e., perpendicular to the cross section), to be approximately 34.4 m [113 ft] on the basis of the cross-sectional area of 600,000 m^2 [6.4 million ft^2] and an average effective porosity of 0.18, as identified in DOE (2009bj, Enclosure 7, p. 4).

DOE used water quality data from the Alluvial Testing Complex (SAR Volume 2, p. 2.4-334) to determine that there were fewer than 500 mg/L [500 ppm] of total dissolved solids in the aquifer at the compliance location.

NRC Staff's Evaluation

The dimensions of the representative volume are required to include the highest concentration level in the plume of contamination [10 CFR 63.332(a)(1)]. The NRC staff reviewed SAR Section 2.4.4 and determined that the dimensions of the representative volume are acceptable and consistent with the YMRP because (i) the dimensions are sufficient to capture the entire radionuclide plume and thus include the highest concentration levels in the plume and (ii) the dimensions are supported by particle tracks that used the hydrologic characteristics of the site and releases from the engineered barrier system.

Specifically, the NRC staff finds the following.

- The representative volume of groundwater analyzed by DOE is within an aquifer containing fewer than 10,000 mg/L of total dissolved solids and no more than 3,000 acre-ft [3.7×10^9 L]

- DOE estimated the dimensions of the representative volume on the basis of the slice of the plume method specified in 10 CFR 63.332(b)(2)

- DOE used (i) average hydrologic characteristics representative of the aquifers along the flow paths in the saturated zone and (ii) the flow paths predicted by the saturated zone site-scale flow model used for the performance assessment (SAR Volume 2, p. 2.4-337)

- The representative volume of groundwater of 3,000 acre-ft is consistent with the water usage of the RMEI (i.e., annual water demand of 3,000 acre-ft [3.7×10^9 L])

- The dimensions of the representative volume (i) do not exclude any radionuclides from the estimate of the concentration of radionuclides in the representative volume (i.e., all radionuclides are assumed to lie within the dimensions of the representative volume that is annually withdrawn) and (ii) are reasonably consistent with the estimated shape of the contaminant plume

Additionally, DOE used a particle tracking approach to support the dimensions of the representative volume. The NRC staff evaluated DOE's particle tracking approach and concludes that the DOE approach is acceptable and consistent with the YMRP.

Specifically, the NRC staff finds the following.

- DOE's particle tracking approach released particles over the entire repository footprint, which provided the initial areal extent of the potential plume. The NRC staff finds that this initial release area is consistent with the performance of the repository regarding the potential for damaged packages over the entire footprint. Further, the NRC staff finds that the performance assessment for demonstrating compliance with the groundwater protection requirements is primarily influenced by the seismic ground motion modeling case (see SAR Figure 2.4-181). DOE presented information in the SAR that shows the seismic ground motion modeling case results in a significant number of codisposal waste packages (e.g., hundreds of waste packages) being breached due to stress corrosion cracks prior to 10,000 years (see SAR Figures 2.4-19 and 2.4-77). SER Sections 2.2.1.3.2 and 2.2.1.4.1 further detail the extent of damage to codisposal waste packages in the seismic ground motion modeling case.

- DOE's particle tracking approach used the saturated zone site-scale model consistent with the performance assessment abstraction for flow paths in the saturated zone (see SER Section 2.2.1.3.8 regarding NRC staff review of the saturated zone site-scale flow model). Thus, the spreading of the plume during transport to the accessible environment uses the same hydrologic characteristics reviewed and accepted for the performance assessment used for individual protection (see SER Sections 2.2.1.3.8 and 2.2.1.3.9).

- The two values DOE estimated from the detailed analysis for the cross-sectional area of the representative volume {one a rectangular shape of 726,000 m^2 [7.8 million ft^2] and the other an irregular shape of 435,000 m^2 [4.7 million ft^2]} bound the value of 600,000 m^2 [6.4 million ft^2] DOE specified for the representative volume. Given dimensions of this magnitude [i.e., hundreds of thousands of square meters (millions of square feet)], it is reasonable to assume a significant portion of the releases of radionuclides into the accessible environment would be captured in the representative volume. DOE assumed all the radionuclides are released into the representative volume; thus, the concentration of radionuclides in the representative volume did not change based on changes to the dimensions of the representative volume for the range of values estimated from the detailed analysis. (As noted previously, the representative volume is specified by regulation to contain 3,000 acre-ft [3.7×10^9 L] of water that is withdrawn annually.)

2.2.1.4.3.3.3 Concentration of Radionuclides in the Representative Volume

Description of DOE Approach

DOE determined the average concentration of radionuclides, due to repository releases, by assuming the annual releases of radionuclides were all included in the representative volume of 3,000 acre-ft [3.7×10^9 L] and determined the dose to the whole body and individual organs for the beta- and photon-emitting radionuclides on the basis of drinking 2 L [0.53 gal] per day of water at the concentration level estimated for the representative volume (SAR Section 2.4.4.1.1.4). DOE also estimated the natural background level of radioactivity presently in the groundwater at Yucca Mountain for Ra-226, Ra-228, and the alpha-emitting radionuclides, excluding radon and uranium (SAR Section 2.4.4.1.1.3).

DOE estimated the combined concentrations for Ra-226 and Ra-228, due to releases from the repository and the natural background radiation presently in the groundwater at Yucca Mountain, was 0.5 pCi/L, with the largest contribution coming from natural background radiation (the largest annual release of Ra-226 and Ra-228 into the representative volume from the repository was estimated to be almost 1 million times less than the natural background level).

DOE estimated the concentration for the gross alpha activity, due to releases from the repository and the natural background radiation presently in the groundwater at Yucca Mountain (excluding radon and uranium), was 0.5 pCi/L, with the largest contribution coming from natural background radiation (the largest annual release of the relevant alpha-emitting radionuclides into the representative volume from the repository was estimated to be more than 1,000 times less than the natural background levels).

DOE estimated the dose from beta- and photon-emitting radionuclides, due to releases from the repository, to be 0.0006 mSv/yr [0.06 mrem/yr] for the whole body and the largest dose to any organ to be 0.0026 mSv/yr [0.26 mrem/yr] (e.g., dose to the thyroid from I-129) as result of drinking 2 L [0.53 gal] of water per day assumed to be at a concentration level of radionuclides in the representative volume. (Natural background radiation is not considered for beta- and photon-emitting radionuclides in the separate groundwater protection standards; see 10 CFR 63.331, Table 1.)

NRC Staff's Evaluation

The NRC staff has reviewed SAR Section 2.4.4 and concludes that DOE's analysis that the level of radioactivity in the representative volume is below the separate groundwater protection limits (10 CFR 63.331) is acceptable and consistent with the YMRP.

Specifically, the NRC staff finds the following.

- The performance assessment used for demonstrating compliance with the individual protection requirements was consistent with the performance assessment used for demonstrating compliance with the separate groundwater protection limits (see SER Section 2.2.1.4.1 for details regarding the review of the performance assessment used for individual protection)

- Unlikely FEPs are excluded from the performance assessment used to demonstrate compliance with the separate standards for the groundwater protection requirements (SAR Volume 2, p. 2.4-328) in compliance with 10 CFR 63.342(b)

- The effects of human intrusion are not included in the performance assessment used for demonstrating compliance with the separate standards for the protection of groundwater (i.e., undisturbed performance was evaluated) (SAR Volume 2, p. 2.4-329)

- The average concentrations from repository releases are consistent with the performance assessment used to demonstrate compliance for individual protection for the initial 10,000 years (e.g., number and types of waste package failures) and the specific limits on the performance assessment used to assess groundwater protection (e.g., exclusion of unlikely FEPs) (see SER Section 2.2.1.4.1 for further details)

- The average concentrations from repository releases are determined by dividing the annual flux of radionuclides crossing the accessible environment boundary by the representative volume of 3,000 acre-ft [3.7 X 10^9 L] that is withdrawn annually (SAR Volume 2, p. 2.4-329)

- DOE estimated the mean natural background activity concentration for the combined Ra-226 and Ra 228 and for the relevant alpha-emitting radionuclides (i.e., excluding radon and uranium) using samples collected in the vicinity of the accessible environment boundary (SAR Section 2.4.4.1.1.3) and other locations

- Dose estimates for beta- and photon-emitting radionuclides consider the highest dose to the whole body or any organ on the basis of drinking 2 L [0.53 gal] per day from the representative volume

- The estimated releases from the repository to the accessible environment and the natural background radiation presently in the groundwater at Yucca Mountain (excluding radon and uranium) result in concentrations and doses that are all below the regulatory limits and, for most radionuclides, significantly below the regulatory limits

2.2.1.4.3.4 Evaluation Findings

The NRC staff reviewed DOE's SAR and other information submitted in support of the license application, which includes information required by 10 CFR 63.21(c)(12). The NRC staff finds, with reasonable expectation, that the requirements of 10 CFR 63.113(c) are satisfied. In particular, the NRC staff finds that

- The average concentrations of combined Ra-226 and Ra-228, gross alpha activity (including Ra-226 but excluding radon and uranium), and combined beta- and photon-emitting radionuclides are below the limits required by 10 CFR 63.303 and 63.331.

- The representative volume of groundwater is within an aquifer containing less than 10,000 mg of total dissolved solids per liter of water to meet a given water demand. Average hydrologic characteristics that are consistent with the repository performance assessment calculations are used to determine the position and dimension of the groundwater aquifers and project average radionuclide concentrations for the representative volume such that the highest concentration levels in the contaminant plume are included. The representative volume contains 3,000 acre-ft [3.7 × 10^9 L] and meets the requirements specified in 10 CFR 63.332(a)(1and 3).

- The dimensions of the representative volume of groundwater are calculated using one of the alternative methods specified in 10 CFR 63.332(b)(1 and 2).

- DOE's performance assessment used for demonstrating compliance with the separate groundwater protection standards satisfies the relevant requirements at 10 CFR 63.114 and 63.342.

2.2.1.4.3.5 References

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CHAPTER 20

2.5.4 Expert Elicitation

2.5.4.1 Introduction

Safety Evaluation Report (SER) Section 2.5.4 evaluates the information provided in the U.S. Department of Energy (DOE) Safety Analysis Report (SAR) for uses of expert elicitation. The applicant's uses are described in SAR Section 5.4 (DOE, 2009av).

Expert elicitation is a formal, structured, and well-documented process for obtaining the judgments of multiple experts. The U.S. Nuclear Regulatory Commission (NRC) routinely accepts, for review, expert judgments used to evaluate and interpret the factual bases of safety analyses. The NRC staff recognizes that DOE could elect to use the subjective judgments of experts, or groups of experts, to interpret data and address technical issues and inherent uncertainties when assessing the long-term performance of a geologic repository. In its SAR, DOE used the results of three formal expert elicitations to complement and supplement other sources of scientific and technical information, such as data collection, analyses, and experimentation. The NRC staff reviewed DOE's use of expert elicitation regarding the proposed geologic repository at Yucca Mountain.

In supporting its SAR, DOE presented the results of three expert elicitations in the areas of seismic hazard (SAR Section 2.2.2.1), igneous activity (SAR Sections 1.1.6.2, 2.2.2.2, and 2.3.11), and saturated zone flow and transport (SAR Section 2.3.9.2). SAR Section 5.4 summarized DOE's bases for its assertion that these elicitations were conducted in a manner that is consistent with NRC guidance on this subject. In conducting its review of DOE's use of expert elicitation, the NRC staff sought to verify that DOE followed the process suggested in NUREG–1563 (NRC, 1996aa), or equivalent stepwise process, such as the detailed seismic hazard analyses outlined in NUREG/CR–6372 (NRC, 1997aa), or the NUREG/CR–6372 implementation guidance described in NUREG–2117 (NRC, 2012aa).

2.5.4.2 Regulatory Requirements and Guidance

The regulatory requirement in 10 CFR 63.21(c)(19) provides that the SAR must include an explanation of how expert elicitation was used. In 1996, the NRC staff published guidance for the use of expert elicitation in NUREG–1563 (NRC, 1996aa). NUREG–1563 provides general guidelines for deciding whether a formal expert elicitation would be useful and suggests a nine-step procedure that could serve as one acceptable process to conduct an elicitation. The guidance explicitly states that the suggested procedure was not provided with the intent that it be rigidly applied. Rather, the guidance in NUREG–1563 (NRC, 1996aa, p. 22) provides that the suggested procedure "…should be viewed as a general framework for a formal elicitation that would be acceptable to the NRC staff."

Subsequent to the publication of NUREG–1563, the NRC staff published NUREG/CR–6372 (NRC, 1997aa). This document, referred to informally as the Senior Seismic Hazard Analysis Committee (SSHAC) report, or the SSHAC guidelines, provided a process for obtaining, communicating, and quantifying the uncertainties associated with elicitation received from seismic experts in the course of conducting Probabilistic Seismic Hazard Assessments (PSHAs) for commercial nuclear power plants and other critical facilities. NUREG–2117 (NRC, 2012aa), which was published in April 2012, supplements the SSHAC guidance with practical implementation guidance based on experience gained from past SSHAC projects.

The stepwise processes for eliciting experts described in NUREG/CR–6372 and NUREG–2117 for the most formal (Level 4) analysis, and those recommended in NUREG–1563, are very similar. While presented in a slightly different order and structure (in seven steps [NUREG/CR–6372 and NUREG–2117], as opposed to nine steps [NUREG–1563]), the three documents recommend essentially the same approach for formally eliciting and documenting expert opinion. For example, the important content identified in NUREG–1563 as Step 4, “Assembly and Dissemination of Basic Information;” Step 6, “Elicitation of Judgments;” and Step 7, “Post-Elicitation Feedback” is not treated as discrete steps in the updated SSHAC guidelines in NUREG/CR–6372 and NUREG–2117. Instead, the SSHAC guidelines encompass the substance of all three in a single Step 5, referred to as “Group Interaction and Individual Elicitation.”

The NRC staff’s review of DOE’s use of expert elicitation was guided by the Yucca Mountain Review Plan (YMRP) Section 2.5.4 (NRC, 2003aa). YMRP Section 2.5.4.3 identifies two acceptance criteria: (i) that the applicant use NUREG–1563 or equivalent procedures and (ii) that any updated elicitations follow appropriate methods and are adequately documented. The NRC staff evaluated the techniques DOE used to conduct three expert elicitations to verify whether these followed procedures suggested by the NRC staff’s guidance or used other equivalent procedures. DOE updated only one of the three elicitations. The NRC staff evaluated the methods DOE used to update that elicitation to verify whether it was updated appropriately and adequately documented.

2.5.4.3 Technical Review

This section briefly summarizes the information provided in SAR Section 5.4 for each of the three expert elicitations the applicant used. The discussion at the end of this section provides the NRC staff’s evaluation of DOE’s SAR using the two acceptance criteria given in YMRP Section 2.5.4.4.

Probabilistic Volcanic Hazard Assessment (PVHA) Expert Elicitation

SAR Section 2.2.2.2 describes the approach used by DOE to develop a volcanic hazard assessment for Yucca Mountain. This overall approach included an expert elicitation to develop a PVHA for Yucca Mountain. DOE conducted the expert elicitation in 1995 and published the final report in 1996 (CRWMS M&O, 1996aa). SAR Section 5.4.1 summarized the applicant’s bases for how its PVHA was conducted in a manner generally consistent with the nine-step procedure suggested in NUREG–1563 (NRC, 1996aa).

For PVHA, DOE selected 10 subject matter experts to assess the relevant technical issues, including a range of conceptual and probability models; associated uncertainties in model parameters; and model sensitivity to these uncertainties. The elicitation consisted of four workshops and two field trips to the Yucca Mountain area. Each panel member made an individual assessment or model of the igneous hazard on the basis of his or her interpretations of various probabilistic models. A logic tree approach was used to combine alternatives and to incorporate uncertainty. The 10 experts’ probability estimates were then combined with equal weight to produce a probability distribution of the annual frequency of intersection of a basaltic dike within the proposed repository footprint.

Subsequent to the PVHA, DOE conducted an aeromagnetic survey and drilling program to increase confidence in site characterization results related to igneous activity. DOE used the results of this program to update the PVHA in a study published as the probabilistic volcanic hazard assessment-update (PVHA-U) (SNL, 2008ah). In SAR Section 5.4.1, DOE stated that the PVHA-U was conducted in a manner that is consistent with NUREG–1563 (NRC, 1996aa) and past practices. DOE also stated that the difference between the PVHA and the updated PVHA (PVHA-U) would not significantly affect the estimates of repository performance over either 10,000 years or 1 million years, and that the PVHA-U results are confirmatory of the original PVHA technical basis (DOE, 2009av; also see Boyle, 2008aa).

The NRC staff's technical evaluation of the volcanic hazards information used to support the PVHA expert elicitation is provided in SER Section 2.2.1.2.2.3.1.

Probabilistic Seismic Hazard Assessment (PSHA) Expert Elicitation

SAR Section 2.2.2.1 describes DOE's overall approach to develop a seismic hazard assessment for Yucca Mountain, including fault displacement hazards. This approach included an expert elicitation to develop a PSHA for Yucca Mountain (CRWMS M&O, 1998aa; BSC, 2004bj). DOE conducted its PSHA in the late 1990s using a methodology that DOE claims is consistent with a Level 4 expert elicitation as described in NUREG/CR–6372 (NRC, 1997aa). SAR Section 5.4.2 summarized DOE's bases and concluded that this methodology was also generally consistent with the nine-step procedure suggested in NUREG-1563 (NRC, 1996aa).

DOE's PSHA also followed the standard framework for PSHAs in using the recurrence curve approach (e.g., Cornell, 1968aa; McGuire, 1976aa). The basic elements of this framework are (i) identification and spatial distribution of seismic sources; (ii) characterization of each source in terms of its activity, recurrence rates for various earthquake magnitudes, and maximum magnitude; (iii) description of ground motion attenuation relationships to model the distribution of the ground motions expected when a given magnitude earthquake occurs on a particular source; and (iv) incorporation of the inputs into a logic tree to integrate the seismic source characterization and ground motion attenuation relationships, along with their associated uncertainties. Each logic tree pathway is intended to represent one expert's weighted interpretations of the seismic hazard at the site. The computation of the hazard for all possible pathways results in a distribution of hazard curves that DOE considers representative of the seismic hazard at a site, including variability and uncertainty.

To accomplish the PSHA, DOE hired two panels of experts. The first expert panel consisted of six three-member teams of geologists and geophysicists (seismic source teams) who developed probabilistic distributions to characterize relevant potential seismic sources in the Yucca Mountain region. These distributions included location and activity rates for fault sources, spatial distributions and activity rates for background sources, distributions of moment magnitude and maximum magnitude, and site-to-source distances. The second panel consisted of seven seismology experts (ground motion experts) who developed probabilistic point estimates of ground motion for a suite of earthquake magnitudes, distances, fault geometries, and faulting styles. These point estimates, expressed along with estimates of their uncertainties, were specific to the regional crustal conditions of the western Basin and Range Province. The ground motion attenuation point estimates were then fitted to yield the ground motion attenuation equations used in the PSHA.

Inputs from the expert teams were combined into a logic tree, and the hazard was computed using a modified version of the FRISK88 computer code (Risk Engineering, Inc., 1998aa). In the integration, DOE gave equal weight to all six source teams and seven ground motion experts. The resulting ground motion hazard curves express increasing levels of ground motion as a function of the annual probability that the ground motion will be exceeded. These curves include estimates of uncertainty.

The seismic source teams also developed a Probabilistic Fault Displacement Hazard Assessment as part of the PSHA. In that aspect of the PSHA elicitation, the experts derived probabilistic fault displacement hazard curves for nine demonstration points at or near Yucca Mountain. These demonstration points represent a range of faulting and related fault deformation conditions in the subsurface and near the sites of proposed surface facilities.

The NRC staff's technical evaluations of the geological, geophysical, and seismological information used to support the PSHA expert elicitation are provided in SER Sections 2.2.1.2.2.3.2 and 2.1.1.1.3.5.2.

Saturated Zone Flow and Transport Expert Elicitation (SZEE)

SAR Section 2.3.9.2.2.6 discussed DOE's use of expert elicitation to address key issues associated with groundwater flow and transport in the saturated zone. SAR Section 5.4.3 summarized the applicant's bases for asserting that SZEE was conducted in a manner generally consistent with the nine-step procedure suggested in NUREG–1563 (NRC, 1996aa).

In 1997, the applicant carried out an expert elicitation to evaluate saturated zone flow and transport at the Yucca Mountain site (CRWMS M&O, 1998ab). The objective of SZEE was to quantify uncertainties associated with models and parameters key to modeling flow and transport in the saturated zone. A second objective was to identify the necessary data collection and modeling that could reduce some of the more significant uncertainties. In this way, the expert elicitation was used to complement and guide data collection already underway, as well as to provide input to iterative performance assessment modeling by DOE.

Over a period of 6 months, a panel of 5 experts in saturated zone hydrology was asked to address 16 technical issues related to the study of saturated zone groundwater flow and radionuclide transport at Yucca Mountain. DOE implemented many of the panel members' recommendations in subsequent site characterization activities. In particular, the panel recommended a range of values for vertical anisotropy, dispersivity, and specific discharge that DOE later used, along with other sources of information, to characterize the uncertainty of flow and transport of radionuclides beneath and down gradient of Yucca Mountain. Written elicitation summaries, prepared by each expert, were included in an appendix to the final elicitation report (CRWMS M&O, 1998ab).

The NRC staff's technical evaluation of the geological, geophysical, and hydrological information used to support the Saturated Zone Flow and Transport expert elicitation, as well as the information developed as a result of it, is provided in SER Section 2.2.1.3.8.

NRC Staff's Review

The NRC staff reviewed DOE's techniques and process for conducting the three expert elicitations by using acceptance criterion 1 in Section 2.5.4.3 of the YMRP (NRC, 2003aa). This acceptance criterion considers the nine-step procedure outlined in NUREG–1563

(NRC, 1996aa), the extent to which DOE follows this procedure, and whether DOE adequately explains and justifies any variance from this guidance. In its evaluation of SAR Section 5.4, the NRC staff recognizes that rigid adherence to the nine-step procedure outlined in NUREG–1563, or strict compliance with other NRC guidance documents, is not a regulatory requirement. As identified explicitly in NUREG–1563, p. 9, "Methods and solutions differing from those set out… will be acceptable if they provide a sufficient basis for the findings requisite to the issuance of a permit or license by the Commission." These steps are discussed as follows, with specific examples cited from the three elicitations, where appropriate.

Definition of Objectives (Step 1)

The NRC staff concludes that, in general, the applicant defined specific objectives for each elicitation, consistent with the guidance in NUREG–1563. This conclusion is based on the NRC staff's review of the descriptions of these elicitation objectives in the SAR (pp. 5.4-4, 5.4-7, and 5.4-10), the discussion of the rationale for the elicitations in the respective elicitation reports, and on direct observation by the NRC staff members of the elicitation workshops and meetings.

In the PVHA elicitation, however, the NRC staff identified issues in the specific definition of objectives. The NRC staff documented in NUREG–1762, Section 5.1.2.2.4.1, Igneous Activity (NRC, 2005aa) that a common definition of an igneous event or event class was not adequately specified at the beginning of the elicitation, and that these terms were not used consistently in the experts' probability models. Probability estimates for intrusive and extrusive events were not calculated separately, but were initially considered as a single probability by the experts. Because separate probability estimates needed to be developed for the DOE Total System Performance Assessment (TSPA), DOE developed extrusive and intrusive probability estimates subsequent to the 1996 PVHA without re-engaging the experts to seek their opinions. The NRC staff's evaluation of DOE's revised extrusive and intrusive probability estimates for the PVHA is provided in SER Section 2.2.1.2.2.3.1. In this section, the NRC staff found that DOE's revised definitions and its corresponding probability estimates would not significantly affect the estimates of repository performance. Therefore, the NRC staff finds that, despite these issues, the applicant's definition of objectives for this elicitation is acceptable because it is consistent with the guidance in NUREG–1563.

Selection of Experts (Step 2)

The NRC staff concludes that the applicant generally followed published guidance in selecting experts. This conclusion is based on review of the criteria DOE used to select experts, as described in SAR Section 5.4; professional information provided about each expert in the elicitation reports; and the NRC staff direct observations of the open, frank, and detailed technical discussion among the experts at the elicitation workshops.

For the PVHA, the NRC staff finds that the 10 experts possessed the necessary knowledge and expertise and showed their ability to apply their knowledge and expertise for the following reasons. All 10 experts were identified in the 1996 PVHA report, and each expert's judgments were clearly documented. As was identified in its 1999 Issue Resolution Status Report (NRC, 1999ae), the NRC staff determined that a greater balance of panel experts would have encompassed a wider range of viewpoints. The NRC staff attributes DOE's inability to achieve this balance, in part, to the fact that some of the experts invited by DOE declined to participate as panel members. Also, subsequent to PVHA, the NRC staff suggested that DOE strive for more thorough documentation of the expert selection processes and identify sources of potential bias and conflicts of interest (Austin, 1997aa, 1996aa). Although the experts were not asked directly to disclose potential conflicts of interest, each expert provided sufficient information about his or her past and current affiliations, consistent with the disclosure criteria of NUREG–1563.

For the PSHA and the SZEE, the NRC staff finds that the experts possessed the necessary knowledge and expertise. The NRC staff also concludes that the assembled experts for the two elicitations collectively represent an appropriately broad spectrum of the larger seismology and hydrology communities.

The NRC staff finds that all of the final elicitation reports appropriately identified the participating subject matter experts, included summaries of their input to the elicitations, and provided rationales for their respective opinions. As the applicant stated in SAR Section 5.4, the experts were not asked directly to disclose potential conflicts of interest, but each expert provided sufficient information about his or her past and current affiliations, consistent with the guidance in NUREG–1563.

Refinement of Issues (Step 3) and Assembly and Dissemination of Basic Information (Step 4)

On the basis of its direct observation and review, the NRC staff finds that the geological, geophysical, hydrological, and seismological information the applicant made available to each panel provided an adequate technical basis to support the three elicitations, consistent with the guidance in NUREG–1563. During the early workshops and field trips, the experts developed lists of the most important sub-issues. This helped organize and focus the discussions in later elicitation workshops. Among the numerous sub-issues that the PVHA experts identified were structural control of igneous activity in the vicinity of Yucca Mountain, the quality and reliability of available age dating, and the selection of relevant natural analogues. The applicant divided the PSHA into two panels of experts, each with its own set of experts. One panel focused on description and characterization of seismic sources, while the other panel focused on ground motion attenuation and modeling. Within the seismic source panel, the applicant further developed three-person elicitation teams, composed of experts with varied expertise in geology, seismology, geophysics, and Basin and Range tectonics. Among the key sub-issues the experts identified in the SZEE were the causes and implications of the large hydraulic gradient, spatial distribution of flow, and the range of uncertainty in groundwater-specific discharge.

Pre-Elicitation Training (Step 5)

On the basis of its review of documentation provided in the elicitation reports, as well as direct observations by the NRC staff at the elicitation workshops, the NRC staff concludes that the subject matter experts received appropriate pre-elicitation training, consistent with NRC guidance in NUREG–1563. The subject matter experts received training on the elicitation process during the first workshop of each elicitation, as well as during subsequent workshops, including presentations on topics such as probability encoding, quantifying uncertainty, and identifying sources of bias.

Most of the workshops were held with sufficient advance notice so that members of the public, affected parties, and the NRC staff could directly observe the discussions among the experts and supporting technical teams. Many of the workshops included presentations by subject matter experts, both from within the teams or external to the elicitation. At later workshops, the experts presented their preliminary interpretations in a discussion format that allowed them to receive direct feedback from other experts or expert teams. Each of the elicitation projects included at least one field trip that allowed the experts to directly observe many of the important geologic features in the Yucca Mountain region. These field trips included discussions with subject matter experts and generalists on specific field investigations carried out on behalf of DOE in support of the site characterization. The applicant provided meeting summaries of all the workshops in the elicitation reports.

Elicitation of Judgments (Step 6)

The NRC staff finds that upon completion of the workshops and field trips, both the facilitation teams, comprised of generalists and normative experts, and the applicant, appropriately conducted comprehensive interviews of the experts to elicit their inputs that included discussion of how the information would be represented in the logic tree format used to calculate the results. These interviews were conducted expert by expert or, where applicable, team by team, and followed up with written documentation of the inputs. Therefore, the NRC staff finds that the applicant's approach on this step is consistent with the guidance in NUREG–1563.

Post-Elicitation Feedback (Step 7)

As documented in the elicitation reports and SAR Section 5.4, the experts were provided with both informal and formal feedback at many of the workshops. At least one workshop in each elicitation was dedicated to feedback and included initial sensitivity studies provided by the facilitation team to quantify the initial expert interpretations and, through sensitivity studies, to show which inputs had the greatest impact on the overall results. The NRC staff finds that this aspect of DOE's elicitation process is consistent with the criteria for timely feedback in NUREG-1563. However, the applicant did not require the experts or expert teams to document the rationale for any changes made to their assessments after the feedback session. As stated in SAR Section 5.4 and in DOE (2009gn), the applicant stated that this requirement could anchor the experts to their initial interpretations. The applicant asserted that the experts would thus be reluctant to revise their interpretations after receiving feedback because doing so would also require them to provide full justification for the change. The applicant also stated that its approach, in this regard, is consistent with the guidance contained in NUREG/CR–6372 (NRC, 1997aa). The NRC staff finds that, in this one respect, DOE's approach is more consistent with that described in NUREG/CR–6372 compared to NUREG–1563, in that the guidance in NUREG/CR–6372 does not specify that experts document the rationale for changes made to their assessment during or after feedback sessions. The NRC staff also finds that this approach is also consistent with the updated guidance on probabilistic seismic hazard assessments provided in NUREG–2117 (NRC, 2012aa), which supplements the guidance in NUREG/CR–6372. Specifically, Section 5.6 of NUREG–2117 recommends "…that expert evaluators do not document in detail their preliminary models and their technical bases in the project report" to prevent anchoring of expert's initial interpretations. The guidance further states that "[t]he reason for this is if the expert has invested considerable effort and time in documenting early-stage assessments, they may become reluctant to update these assessments, even if the results or discussions with other expert evaluators prompt them to do so." Accordingly, the NRC staff finds that DOE's approach is consistent with current guidance on expert assessment processes.

Treatment of Disparate Views and Aggregation of Judgments (Step 8)

The NRC staff finds that, for all three elicitations, the applicant appropriately used equal weighting methodology to aggregate the elicited results, demonstrated by the following examples. In the case of PSHA, results were aggregated giving equal weights to the inputs from the source teams, and equal weights to the ground motion models from the individual ground motion experts. In the other cases, equal weight was assigned to the results from each expert. The elicitation reports appropriately provided summaries of each expert's (or source teams) input, including sensitivity information to demonstrate the impact each expert or each source team's interpretations had on the final result. Therefore, the NRC staff finds that the applicant's approach on this step is consistent with the guidance in NUREG–1563.

Documentation (Step 9)

The NRC staff concludes that DOE properly documented all three elicitations, demonstrated by the following examples. The elicitation reports provided comprehensive records of each elicitation, with the noted exception being formal documentation of individual experts' reasons for revising their interpretations during the elicitation process. The applicant explained its rationale for this deviation in SAR Section 5.4. As stated previously, the NRC staff finds that DOE selected an approach that in this regard is similar to that of NUREG/CR–6372 (NRC, 1997aa) and NUREG–2117 (NRC, 2012aa).

The second acceptance criterion in the YMRP guidance for review of expert elicitation also considers the documentation and methodology used in updates to an elicitation. The applicant reconvened the PVHA elicitation in 2004 to consider new information and to rely on a consistent set of event definitions and extrusive scenarios. The NRC staff attended the public PVHA-U workshops as observers. DOE published the results from the updated PVHA, or PVHA-U, after it submitted the SAR (SNL, 2008ah). DOE did not directly use the PVHA-U results in its SAR or in direct support of models or parameters in the TSPA. Estimates of the probability of igneous activity in the TSPA are based solely on the original 1996 PVHA. In a letter providing the PVHA-U report to the NRC (Boyle, 2008aa), DOE characterized the PVHA-U results as information that supports the results of the 1996 PVHA. While the NRC staff concludes that the results are similar, the underlying technical bases for the two elicitations are not the same. Because DOE referred to the PVHA-U results as confirming the 1996 PVHA results, the NRC staff reviewed the PVHA-U report. On the basis of the NRC staff's subsequent review and direct observation of the PVHA-U workshops, the NRC staff finds that the PVHA-U was conducted in a manner consistent with the procedure suggested in NUREG–1563 (NRC, 1996aa). For these reasons, the NRC staff finds that the PVHA-U was adequately documented and used appropriate elicitation methods, consistent with the guidance in NUREG-1563.

The applicant did not update the PSHA or the SZEE elicitations. The NRC staff finds that the decision not to update these two elicitations is acceptable, as there was no new or updated information developed after the elicitations were completed that required additional consideration by the respective expert panels.

The NRC staff's technical evaluation of the applicant's estimates of igneous event probability, as they relate to the PVHA-U, is given in SER Section 2.2.1.2.2.

2.5.4.4 Evaluation Findings

The NRC staff reviewed the SAR and other information submitted in support of the license application against the requirements in 10 CFR 63.21(c)(19) and the applicable guidance sections of the YMRP. The NRC staff finds that DOE's SAR is consistent with guidance in NUREG–1563, NUREG/CR–6372, and NUREG–2117 and adequately explained how expert elicitation was used to support the license application. Therefore, on the basis of this review, the NRC staff concludes, with reasonable assurance, that the requirements in 10 CFR 63.21(c)(19) are satisfied.

2.5.4.5 References

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CHAPTER 21

Conclusions

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed and evaluated the U.S. Department of Energy's (DOE or the applicant) Safety Analysis Report (SAR), Chapter 2: Repository Safety After Permanent Closure and the other information submitted in support of its license application and has found that DOE submitted applicable information required by 10 CFR 63.21. The NRC staff has also found, with reasonable expectation, that (i) the proposed Yucca Mountain repository design meets the applicable performance objectives in Subpart E, including the requirement that the repository be composed of multiple barriers and (ii) based on performance assessment evaluations that are in compliance with applicable regulatory requirements, meets the 10 CFR Part 63, Subpart L limits for individual protection, human intrusion, and separate standards for protection of groundwater.

CHAPTER 22

Glossary

This glossary is provided for information and is not exhaustive. The glossary provides explanations for the terms shown in italics.

absorption: The process of taking up by capillary, osmotic, solvent, or chemical action of molecules (e.g., *absorption* of gas by water) as distinguished from *adsorption*.

abstracted model: A *model* that reproduces, or *bounds*, the essential elements of a more detailed *process model* and captures *uncertainty* and v*ariability* in what is often, but not always, a simplified or idealized form. See *abstraction*.

abstraction: Representation of the essential components of a *process model* into a form suitable for use in a *total system performance assessment*. A *model abstraction* is intended to maximize the use of limited computational resources while allowing a sufficient range of sensitivity and *uncertainty* analyses.

adsorb: To collect a gas, liquid, or dissolved substance on a surface as a condensed layer.

adsorption: The adhesion by chemical or physical forces of molecules or ions (as of gases or liquids) to the surface of solid bodies. For example, the transfer of *solute* mass, such as *radionuclides*, in groundwater to the solid geologic surfaces with which it comes in contact. The term *sorption* is sometimes used interchangeably with this term.

advection: The process in which *solutes*, particles, or molecules are transported by the motion of flowing fluid.

aging: The retention of *commercial spent nuclear fuel* on the surface in *dry storage* to reduce its thermal output as necessary to meet proposed repository thermal management goals.

airborne mass loading: The amount of fine particulates resuspending above a surface deposit, generally expressed as mass per unit volume of air.

aleatory uncertainty: An *uncertainty* associated with the chance of occurrence of a feature, event, or process of a physical system or the environment such as the timing of a volcanic event. Also referred to as irreducible *uncertainty* because no amount of knowledge will determine whether or not a chance event will or will not occur. See also *epistemic uncertainty*.

Alloy 22: A nickel-based, *corrosion*-resistant alloy containing approximately 22 weight percent chromium, 13 weight percent molybdenum, and 3 weight percent tungsten as major alloying elements. This alloy is used as the outer container material in U.S. Department of Energy's waste package design.

Alluvial, alluvial fan: Pertaining to the process of moving sediment by running water (see *alluvium*). An *alluvial fan* is a wedge-shaped (fan-shaped in plane view) sedimentary deposit of *alluvium* formed at the base of a slope in arid regions.

alluvium: Detrital (sedimentary) deposits made by flowing surface water on river beds, flood plains, and *alluvial fans*. It does not include subaqueous sediments of seas and lakes.

alternative: In the context of system analysis, plausible interpretations or designs that use assumptions other than those used in the base case, which could also be applicable or reasonable given the available scientific information. When propagated through a quantitative tool such as performance assessment, *alternative* interpretations can illustrate the significance of the *uncertainty* in the base case interpretation chosen to represent the system's probable behavior.

ambient: Undisturbed, natural conditions, such as *ambient* temperature caused by *climate* or natural subsurface thermal gradients, and other surrounding conditions.

anisotropy: Variation in physical properties when measured in different directions. For example, in layered rock, *permeability* is often greater within the horizontal layers than across the horizontal layers.

annual frequency: The number of occurrences of an event in 1 year.

aqueous: Pertaining to water, such as *aqueous phase*, *aqueous* species, or *aqueous transport*.

ash: Fragments of volcanic rock that are broken from *magma* and/or country rock during an explosive volcanic eruption to less than 2 mm [0.08 in] in diameter. See also *tephra* and *pyroclastic.*

ash flow tuff: A type of volcanic rock formed by the deposition and accumulation of dominantly ash-size particles during an explosive eruption. *Ash flows* (also called *pyroclastic* flows) commonly result from eruptions of more viscous, silica-rich *magma* such as *rhyolite*. *Ash flow tuff* forms the *host horizons* for the proposed geologic repository at Yucca Mountain. See also *tuff* and *welded tuff*.

basalt: A common type of *igneous* rock (and/or low-viscosity *magma*) that forms black, rubbly-to-smooth-surfaced lavas and black-to-red *tephra* deposits.

borosilicate glass: A predominantly noncrystalline, relatively homogenous glass formed by melting silica and boric oxide together with other constituents such as alkali oxides. *Borosilicate glass* is a high-level radioactive waste material in which boron takes the place of the lime used in ordinary glass mixtures.

boundary condition: For a *model*, the establishment of a set condition for a given *variable,* often at the geometric edge of the *model*. An example is using a specified groundwater flux for *net infiltration* as a *boundary condition* for an *unsaturated zone flow model*.

bound: An analysis or selection of *parameter* values that yields limiting results, such that any actual result is certain to exceed these limits only with an extremely small likelihood.

breach: A penetration in the waste package caused by failure of the outer and inner containers or barriers that allows the *spent nuclear fuel* or the high-level radioactive waste to be exposed to the external environment and may eventually permit *radionuclide* release.

burnup: A measure of nuclear reactor fuel consumption expressed either as the percentage of fuel atoms that have undergone fission, or as the amount of energy produced per unit weight of fuel.

burnup credit: The concept of taking credit for the reduction in reactivity (ability to undergo fission) due to fuel irradiation. The reduction in reactivity is due to the net reduction of fissile nuclides and the production of parasitic neutron-absorbing nuclides.

caldera*:* A volcanic depression in the Earth's surface more than 1 km [0.7 mi] wide, formed by the collapse of the upper crust into an evacuated *magma* chamber during or after a large volcanic eruption. Many *calderas* resulting from the explosive eruption of large amounts of *rhyolite magma* are several tens of kilometers [up to 20 mi] wide.

calibration: (1) Comparison of *model* results with actual data or observations, and adjusting *model parameters* to increase the precision and/or accuracy of *model* results compared to actual data or observations. (2) For tools used for field or lab measurements, the process of taking instrument readings on standards known to produce a certain response, to check the accuracy and precision of the instrument.

canister: An unshielded cylindrical metal receptacle that facilitates handling, transportation, storage, and/or disposal of high-level radioactive waste. It may serve as (i) a pour mold and container for vitrified high-level radioactive waste; (ii) a container for loose or damaged fuel rods, nonfuel components and assemblies, and other debris containing *radionuclides*; or (iii) a container that provides *radionuclide* confinement. *Canisters* are used in combination with specialized overpacks that provide structural support, shielding, or confinement for storage, transportation, and emplacement. Overpacks used for transportation are usually referred to as transportation *casks*; those used for emplacement in a proposed repository are referred to as waste container.

carbon steel: A steel made with carbon up to about 2 weight percent and only residual quantities of other elements. *Carbon steel* is a tough but ductile and malleable material that is used in some components in U.S. Department of Energy's design of the engineered barrier system.

cask*:* (1) A heavily shielded container used for the *dry storage* or shipment (or both) of radioactive materials such as *spent nuclear fuel* or other high-level radioactive waste. *Casks* are often made from lead, concrete, or steel. *Casks* must meet regulatory requirements and are not intended for long-term disposal in a proposed repository. (2) A heavily shielded container that the U.S. Department of Energy would use to transfer *canisters* between waste handling facilities at the proposed repository.

cinder cone*:* A steep, conical hill formed by the accumulation of *ash* and coarser erupted material *(tephra*) around a volcanic *vent*. Synonymous with *scoria cone*.

cladding: The metal outer sheath of a fuel rod generally made of a zirconium alloy or *stainless steel*, intended to protect the uranium dioxide pellets, which are the nuclear fuel, from *dissolution* by exposure to high-temperature water under operating conditions in a reactor.

climate: Weather conditions, including temperature, wind velocity, precipitation, and other factors, that prevail in a region.

climate states: Representations of *climate* conditions.

colloid: As applied to *radionuclide migration*, *colloids* are large molecules or very small particles, having at least one dimension with the size range of 10^{-6} to 10^{-3} mm [10^{-8} to 10^{-5} in] that are suspended in a solvent. *Colloids* in groundwater arise from clay minerals, organic materials, or (in the context of a proposed geologic repository) from *corrosion* of engineered materials.

commercial spent nuclear fuel: Nuclear fuel rods, forming a fuel assembly, that have been removed from a nuclear power plant after reaching the specified *burnup*.

conceptual model: A set of qualitative assumptions used to describe a system or subsystem for a given purpose. Assumptions for the *model* are compatible with one another and fit the existing data within the context of the given purpose of the *model*.

conduit*:* A *pathway* along which *magma* rises to the surface during a volcanic eruption. *Conduits* are usually cylindrical and flare upwards toward the surface *vent*. *Conduits* are near-surface *features* and develop along *dikes*, focusing *magma flow* from the longer and possibly narrower *dike* to the vent.

consequence: A measurable or calculated outcome of an event or process that, when combined with the *probability* of occurrence, gives a measurement of *risk*.

conservative: A condition of an analysis or a *parameter* value such that its use provides a pessimistic result, which is worse than the actual result expected.

corrosion: The deterioration of a material, usually a metal, as a result of a chemical or electrochemical reaction with its environment. *Corrosion* includes, but is not limited to, general *corrosion*, microbially influenced *corrosion*, localized *corrosion*, *galvanic corrosion*, and stress corrosion cracking.

coupled processes: A representation of the interrelationships between *processes* such that the effects of variation in one process are accurately propagated among all interrelated *processes*.

criticality: The condition in which a fissile material sustains a chain reaction. It occurs when the number of neutrons present in one generation cycle equals the number generated in the previous cycle. The state is considered critical when a self-sustaining nuclear chain reaction is ongoing.

diffusion: (1) The spreading or dissemination of a substance caused by concentration gradients. (2) The gradual mixing of the molecules of two or more substances because of random thermal motion.

diffusive transport: *Diffusive transport* is the process in which substances carried in groundwater move through the subsurface by means of *diffusion* because of a concentration gradient.

dike: A tabular, generally vertical body of *igneous* rock that cuts across the *structure* of adjacent rocks. *Dikes transport* molten rock (*magma*) from depth to an erupting volcano, but not all *dikes* feed an eruption.

dimensionality: Modeling in one, two, or three dimensions.

direct exposure: The manner in which an individual receives dose from being in close proximity to a source of radiation. *Direct exposures* present an external dose *pathway*.

dispersion (hydrodynamic dispersion): (1) The tendency of a *solute* to spread out from the path it is expected to follow if only the bulk motion of the flowing fluid were to move it. The tortuous path the *solute* follows through openings (pores and *fractures*) causes part of the dispersion effect in the rock. (2) The macroscopic outcome of the actual movement of individual *solute* particles through a porous medium. Dispersion dilutes *solutes*, including *radionuclides*, in groundwater.

disruptive event: An unlikely, off-normal event that, in the case of the proposed repository at Yucca Mountain, could include *volcanic* activity, *seismic* activity, and nuclear *criticality*. *Disruptive events* alter the normal or likely behavior of the system.

dissolution: Dissolving a substance in a solvent.

distribution: In a *total system performance assessment*, the overall scatter of values for a specific set of numbers (e.g., *corrosion* rates, values used for a particular *parameter*, dose results). A term used synonymously with *frequency distribution* or *probability distribution* function. *Distributions* have *structures* that are the *probability* that a given value occurs in the set.

drift: From mining terminology, a horizontal or sub-horizontal underground passage. In the proposed Yucca Mountain repository design, *drifts* include excavations for emplacement of waste canisters (*emplacement drifts*) and access (access mains).

drift degradation: The progressive accumulation of rock rubble in a *drift* created by weakening and collapse of *drift* walls in response to stress from heating or earthquakes.

drip shield: A metallic *structure* placed along the extension of the *emplacement drifts* and above the *waste packages* to prevent *seepage* water from directly dripping onto the *waste package* outer surface. The *drip shield* may also prevent the *drift* ceiling rocks (e.g., due to *drift* spallation) from falling on the *waste package*.

dry storage: Storage of *spent nuclear fuel* without immersion of the fuel in water for cooling or shielding; it involves the encapsulation of spent fuel in a steel cylinder that might be in a concrete or massive steel *cask* or *structure*.

effective porosity: The fraction of a porous medium volume available for fluid flow and/or *solute* storage, as in the saturated zone. *Effective porosity* is less than or equal to the total void space (*porosity*).

empirical: Reliance on observation or experimentation rather than on a theoretical understanding of fundamental *processes*.

emplacement drift: See *drift*.

enrichment: The act of increasing the concentration of fissile isotopes from their value in natural uranium. The *enrichment* (typically reported in atom percent) is a characteristic of nuclear fuel.

eolian: Relating to *processes* caused by near-surface winds.

epistemic uncertainty: A *variability* that is due to a lack of knowledge of quantities or *processes* of the system or the environment. Also referred to as reducible *uncertainty*, because the state of knowledge about the exact value of a quantity or process can increase through testing and data collection. See also *aleatory uncertainty.*

Equilibrium (chemical): The state of a chemical system in which the *phases* do not undergo any spontaneous change in properties or proportions with time; a dynamic balance.

events: In a *total system performance assessment*, (1) occurrences of phenomena that have a specific starting time and, usually, a duration shorter than the time being simulated in a *model* or (2) uncertain occurrences of phenomena that take place within a short time relative to the time frame of the *model*.

event tree*:* A modeling tool that illustrates the logical sequence of *events* that follow an initiating event.

expected annual dose: The average annual radiological dose calculated for the *reasonably maximally exposed individual*, which includes the likelihood of the individual receiving a dose from all relevant exposure *scenarios*.

expert elicitation: A formal, highly *structured*, and well-documented process whereby expert judgments, usually of multiple experts, are obtained.

Exploratory Studies Facility: An underground laboratory at Yucca Mountain that includes a 7.9-km [4.9-mi] main loop (tunnel); a 2.8-km [1.75-mi] cross *drift*; and a research alcove system constructed for performing underground studies during site characterization.

extrusive (extrusion)*:* In relation to *igneous* activity, an event where *magma* erupts at the surface. An extrusion is the deposit formed by an extrusive event. See also *intrusive*.

fault (geologic): A planar or gently curved *fracture* across which there has been displacement of rocks or sediment parallel to the *fracture* surface.

features: Physical, chemical, thermal, or temporal characteristics of the site or proposed repository system at Yucca Mountain. For the purposes of screening *features*, *events*, and *processes* for the *total system performance assessment*, a feature is defined to be an object, *structure*, or condition that has a potential to affect disposal system performance.

fissure: In relation to *igneous* activity, a *fissure* is an elongated vent or line of *vents*, formed when a *dike* breaks to the surface to start a *volcanic eruption*.

flow: The movement of a fluid such as air, water, or *magma*. *Flow* and *transport* are *processes* that can move *radionuclides* from the proposed repository to the *receptor* group location.

flow pathway: The subsurface course that water or a *solute* (and dissolved material) would follow in a given groundwater velocity field, governed principally by the hydraulic gradient.

fluvial: Processes related to the downslope movement of water in streams and rivers on the Earth's surface.

fracture: A planar discontinuity in rock along which loss of cohesion has occurred. It is often caused by the same stresses that cause folding and faulting. A *fracture* along which there has been displacement of the sides relative to one another is called a *fault*. A *fracture* along which no appreciable movement has occurred is called a joint. *Fracture*s may act as paths for fast groundwater movement.

fragility: *Fragility* of a *structure, system, or component* is defined as the conditional *probability* of its failure, given a value of the ground motion, or response *parameter*, such as stress, bending moment, and spectral acceleration.

frequency: The number of occurrences of an observed or predicted event during a specific time period.

galvanic: Pertains to an electrochemical process in which two dissimilar electronic conductors are in contact with each other and with an electrolyte, or in which two similar electronic conductors are in contact with each other and with dissimilar electrolytes.

geochemical: The *distribution* and amounts of the chemical elements in minerals, ores, rocks, soils, water, and the atmosphere; the movement of the elements in nature on the basis of their properties.

geophysics (geophysical survey; geophysical magnetic survey): Study of the physical properties of rocks and sediment and interpretation of data derived from measurements made. Properties commonly measured are the velocity of sound (*seismic* waves) in rocks, density, and magnetic character. A program of measurements made on a series of rocks is usually termed a survey.

half-life: The time required for a radioactive substance to lose half of its activity due to *radioactive decay*. At the end of one *half-life*, 50 percent of the original radioactive material has decayed.

heterogeneity: The condition of being composed of parts or elements of different kinds. A condition in which the value of a *parameter* varies over the space an entity occupies, such as the area around the proposed repository, or with the passage of time.

host horizon, host rock: The rocks in which the proposed Yucca Mountain geologic repository are intended to be mined.

hydrologic: Pertaining to the properties, *distribution*, and circulation of water on the surface of the land, in the soil and underlying rocks, and in the atmosphere.

igneous: (1) An activity or process related to the formation and movement of *magma*, either in the subsurface (*intrusive*) or on the surface (*extrusive*, or *volcanic*). (2) A type of rock that has formed from a molten, or partially molten, material, or *magma*.

infiltration: The process of water entering the soil at the ground surface. *Infiltration* becomes percolation when water has moved below the depth at which evaporation or *transpiration* can return it to the atmosphere. See also *net infiltration*.

***intrusive (intrusion)**:* In relation to *igneous* activity, an event where *magma* approaches the surface but does not break through in an eruption (or the unerupted *magma* during an *igneous* event). An intrusion is the solidified rock formed below the surface by an *intrusive* event. See also *extrusive*.

invert: A constructed surface that would provide a level *drift* floor and enable emplacement and support of the *waste packages*.

license application: An application from the U.S. Department of Energy to the U.S. Nuclear Regulatory Commission for a license to construct and operate the proposed repository at Yucca Mountain.

lithophysal: Containing lithophysae, which are holes in *tuff* and other volcanic rocks. One way lithophysae are created is by the accumulation of volcanic gases during the formation of the *tuff*.

magma: Molten or partially molten rock that is naturally occurring and is generated within the Earth. *Magma* may contain crystals along with dissolved gasses.

mathematical model: A mathematical description of a *conceptual model*.

matrix (geology): In general terms, rock material and its pore space. For Yucca Mountain, the rock is conceptually divided into matrix and *fractures*; the matrix is the portion of rock between *fractures*. The pore space in the matrix can be referred to as the primary *porosity*, as opposed to the pore space in *fractures* that can be referred to as secondary *porosity*.

matrix diffusion: The process by which molecular or ionic *solutes*, such as *radionuclides* in groundwater, move from areas of higher concentration to areas of lower concentration. For the proposed Yucca Mountain repository, this process refers to the movement of *radionuclides* by *diffusion* between the *fracture* and *matrix* continua.

matrix permeability: The capability of the *matrix* to transmit fluid.

mean (statistical): For a statistical data set, the sum of the values divided by the number of items in the set. The arithmetic average, sometimes referred to as expected value.

mechanical disruption: Damage to the *drip shield* or *waste package* because of external forces.

median (statistical): A value such that one-half of the observations are less than the value and one-half are greater than the value.

meteorology: The study of climatic conditions such as precipitation, wind, temperature, and relative humidity.

microbe: An organism too small to be viewed with the unaided eye. Examples of *microbes* are bacteria, protozoa, and some fungi and algae.

migration: *Radionuclide* movement from one location to another within the engineered barrier system or the environment.

mineralogical: Of or relating to the chemical and physical properties of minerals, their occurrence, and their classification.

mode (statistical): A statistic for a set of data values that represents the value that occurs most frequently in that set.

model: A depiction of a system, phenomenon, or process, including any hypotheses required to describe the system or explain the phenomenon or process.

model support: A process used to gain confidence in the reasonableness of *model* results through comparison with outputs from detailed process-level *models* and/or *empirical* observations such as laboratory tests, field investigations, and *natural analogues*.

natural analogues: Naturally-occurring, observable *features, events,* or *processes*, equivalent to those that might affect the repository in the future. These provide insights on similar *features, events, or processes* that are required to be examined for the proposed Yucca Mountain repository system. An example might be a *dike* in an existing volcanic system, or a fault that affects similar rocks to those at the repository, both occurring near the repository site or directly relatable to it.

near-field: The area and conditions within the proposed repository including the *drifts* and *waste packages* and the rock immediately surrounding the *drifts*. The *near-field* is the region in and around the proposed repository where the excavation of the proposed repository *drifts* and the emplacement of waste have significantly impacted the natural *hydrologic* system.

net infiltration: The downward flux of infiltrating water that escapes below the zone of evapotranspiration. The bottom of the zone of evapotranspiration generally coincides with the lowermost extent of plant roots.

nominal scenario class: The *scenario*, or set of related *scenarios*, that describes the expected or nominal behavior of the natural system as perturbed only by the presence of the proposed repository at Yucca Mountain. The nominal *scenarios* contain all likely *features, events*, and *processes* that have been retained for analysis.

numerical model: An approximate representation of a *mathematical model* that is constructed using a numerical description method such as finite volumes, finite differences, or finite elements. A *numerical model* is typically represented by a series of program statements that are executed on a computer.

occupational dose*:* The dose received by an individual in the course of employment in which the individual's assigned duties involve exposure to radiation or to radioactive material from licensed and unlicensed sources of radiation, whether in the possession of the licensee or other person. *Occupational dose* does not include doses received from background radiation, from any medical administration the individual has received, from exposure to individuals who were administered radioactive material and released under 10 CFR 35.75, from voluntary participation in medical research programs, or as a member of the public (10 CFR 20.1003, "*Occupational dose*").

oxidation: A *corrosion* reaction in which the corroded metal forms an oxide, usually applied to reaction with a gas containing elemental oxygen, such as air.

parameter: Data, or values, such as those that are input to computer codes for a *total system performance assessment* calculation.

patch: In the U.S. Department of Energy modeling of *waste package corrosion*, a *patch* is the minimal surface area of the *waste package* over which general corrosion occurs, as opposed to localized corrosion in *pits*.

pathway: A potential route by which *radionuclides* might reach the accessible environment and pose a threat to humans. For example, *direct exposure* is a human external *pathway*, and inhalation and ingestion are human internal *pathway*s.

permeability: A measure of the ease with which a fluid such as water or air moves through a rock, soil, or sediment.

phase: A physically homogeneous and distinct portion of a material system, such as the gaseous, liquid, and solid *phases* of a substance. In liquids and solids, single *phases* may coexist.

phase stability: A measure of the ability of a particular *phase* to remain without transformation.

pit (in material science): A small cavity formed in a solid as a result of localized corrosion.

porosity: The ratio of the volume occupied by openings, or voids, in a soil or rock, to the total volume of the soil or rock. Porosity is expressed as a decimal fraction or as a percentage.

probabilistic: Based on or subject to *probability*.

probability: The chance that an outcome will occur from the full set of possible outcomes. Knowledge of the exact *probability* of an event is usually limited by the inability to know, or compile, the complete set of possible outcomes over time or space.

probability distribution: The set of outcomes (values) and their corresponding probabilities for a random *variable*. See *distribution*.

processes: Phenomena and activities that have gradual, continuous interactions with the system being modeled.

process model: A depiction or representation of a *process*, along with any hypotheses required to describe or to explain the *process*.

pyroclastic: In relation to *igneous volcanic* activity, this describes fragments or fragmental rocks and deposits produced by explosive eruptions, where the *magma* is ripped apart during the release of gas and/or by interaction with surface and near-surface water.

Quaternary*:* The period of geologic time from about 2 million years ago to the present day.

radioactive decay: The process in which one *radionuclide* spontaneously transforms into one or more different *radionuclides*, which are called daughter *radionuclides*.

radioactivity: The property possessed by some elements (such as uranium) of spontaneously emitting energy in the form of radiation as a result of the decay (or disintegration) of an unstable atom. *Radioactivity* is also the term used to describe the rate at which radioactive material emits radiation.

radiolysis: Chemical decomposition by the action of radiation.

radionuclide: An unstable isotope of an element that decays or disintegrates spontaneously, thereby emitting radiation. Approximately 5,000 natural and artificial radioisotopes have been identified.

range (statistical): The numerical difference between the highest and lowest value in any set.

reasonably maximally exposed individual: A hypothetical person meeting the criteria of 10 CFR 63.312.

receptor: An individual for whom radiological doses are calculated or measured.

redistribution: Mobilization and *transport* of surface deposits by wind and water.

***reliability**:* The *probability* that the item will perform its intended function(s) under specified operating conditions for a specified period of time.

repository footprint: The outline of the outermost locations of where the waste is proposed to be emplaced in the proposed Yucca Mountain repository.

retardation: Slowing or stopping *radionuclide* movement in groundwater by mechanisms that include *sorption* of *radionuclides*, *diffusion* into *rock matrix* pores and microfractures, and trapping of particles in small pore spaces or dead ends of microfractures.

***rhyolite**:* A common type of *igneous* rock that forms light-colored, rough blocky surfaced lavas and white-grayish-yellow *tephra* deposits. A common fragment type is pumice. Rhyolitic *magma* has a high viscosity, and the resulting lava flows are usually quite short and thick. It more frequently erupts explosively from the volcano and forms *ash-flow tuffs*.

risk: The *probability* that an undesirable event will occur, multiplied by the *consequences* of the undesirable event.

risk assessment: An evaluation of potential *consequences* or hazards that might be the outcome of an action, including the likelihood that the action might occur. This assessment focuses on potential negative impacts on human health or the environment.

risk-informed, performance-based: A regulatory approach in which *risk* insights, engineering analysis and judgments, and performance history are used to (i) focus attention on the most important activities; (ii) establish objective criteria on the basis of *risk* insights for evaluating performance; (iii) develop measurable or calculable *parameters* for monitoring system and licensee performance; and (iv) focus on the results as the primary basis for regulatory decision making.

rockfall: In terms of the proposed Yucca Mountain repository, the release of *fracture*-bounded blocks of rock from the *drift* wall, usually in response to an earthquake.

rock matrix: See *matrix*.

runoff: Lateral movement of water at the ground surface, such as down steep hillslopes or along channels, that is not able to infiltrate at a specified location.

scenario: A well-defined, connected sequence of *features, events*, and *processes* that can be thought of as an outline of a possible future condition of the proposed repository system. Scenarios can be undisturbed, in which case the performance would be the expected, or nominal, behavior for the system. Scenarios can also be disturbed, if altered by *disruptive events* such as human intrusion or natural phenomena such as *volcanism* or nuclear *criticality*.

scenario class: A set of related *scenarios* sharing sufficient similarities that they can usefully be aggregated for screening or analysis. The number and breadth of *scenario classes* depend on the resolution at which scenarios have been defined.

scoria; scoria cone: Scoria is the *basalt*ic equivalent of pumice, a frothy material due to gas-expansion in the *magma*. For scoria cone, see *cinder cone*.

seepage: Water dripping into a *drift.* This usage is specific to Yucca Mountain.

seismic: Pertaining to, characteristic of, or produced by earthquakes or Earth vibrations.

seismic hazard curve: A graph showing the ground motion *parameter* of interest, such as peak ground acceleration, peak ground velocity, or spectral acceleration at a given *frequency*, plotted as a function of its annual *probability* of exceedance.

seismic performance: *Seismic performance* of *structures, systems, and components* refers to their ability to perform intended safety functions during a *seismic* event, expressed as the annual *probability* of exceeding a specified limit condition (stress, displacement, or collapse). This is also referred to as the *probability* of failure, or *probability* of unacceptable performance, P_F.

sill: A tabular, generally flat-lying body of *intrusive igneous* rock that lies along (is concordant with) the *structure* of adjacent rocks. *Sills* are part of the *transport* system for molten rock (*magma*) rising from depth to the surface. See also *dike*.

sorb: To undergo a process of *sorption*.

solute: A substance that is dissolved in a solution (e.g., radioactive waste dissolved in groundwater)

sorption: The binding, on a microscopic scale, of one substance to another. *Sorption* is a term that includes both *adsorption* and *absorption* and refers to the binding of dissolved *radionuclides* onto geologic solids or *waste package* materials by means of close-range chemical or physical forces. *Sorption* is a function of the chemistry of the radioisotopes, the fluid in which they are carried, and the material they encounter along the *flow* path.

sorption coefficient (K_d): A numerical means to represent how strongly one substance *sorbs* to another.

source term: Types and amounts of *radionuclides* that are the source of a potential release.

spatial variability: A measure of how a property, such as rock *permeability*, varies at different locations in an object such as a rock formation.

speciation: The existence of the elements, such as *radionuclides*, in different molecular forms in the *aqueous phase*.

spent nuclear fuel: Fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated by reprocessing.

stainless steel*:* A class of iron-base alloys containing a minimum of approximately 10 percent chromium to provide *corrosion* resistance in a wide variety of environments.

stratigraphy: The branch of geology that deals with the definition and interpretation of rock strata; the conditions of their formation, character, arrangement, sequence, age, and *distribution*; and especially their correlation by the use of fossils and other means of identification. See *stratum*.

stratum (*plural *strata): A layer of rock or soil with geologic characteristics that differ from the layers above or below it.

structure: In geology, the geometric arrangement of rocks, or geologic *features* (or areas of interest) such as folds and *faults*. Includes *features* such as *fractures* created by faulting, and joints caused by various *processes*, including those associated with the heating of rock. For engineering usage, see *structures, systems, and components*.

structures, systems, and components: A *structure* is an element, or a collection of elements, that provides support or enclosure, such as a building, *aging* pad, or *drip shield*. A *system* is a collection of components, such as piping; cable trays; *conduits*; or heating, ventilation, and air conditioning equipment that are assembled to perform a function. A *component* is an item of mechanical or electrical equipment, such as a *canister* transfer machine, transport and emplacement vehicle, pump, valve, or relay.

tectonic: Pertaining to geologic *features* or *events* created by deformation of the Earth's crust.

tephra: A collective term for all clastic (fragmental) materials ejected from a volcano during an eruption and *transported* through the air.

thermal chemical: Of or pertaining to the effect of heat on chemical conditions and reactions.

thermohydrologic: Of or pertaining to changes in groundwater movement due to the effects of changes in temperature.

thermal mechanical: Of or pertaining to changes in mechanical properties from effects of changes in temperature.

total system performance assessment: A *risk assessment* that quantitatively estimates how the proposed Yucca Mountain repository system will perform in the future under the influence of specific *features, events*, and *processes*, incorporating *uncertainty* in the *models* and *uncertainty* and *variability* of the data.

transparency: The ease of understanding the process by which a study was carried out, which assumptions are driving the results, how they were arrived at, and the rigor of the analyses leading to the results. A logical structure ensures completeness and facilitates in-depth review of the relevant issues. *Transparency* is achieved when a reader or reviewer has a clear picture of what was done in the analysis, why it was done, and the outcome.

transpiration: The removal of water from the ground by vegetation (roots).

transport: A process that allows substances such as contaminants, *radionuclides*, or *colloids*, to be carried in a fluid from one location to another. Transport *processes* include the physical mechanisms of *advection*, convection, *diffusion*, and *dispersion* and are influenced by the chemical mechanisms of *sorption*, leaching, precipitation, *dissolution*, and complexation.

tuff: A general term for volcanic rocks that formed from fragmented *magma* and fragments of other rocks, and that erupted from a volcanic vent, flowed away from the vent as a suspension of solids and hot gases, or fell from the eruption cloud, and consolidated at the location of deposition. *Tuff* is the most abundant type of rock at the proposed Yucca Mountain repository site. *Welded tuff* is one type.

uncertainty: How much a calculated or measured value varies from the unknown true value. See also *aleatory uncertainty* and *epistemic uncertainty.*

unsaturated zone: The zone between the land surface and the regional water table. Generally, fluid pressure in this zone is less than atmospheric pressure, and some of the voids may contain air or other gases at atmospheric pressure. Beneath flooded areas or in perched water bodies, the fluid pressure locally may be greater than atmospheric.

unsaturated zone flow: The movement of water in the *unsaturated zone*, as driven by capillary, viscous, gravitational, inertial, and evaporative forces.

vadose zone: Synonymous with *unsaturated zone*.

variable: A nonunique property or attribute used to represent the *parameters* or unknowns in an equation or formula.

variably saturated zone: Synonymous with *unsaturated zone*.

variability (statistical): A measure of how a quantity varies over time or space.

Vent (geology): The point on the Earth's surface at which *magma* extrudes to form a volcanic eruption. May include geologic deposits or structures associated with the *vent*.

volcanic, volcanic activity, volcanism: Pertaining to extrusive *igneous* activity.

wash: In relation to landforms (geomorphology), a streambed, dry or running, usually in a semi-arid or arid environment.

waste package: The waste form and any containers, shielding, packing, and other absorbent materials immediately surrounding an individual waste container.

watershed: Used to indicate an area of land from which all water falling as precipitation would *flow* toward a single point. *Watershed* is also sometimes used for drainage area (i.e., the area drained by a single stream-river system including the adjacent ridges and hillslopes). The upstream boundaries of *watersheds* are the high points (ridges, etc.) that separate two drainage areas.

welded tuff*:* A *tuff* deposited under conditions where the particles that make up the rock remain sufficiently hot to weld or sinter together. In contrast to nonwelded *tuff*, *welded tuff* is denser, less porous, and more likely to be *fractured* (which increases *permeability*).

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10. SUPPLEMENTARY NOTES

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11. ABSTRACT (200 words or less)

This is volume 3 of the U.S. Nuclear Regulatory Commission (NRC) staff's "Safety Evaluation Report Related to Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada." It documents the review and evaluation of the U. S. Department of Energy's (DOE) Safety Analysis Report (SAR), Chapter 2: Repository Safety After Permanent Closure, provided in DOE's June 3, 2008, license application, as updated by DOE on February 19, 2009. In its application, DOE seeks authorization from the Commission to construct a repository at Yucca Mountain.

The NRC staff finds, with reasonable expectation, that DOE has demonstrated compliance with the NRC regulatory requirements for postclosure safety, including, but not limited to, "Performance objectives for the geologic repository after permanent closure" in 10 CFR 63.113, "Requirements for performance assessment" in 10 CFR 63.114, "Requirements for multiple barriers" in 10 CFR 63.115, and "Postclosure Public Health and Environmental Standards" in 10 CFR Part 63, Subpart L. In particular, the NRC staff finds that the proposed repository at Yucca Mountain (1) is comprised of multiple barriers and (2) based on performance assessment evaluations that are in compliance with applicable regulatory requirements, meets the 10 CFR Part 63, Subpart L limits for individual protection, human intrusion, and separate standards for protection of groundwater.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)	13. AVAILABILITY STATEMENT
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