NUREG-1762

Integrated Issue Resolution Status Report

U.S. Nuclear Regulatory Commission
Office of Nuclear Materials Safety and Safeguards
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ABSTRACT

This report provides background information on the status of prelicensing interactions between the U.S. Department of Energy (DOE) and the U.S. Nuclear Regulatory Commission (NRC) concerning a potential high-level waste geologic repository at Yucca Mountain, Nevada. The NRC staff have, for many years, engaged in extensive interactions with DOE and various stakeholders. In recent years, the interactions focused on what the NRC staff termed key technical issues important to repository performance.

This report provides background information pertaining to the recent interactions with DOE (to October 2001), particularly the technical bases for the staff views presented in the public meetings with DOE from August 2000 to September 2001. The report also documents the information staff considered in formulating their views, including the results of the in-depth review of DOE and contractor documents; the independent work of NRC and its contractor, the Center for Nuclear Waste Regulatory Analyses; published literature; and other publicly available information.

This report may be of value to stakeholders interested in understanding the staff technical rationale for identifying certain information which, if provided by DOE, would address the staff questions concerning the manner in which DOE is responding to the key technical issues.

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EXECUTIVE SUMMARY

Introduction

This report provides background information on the status of prelicensing interactions between the U.S. Nuclear Regulatory Commission (NRC) staff and the U.S. Department of Energy (DOE) concerning a potential high-level waste geologic repository at Yucca Mountain, Nevada. The NRC staff have, for many years, engaged in extensive interactions with DOE and various stakeholders including the State of Nevada, Indian Tribes, affected units of local government, representatives of the nuclear industry, and interested members of the public. In recent years, the interactions focused on what the NRC staff termed key technical issues. Defined by the NRC staff in 1995–1996, the intent of the key technical issues is to focus prelicensing work on those topics most critical to the postclosure performance of the proposed geological repository.

To address and document the key technical issues, the NRC staff initiated a formal issue resolution process that includes reviewing the DOE documents; conducting independent analyses, experiments, and field work; interacting with DOE in public technical meetings; and identifying the information that DOE will need to provide in any potential license application. Over the past several years, the NRC documented the status of issue resolution through individual status reports for each of the key technical issues. More recently, the NRC staff intensified their prelicensing interactions with DOE. During the period August 2000 to September 2001, the NRC staff and DOE held 16 technical exchanges to address and resolve remaining current questions and concerns. The public meetings were used to discuss the status of issue resolution and reach agreements documenting the additional DOE work pertaining to a potential license application.

Results of the intensified interactions have already been presented to DOE in formal letters and public meetings and were summarized in an attachment to the NRC November 13, 2001, letter to DOE, providing the Commission preliminary comments regarding a possible geologic repository at Yucca Mountain.[1]

This report provides additional background information pertaining to the more recent staff interactions with DOE (to October 2001), particularly the technical bases for staff views presented in the public meetings with DOE August 2000 to September 2001. The report also documents the information staff considered in formulating their views, including the results of the in-depth review of DOE and contractor documents; the independent work of NRC and its contractor, the Center for Nuclear Waste Regulatory Analyses (CNWRA); published literature; and other publicly available information. The report uses the review methods and acceptance criteria outlined in the Yucca Mountain Review Plan (NRC, 2002)

The information in this report may be of value to stakeholders interested in understanding the staff technical rationale for identifying certain information which, if provided by DOE, would address the staff questions concerning the manner in which DOE is addressing the key technical issues.

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[1] Meserve, R.A. Letter (November 13, 2001) to R. Card, DOE. Washington, DC: NRC. 2001

Background

In the Nuclear Waste Policy Act of 1982 (1982), the U.S. Congress directed DOE to submit information on site characterization activities to NRC before submittal of a license application for a potential high-level waste geologic repository at Yucca Mountain, Nevada. The U.S. Congress also directed (i) that the NRC preliminary comments concerning the extent to which the at-depth site characterization analysis and the waste form proposal for such site seem sufficient for inclusion in any application that should be submitted by DOE as part of the site recommendation process, and (ii) that NRC shall issue a final decision approving or disapproving the issuance of a construction authorization not later than the expiration of 3 years after the date of the submission of such application (except that NRC may extend such deadline by not more than 12 months).

As a result of this direction, NRC and DOE made issue resolution a major part of the prelicensing interaction specified in the Nuclear Waste Policy Act of 1982 (1982). The NRC staff issue resolution process includes reviewing the DOE documents, interacting with the DOE staff in public technical meetings, and identifying the information DOE will need to provide in any potential license application. The public meetings involve DOE and other stakeholders (including the State of Nevada, Tribal governments, affected units of local governments, and interested members of the public) who have the opportunity to participate. Although public meetings are conducted on a variety of topics, the information presented in this report relates primarily to technical exchanges, which are public meetings to achieve issue resolution. In this context, issues are defined as resolved when there are no further questions at the staff level; however, issue resolution does not signify that a licensing decision has been reached. Additional information (e.g., changes in the DOE design parameters) could raise new questions or comments regarding a previously resolved issue.

The NRC staff risk-informed, performance-based approach to high-level waste disposal makes use of results from the DOE and NRC laboratory and field experiments, natural analog studies, expert elicitations, and performance assessments. In 1996, these activities led to the identification of what the NRC staff termed key technical issues identified as important to the performance of a potential repository. The NRC staff continue to emphasize these key technical issues in the prelicensing interactions with DOE.

As understanding of the site, the potential design and key technical issues evolved through prelicensing interactions with DOE, results from NRC confirmatory studies, and consideration of independent investigations and evaluations by other stakeholders, the individual key technical issues were refined into subissues that more clearly specified important areas that the NRC staff determined DOE needed to address. In the process, NRC made publicly available numerous technical and program status reports that reviewed the DOE site characterization and design work and identified additional information that DOE would need to submit a license application. The NRC staff consistently emphasized that the completeness and acceptance for review of any license application were dependent on the extent to which DOE addressed the key technical issues in preparing a safety case for Yucca Mountain.

In previous years, NRC reported on the status of issue resolution through individual status reports for each of the key technical issues. Beginning in fiscal year 2001, the NRC staff decided that the issue resolution process was mature enough to develop a single Integrated Issue Resolution Status Report that would clearly and consistently reflect the interrelationships among the various key technical issue subissues and the overall resolution status. In addition, it was decided that sections on preclosure topics, performance confirmation, and quality assurance would be added to the Integrated Issue Resolution Status Report. Thus, this report captures the status of the majority of the NRC reviews related to the proposed repository at the Yucca Mountain site up to October 2001.

Report Structure

This report is organized into two main sections: preclosure and postclosure performances of the proposed repository at Yucca Mountain. Information on NRC review of DOE information provided to NRC prior to the end of October 2001 is provided in this report.

Based on 10 CFR Part 63 and review of DOE reports (CRWMS M&O, 2000, 2001), and other support documents, NRC staff preliminarily identified 10 preclosure topics that DOE should address in any future license application regarding the potential high-level waste repository at Yucca Mountain: (i) Site Description As It Pertains to Preclosure Safety Analysis; (ii) Description of Structures, Systems, Components, Equipment, and Operational Process Activities; (iii) Identification of Hazards and Initiating Events; (iv) Identification of Event Sequences; (v) Consequence Analyses; (vi) Identification of Structures, Systems, and Components Important to Safety; Safety Controls; and Measures to Ensure Availability of the Safety Systems; (vii) Design of Structures, Systems, and Components Important to Safety and Safety Controls; (viii) Meeting the 10 CFR Part 20 As Low As Is Reasonably Achievable for Normal Operations and Category 1 Event Sequences; (ix) Plans for Retrieval and Alternate Storage of Radioactive Wastes; and (x) Plans for Permanent Closure and Decontamination, or Decontamination and Dismantlement of Surface Facilities.

The postclosure section of this report is organized according to a set of integrated subissues. The NRC and CNWRA staffs used an integrated subissue approach, adapted from independent performance assessments conducted by NRC, DOE and other stakeholders, in preparing information for many of the technical exchanges August 2000 to September 2001. This approach provides an integrated, transparent issue structure to review the DOE information pertaining to the key technical issues. To clarify the issue structure, charts were constructed to depict elements of a safety review and the relationships among various components of a postclosure performance assessment for the proposed repository at Yucca Mountain (see Section 1.1 for additional details). These charts showed that an efficient way to review the DOE postclosure safety case and its associated performance assessment is to follow the partitioning depicted in Figure 1. This partitioning is primarily based on the natural progress of moisture downward to the repository level, various processes in the vicinity of the emplaced waste, and potential radionuclide release and transport to a receptor group distant from the Yucca Mountain site. Processes and events that could potentially disrupt the repository are also considered. The topics at the most detailed level of decomposition (14 in all) in Figure 1 are called integrated subissues or model abstractions, mainly because each integrated subissue draws information from multiple key technical issues. The integrated subissues represent an interdisciplinary and logical approach to reviewing the DOE performance assessment. The integrated subissue format and the interdisciplinary questions posed for each of the integrated subissues assist the staff in more formally integrating the related processes and effects of the key technical issue subissues. This structure was used by the staff in developing the postclosure portions of the Yucca Mountain Review Plan (NRC, 2002)]. For consistency, this Integrated Issue Resolution Status Report follows the same structure.

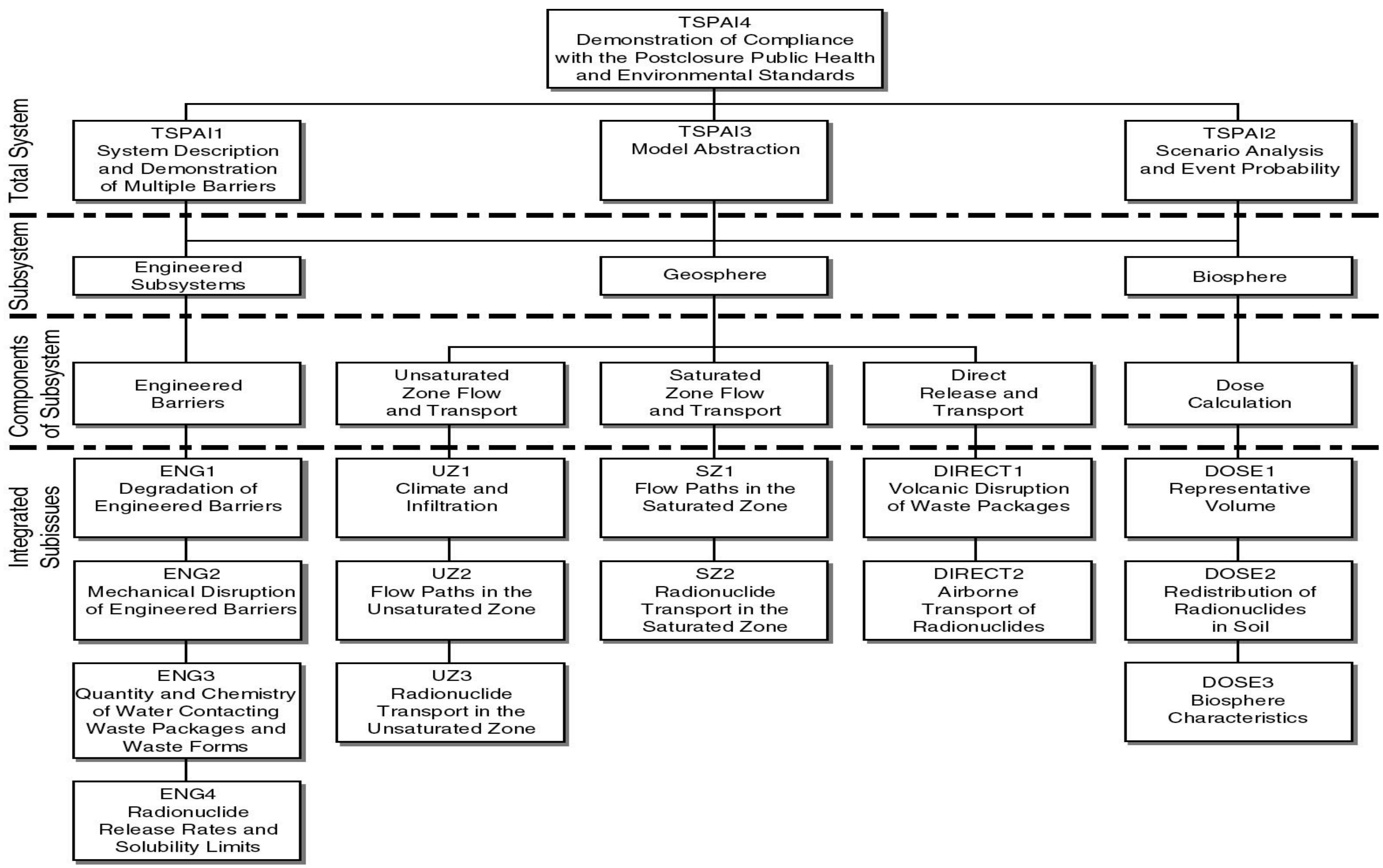

Figure 1. Components of Postclosure Performance Assessment Review

Preclosure Summary

Because significant experience already exists at NRC in regulating safety during construction and operation of other nuclear facilities, the NRC staff emphasized developing licensing review capabilities with respect to postclosure during the early years of the program. Beginning in fiscal year 2000, however, the importance of preclosure safety was elevated in view of the DOE plans to proceed with a design and submit a possible site recommendation.

During past DOE and NRC preclosure interactions and conversations, technical issues associated with preclosure topics (i) through (vii) have been discussed. Technical concerns will continue to be identified and clarified as the review of DOE documents proceeds. Not all the preclosure technical issues identified in this report were addressed in the July 2001 Technical Exchange Meeting on Preclosure Safety.[2] While the issue resolution process in the preclosure area moves forward, NRC will (i) conduct Appendix 7 meetings with DOE to monitor the progress of addressing the agreements reached during the previous technical exchange meetings; (ii) continue review of the DOE preclosure-related documents when they become available and identify technical concerns, if any; (iii) conduct technical exchange meetings to discuss the remaining preclosure concerns identified thus far through reviewing DOE preclosure-related documents; and (iv) conduct independent preclosure safety analyses, as needed, to identify potential omissions and weaknesses in the DOE design and related safety case and to better risk-inform issue resolution activities.

Postclosure Summary

Consistent with the issue resolution process, NRC staff intensified its prelicensing interactions with DOE during the last two years to address and resolve remaining questions. Since August 2000, DOE and NRC have held numerous technical exchanges focused specifically on issues relevant to these questions. Multi-day public meetings were used to discuss the status of issue resolution. Results from this increased prelicensing interaction have been documented in formal letters to DOE and in agreements reached in public meetings between DOE and NRC. These activities were summarized in an attachment to the NRC November 13, 2001, letter to DOE.

As the issue resolution process in the postclosure area moves forward, NRC will (i) conduct technical exchange and Appendix 7 meetings with DOE to discuss and monitor the progress of addressing the agreements reached during the previous technical exchange meetings; (ii) continue review of the DOE postclosure-related documents when they become available and identify technical concerns, if any; and (iii) conduct independent analyses, as needed, to identify potential omissions and weaknesses in the DOE design and related safety case and to better risk-inform issue resolution activities.

[2]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

Summary

This report provides background information on the status of the NRC staff issue resolution activities pertaining to a potential high-level waste repository at Yucca Mountain. The report, which covers staff activities prior to October 2001, provides a description of the technical bases supporting staff identification of information from DOE to address the staff key technical issues. For the NRC preliminary views on the DOE information, readers should consult the Commission's November 13, 2001, letter to DOE.

References

CRWMS M&O. "Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Consideration." TDR–WIS–RL–000001. Revision 04 ICN 01. Las Vegas, Nevada: TRW Environmental Safety Systems, Inc. 2000.

———. "Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation." TDR–MGR–SE–000009. Revision 00 ICN 03. Las Vegas, Nevada: DOE. 2001.

NRC. NUREG–1804, "Yucca Mountain Review Plan—Draft Report for Comment." Revision 2. Washington, DC: NRC. March 2002.

Nuclear Waste Policy Act of 1982. Pub. L. 97–425. 96 Stat. 2201 (1982).

PREFACE

This Integrated Issue Resolution Status Report documents the prelicensing resolution status of preclosure and postclosure technical issues related to the proposed high-level nuclear waste repository at Yucca Mountain. The process of issue resolution during the prelicensing phase is based on review of information (i) contained in the U.S. Department of Energy (DOE) and DOE contractor documents; (ii) obtained during technical exchanges, which are meetings open to the public; (iii) obtained from independent investigations conducted by the U.S. Nuclear Regulatory Commission (NRC) and its contractor, the Center for Nuclear Waste Regulatory Analyses (CNWRA); and (iv) available from a variety of open literature sources. The Nuclear Waste Policy Act of 1982 (1982) directs NRC to engage DOE in prelicensing consultations.

This Integrated Issue Resolution Status Report tracks progress toward the resolution of issues and provides this information in a single document to interested parties. NRC intends to update this report when sufficient new information becomes available. Because of the broad scope of this report, however, publication will always lag a few months behind availability of the information. For example, this version of the report includes technical information through October 2001. This version includes regulatory information through March 2002, such as the final U.S. Environmental Protection Agency Standard for Yucca Mountain at 40 CFR Part 197, the final NRC regulations at 10 CFR Part 63, the final DOE regulations at 10 CFR Part 963, and the NRC Yucca Mountain Review Plan (NRC, 2002). Information from other sources that may become available will be included in the next update of this report.

The reader should also note that in this version of the report, some sections are absent and others are incomplete. For example, only certain sections are included in Chapter 2, which is devoted to repository safety before permanent closure. All other sections of Chapter 2 will be completed after future technical exchanges with DOE on preclosure issues.

References

NRC. NUREG–1804, "Yucca Mountain Review Plan—Draft Report for Comment." Revision 2. Washington, DC: NRC. March 2002.

Nuclear Waste Policy Act of 1982. Pub. L. 97–425. 96 Stat. 2201 (1982).

ACKNOWLEDGMENTS

This Integrated Issue Resolution Status Report is a joint product of the U.S. Nuclear Regulatory Commission (NRC) and the Center for Nuclear Waste Regulatory Analyses (CNWRA). The CNWRA performed its work under contract No. NRC–02–97–009. Although specific staff were responsible for developing the text included in this report, others contributed information through attending technical exchanges and by authoring other reports, the contents of which are abstracted in various sections of this report. Various staff from both organizations participated as technical, editorial, or programmatic reviewers of the sections. The report was coordinated by James W. Andersen at NRC and Budhi Sagar at CNWRA. They thank all participants for their hard work and patience in preparing this first integrated issue resolution product.

This document was produced following the quality assurance requirements described in the CNWRA Quality Assurance Manual. Data and analyses from many sources other than the CNWRA are included in this document. Referenced sources of data and analyses should be consulted for determining levels of quality assurance.

1 INTRODUCTION

1.1 Background and Report Structure

This report documents the prelicensing resolution status of preclosure and postclosure issues. Issue resolution at the staff level has been determined by the U.S. Nuclear Regulatory Commission (NRC) staff to be important to increasing the likelihood of a high-quality license application for a proposed high-level waste repository at Yucca Mountain if, after a presidential decision on site suitability, the U.S. Department of Energy (DOE) decides to submit a license application. A license application is considered high quality if it contains sufficient information for making regulatory decisions: high quality does not imply NRC judgment regarding the regulatory decisions , which will be made after review of any license application. In the Nuclear Waste Policy Act of 1982 (1982), the U.S. Congress directed DOE to submit information on site characterization activities to NRC before submittal of a license application for a potential high-level waste geologic repository at Yucca Mountain, Nevada. The U.S. Congress also directed (i) that the NRC preliminary comments concerning the extent to which the at-depth site characterization analysis and the waste form proposal for such site seem sufficient for inclusion in any application that should be submitted by DOE as part of the site recommendation process, and (ii) that NRC shall issue a final decision approving or disapproving the issuance of a construction authorization not later than the expiration of 3 years after the date of the submission of such application (except that NRC may extend such deadline by not more than 12 months).

As a result of this direction, NRC and DOE made issue resolution a major part of the prelicensing interaction specified in the Nuclear Waste Policy Act of 1982 (1982). Prelicensing interactions take the form of public meetings at which all stakeholders including State of Nevada, Tribal governments, affected units of local governments, and interested members of the public have the opportunity to participate. Issue resolution is based on an in-depth review of the DOE and contractor documents; independent work of NRC and its contractor, the Center for Nuclear Waste Regulatory Analyses (CNWRA); published literature; and other publicly available information. The prelicensing consultations and the issue resolution process are in conformance with the NRC efforts to streamline its high-level waste program (NRC, 1999a) and prepare for an efficient and competent review of any license application that the DOE may submit.

It is the responsibility of DOE to ensure that any future license application is complete in all respects. Therefore, DOE must fully address all aspects of repository performance in an acceptable manner in its license application. In addition to an acceptance review, the NRC staff will perform an audit review of all information presented in the license application and choose for detailed review those topics that are most important to overall repository performance. The selection of topics for detailed license application review or as focal points during the prelicensing issue resolution process, however, does not mean DOE should include only those topics in its license application. DOE has the responsibility to present a high-quality application that will demonstrate compliance with all NRC regulatory requirements. For example, in addition to adequately considering in its safety case the features, events, and processes that affect repository safety, DOE must also provide adequate technical bases for the exclusion of features, events, and processes that are deemed to be not important. The risk-informed audit nature of the staff review does not relieve DOE of these obligations.

Introduction

In 1995–1996, the NRC high-level waste program was realigned to focus prelicensing work on those topics most critical to the postclosure performance of the proposed geologic repository. At that time, the staff identified 10 postclosure key technical issues (Sagar, 1997) and their associated subissues as listed in Table 1.1-1.

Of the 10 key technical issues, the first 9 are directly related to the objective of this report; the last pertains to development of the NRC regulation in 10 CFR Part 63.[1] A brief discussion of 10 CFR Part 63, as well as other applicable regulations, is included in Section 1.3. Technical issues related to preclosure safety were not defined in the mid-1990s, but they are included in this report as explained in the following.

The status of the NRC staff work on all 10 key technical issues was documented in a 1997 report (Sagar, 1997). Starting with fiscal year 1997, it was decided to document issue resolution for each key technical issue in individual reports; Revision 0 of the Issue Resolution Status Reports was issued in 1997–1998 except for the Radionuclide Transport Key Technical Issue, work on which was delayed, and the Activities Related to the Development of U.S. Nuclear Regulatory Commission Yucca Mountain Regulations Key Technical Issue that was documented in the proposed rule. Taking into account changes to the DOE overall approach and new information provided in the DOE documents, these reports were updated every year, reaching Revision 3 in the year 2000. In the latter part of fiscal year 2000, DOE and NRC agreed to hold technical exchanges and management meetings focused specifically on issue resolution and to reach agreement on what additional information DOE needed to provide to resolve the key technical issues. Beginning in fiscal year 2001, the NRC management decided that the issue resolution process was mature enough to develop a single Integrated Issue Resolution Status Report that would clearly and consistently reflect the interrelationships between the various key technical issue subissues, integrated subissues, and the overall resolution status. In addition, it was decided that sections on preclosure issues, performance confirmation, and quality assurance would be added to the Integrated Issue Resolution Status Report. In this way, an Integrated Issue Resolution Status Report would capture the status of the majority of the NRC reviews related to the proposed repository at the Yucca Mountain site. This document is the result of implementing that integration initiative.

In the issue resolution status reports for individual key technical issues, issue resolution was documented subissue by subissue. The nine key technical issues represent major processes and related staff concerns regarding the postclosure safety of a geologic repository. Some processes were shared among key technical issues, making discussion and resolution cumbersome. As the NRC and CNWRA staffs conducted independent performance assessment exercises over the years and reviewed similar exercises by the U.S. Department of Energy Yucca Mountain Project, Electric Power Research Institute, the U.S. Department of Energy Waste Isolation Pilot Project, and other international programs, it became clear that a more integrated and transparent issue structure was needed.

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[1]Throughout this document, in-text citations for the Code of Federal Regulations (CFR) will include the title number, CFR, and the part or section numbers only. Also, CFRs will not be listed in References.

Table 1.1-1. Key Technical Issues and Associated Subissues

Key Technical Issue	Associated Subissues					
Igneous Activity	IA1—Probability of Igneous Activity	IA2—Consequences of Igneous Activity	—	—	—	—
Structural Deformation and Seismicity	SDS1—Faulting What are the viable models of faults and fault displacements at Yucca Mountain?	SDS2—Seismicity What are the viable models of seismic sources and seismic ground motions at Yucca Mountain?	SDS3—Fracturing and Structural Framework of the Geologic Setting What are the viable models of fractures and structural controls of flow at Yucca Mountain?	SDS4—Tectonic Framework of the Geologic Setting What are the viable tectonic models and crustal conditions at Yucca Mountain?	—	—
Evolution of Near-Field Environment	ENFE1—Effects of Coupled Thermal-Hydrologic-Chemical Processes on Seepage and Flow	ENFE2–Effects of Coupled Thermal-Hydrologic-Chemical Processes on the Waste Package Chemical Environment	ENFE3—Effects of Coupled Thermal-Hydrologic-Chemical Processes on the Chemical Environment for Radionuclide Release	ENFE4—Effects of Coupled Thermal-Hydrologic-Chemical Processes on Radionuclide Transport Through Engineered and Natural Barriers	ENFE5—Effects of Coupled Thermal-Hydrologic-Chemical Processes on Potential Nuclear Criticality in the Near Field	—
Container Life and Source Term	CLST1—The Effects of Corrosion Processes on the Lifetime of the Containers	CLST2—The Effects of Phase Instability of Materials and Initial Defects on the Mechanical Failure and Lifetime of the Containers	CLST3—The Rate at Which Radionuclides in Spent Nuclear Fuel Are Released from the Engineered Barrier Subsystem Through He Oxidation and Dissolution of Spent Fuel	CLST4—The Rate at Which Radionuclides in High-Level Waste Glass Are Leached and Released from the Engineered Barrier Subsystem	CLST5—The Effect of In-Package Criticality on Waste Package and Engineered Barrier Subsystem Performance	CLST6—The Effects of Alternate Engineered Barrier Subsystem Design Features on Container Lifetime and Radionuclide Release from the Engineered Barrier Subsystem

Table 1.1-1. Key Technical Issues and Associated Subissues (continued)

Key Technical Issue	Associated Subissues					
Thermal Effects on Flow	TEF1—Features, Events, and Processes Related to Thermal Effects on Flow	TEF2—Thermal Effects on Temperature, Humidity, Saturation, and Flux	—	—	—	—
Repository Design and Thermal-Mechanical Effects	RDTME1—Design Control Process Implementation of an effective design control process within the overall Quality Assurance program	RDTME2—Seismic Design Methodology Design of the geologic repository operations area for the effects of seismic events and direct fault disruption [including implications for drift stability, key aspects of emplacement configuration (i.e., fault offset distance, retrievability, and waste package damage)]	RDTME3—Thermal-Mechanical Effects Consideration of thermal-mechanical effects on underground facility design and performance (including implications for drift stability, key aspects of emplacement configuration that may influence thermal loads and associated thermomechanical effects, retrievability, the change in geometry and flow into and out of emplacement drifts, and fault setback distance)	RDTME4—Design and Long-Term Contribution of Seals to Performance Design and long-term contribution of repository seals in meeting the postclosure performance objectives (including implications for inflow of water and release of radionuclides to the environment)	—	—
Total System Performance Assessment and Integration	TSPAI1—System Description and Demonstration of Multiple Barriers	TSPAI2—Scenario Analysis and Event Probability	TSPAI3—Model Abstraction	TSPAI4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards	—	—

Table 1.1-1. Key Technical Issues and Associated Subissues (continued)

Key Technical Issue	Associated Subissues					
Unsaturated and Saturated Flow Under Isothermal Conditions	USFIC1—Climate Change What is the likely range of future climates at Yucca Mountain?	USFIC2—Hydrologic Effects of Climate Change What are the likely effects of climate change?	USFIC3—Shallow Infiltration What is the estimated amount and spatial distribution of present day shallow infiltration?	USFIC4—Deep Percolation What is the estimated amount and spatial distribution of percolation through the proposed repository horizon (present day, and through the period of repository performance)?	USFIC5—Saturated Zone What are the ambient flow conditions in the saturated zone, and what are the likely dilution mechanisms?	USFIC6—Matrix Diffusion To what degree does matrix diffusion occur in the unsaturated and saturated zones?
Radionuclide Transport	RT1—Radionuclide Transport Through Porous Rock	RT2—Radionuclide Transport Through Alluvium	RT3—Radionuclide Transport Through Fractured Rock	RT4—Nuclear Criticality in the Far Field	—	—
Activities Related to Development of the U.S. Nuclear Regulatory Commission Yucca Mountain Regulations	—	—	—	—	—	—

To clarify the issue structure, charts were constructed to depict components of a safety review (Figure 1.1-1) and the relationships among various components of a postclosure performance assessment for the proposed repository at Yucca Mountain (Figure 1.1-2). These charts showed that an efficient way to review the DOE postclosure safety case and its associated performance assessment is to follow the partitioning depicted in Figure 1.1-2. This partitioning is primarily based on the natural progress of potential radionuclide release and transport to a receptor group at the Yucca Mountain site. The topics at the most detailed level of decomposition (14 in all) in Figure 1.1-2 are called integrated subissues or model abstractions, mainly because each integrated subissue draws information from multiple key technical issues. The integrated subissues represent an interdisciplinary and logical approach to reviewing the DOE performance assessment. The integrated subissue format and the interdisciplinary questions posed for each of the integrated subissues should more formally integrate the contribution of the key technical issue subissues. Therefore, it was decided to adopt this structure in developing the postclosure portions of the standard review plan [known as the Yucca Mountain Review Plan (NRC, 2002)] applicable to the proposed repository at Yucca Mountain. NRC (2002) documents guidance to the staff for the review of any license application submitted by DOE. NRC (2002) documents the methods to be used for review and the criteria to be applied for accepting the DOE analyses and suggests language for staff findings. To create traceability and transparency through better correlation of current reviews with future reviews of the potential license application, the same structure is also followed for the postclosure portion of this document. The generic review methods used for developing this Integrated Issue Resolution Status Report are described in Section 1.5.

It is emphasized that this document provides a status report on progress toward issue resolution at the staff level. It is not a licensing review, and no conclusions are drawn with respect to whether or not the Yucca Mountain site is licensable or whether it meets applicable NRC regulatory requirements. The licensing review will begin only after a license application is docketed. The NRC staff review of a future license application will be documented in a safety evaluation report.

The geologic repository would be a first-of-a-kind facility, and there is little experience regarding its postclosure long-term performance. For this reason, and also because significant experience already exists at NRC in regulating safety during construction and operation of other nuclear facilities, the staff emphasized developing licensing review capabilities with respect to postclosure during the early years of the program. Beginning in fiscal year 2000, however, the importance of preclosure safety was elevated in view of the DOE plans to proceed with a design and submit a possible site recommendation in 2001. Although the preclosure program is not as mature as the postclosure program, preclosure safety is important as well as postclosure safety. Accordingly, Chapter 2 provides a status of the preclosure issues. The 10 preclosure topics defined for this purpose are (i) Site Description As It Pertains to Preclosure Safety Analysis; (ii) Description of Structures, Systems, Components, Equipment, and Operational Process Activities; (iii) Identification of Hazards and Initiating Events; (iv) Identification of Event Sequences; (v) Consequence Analyses; (vi) Identification of Structures, Systems, and Components Important to Safety; Safety Controls; and Measures to Ensure Availability of the Safety Systems; and (vii) Design of Structures, Systems, and

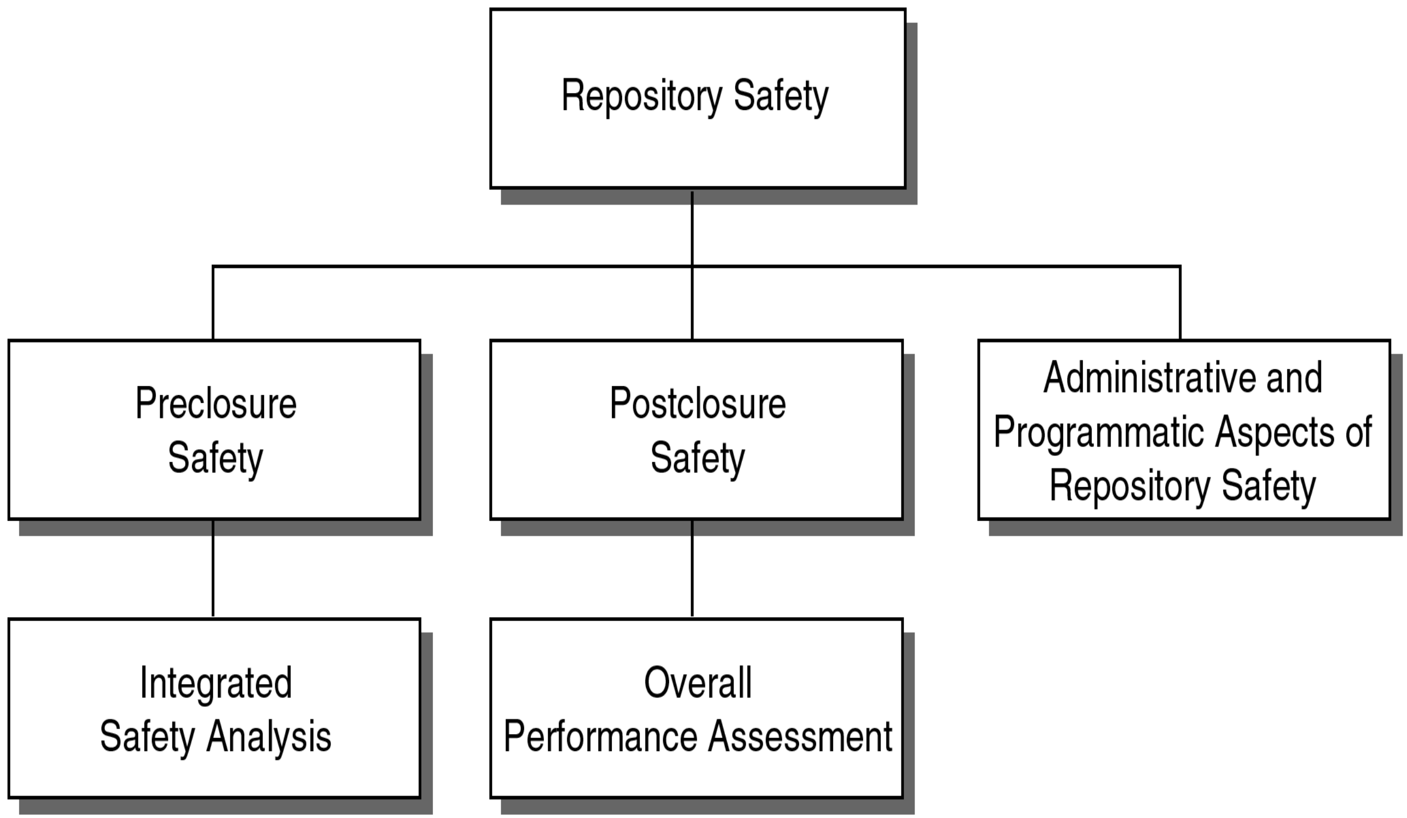

Figure 1.1-1. Review Components of Repository Safety

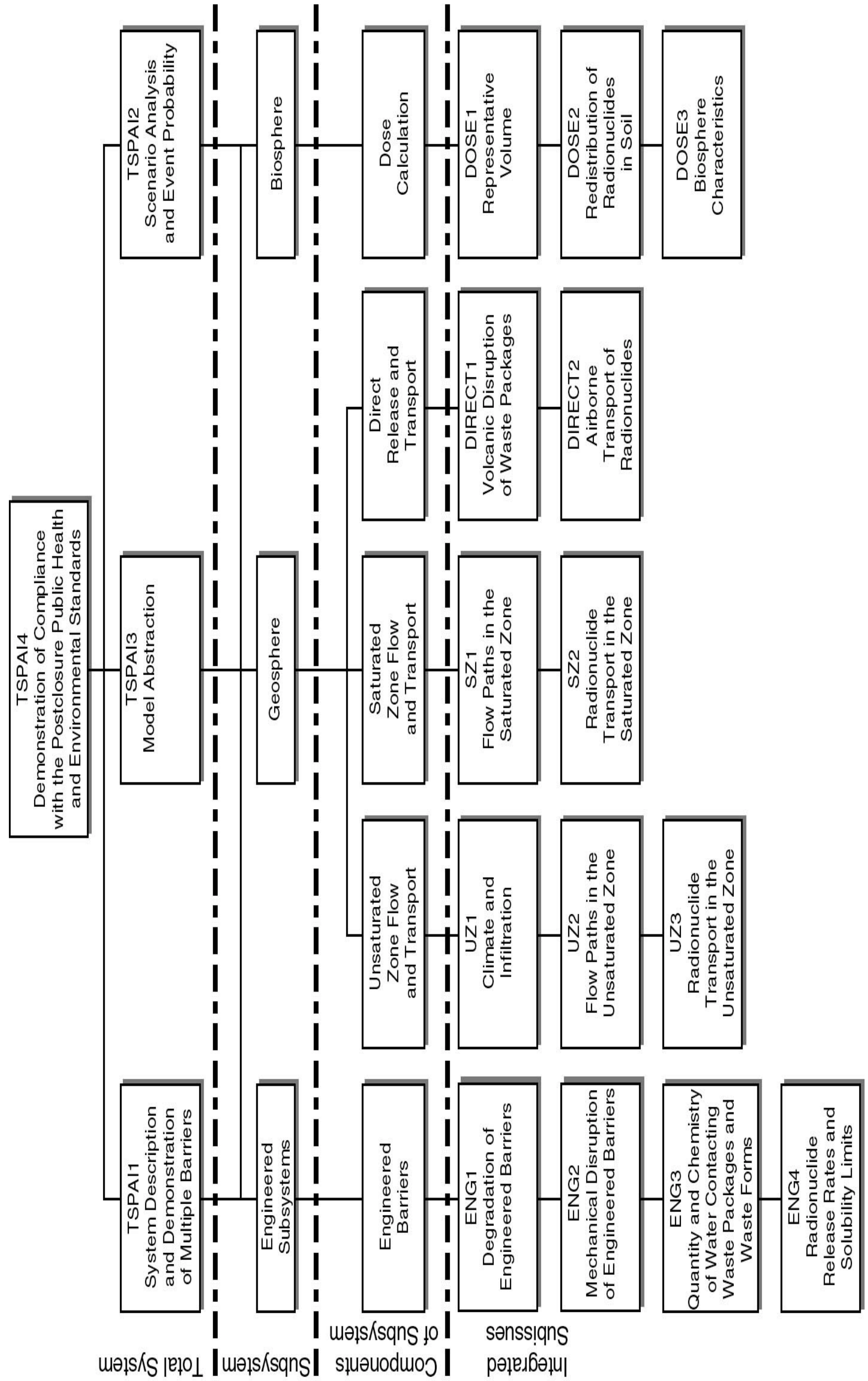

Figure 1.1-2. Components of Postclosure Performance Assessment Review

Components Important to Safety and Safety Controls; (viii) Meeting the 10 CFR Part 20 as Low as is Reasonably Achievable Requirements for Normal Operations and Category 1 Event Sequences; (ix) Plans for Retrieval and Alternate Storage of Radioactive Wastes; and (x) Plans for Permanent Closure and Decontamination, or Decontamination and Dismantlement of Surface Facilities.[2]

Chapter 3 of this report documents the status of issue resolution for the 14 integrated subissues for postclosure performance. To put the review of the integrated subissues in the context of the total system performance assessment, four additional review issues are defined (Figure 1.1-2): (i) TSPAI1—System Description and Demonstration of Multiple Barriers; (ii) TSPAI2—Scenario Analysis and Event Probability; (iii) TSPAI3—Model Abstraction; and (iv) TSPAI4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards. These topics are also discussed in Chapter 3. As noted previously, each integrated subissue draws information from various key technical issue subissues, which are clearly identified in the text; their relationships are also described in Table 1.1-2.

The NRC regulations call for DOE to conduct performance confirmation activities. The objective of performance confirmation is to acquire information by conducting monitoring, *in-situ* experiments, laboratory experiments, and analyses that will provide confidence that the repository will continue to perform both during preclosure and postclosure periods in a safe manner. Chapter 4 discusses this aspect of the repository program. The DOE research and development programs to resolve any safety questions are also discussed in Chapter 4. DOE published a performance confirmation plan [Civilian Radioactive Waste Management System Management and Operating Contractor (CRWMS M&O), 2000a] as discussed in Section 4.2.

Confidence in the estimated preclosure and postclosure safety indicators and performance measures will be based in part on the premise that data were collected and analyses conducted following the Quality Assurance program required by NRC and akin to that stipulated in Appendix B of 10 CFR Part 50. The NRC has followed the development and implementation of the Quality Assurance program for the quality-affecting activities of the Yucca Mountain project. This was accomplished by participating as observers during quality assurance audits conducted by DOE and assessing the status of the Quality Assurance program through periodic meetings. The quality assurance aspects of the Yucca Mountain project are discussed in Chapter 5.

Finally, Chapter 6 provides a summary and conclusions. The DOE and NRC key technical issue exchange agreements are listed in Appendix A.

On November 13, 2001, NRC submitted preliminary comments to DOE on the sufficiency of the DOE at-depth site characterization analysis and waste form proposal. The NRC preliminary comments summarized the many years of extensive prelicensing interaction among the NRC staff, DOE, and various stakeholders, which served as the basis of the NRC comments.

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[2]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange on Pre-Closure Issues." Letter (April 27) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

Table 1.1-2. Relationships Between Integrated Subissues and Key Technical Issues

Key Technical Issue Subissue	Integrated Subissues													
	ENG1	ENG2	ENG3	ENG4	UZ1	UZ2	UZ3	SZ1	SZ2	Direct1	Direct2	Dose1	Dose2	Dose3
USFIC1					■									■
USFIC2					■			■						■
USFIC3					■									■
USFIC4			■			■	■							
USFIC5								■	■			■		■
USFIC6							■	■	■					
TEF1			■			■								
TEF2	■		■			■	■							
ENFE1			■			■								
ENFE2	■		■											
ENFE3	■		■	■			■							
ENFE4			■	■			■							
ENFE5	■		■	■										
CLST1	■	■	■											
CLST2	■	■								■				
CLST3			■	■										
CLST4			■	■										
CLST5	■	■	■	■					■					
CLST6	■	■	■	■										
RT1						■	■	■	■					
RT2							■	■	■					
RT3						■	■	■	■					■
RT4			■				■		■					
TSPAI1	■	■	■	■	■	■	■	■	■	■	■	■	■	■
TSPAI2	■	■	■	■	■	■	■	■	■	■	■	■	■	■
TSPAI3	■	■	■	■	■	■	■	■	■	■	■	■	■	■
TSPAI4	■	■	■	■	■	■	■	■	■	■	■	■	■	■
IA1		■								■				
IA2		■								■	■		■	■
SDS1		■						■						
SDS2		■												
SDS3		■	■		■	■	■	■	■					
SDS4		■												
RDTME1		■												
RDTME2		■				■								
RDTME3	■	■	■			■								
RDTME4														

ENG1 ENG–Degradation of Engineered Barriers
ENG2 ENG–Mechanical Disruption of Engineered Barriers
ENG3 ENG–Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms
ENG4 ENG–Radionuclide Release Rates and Solubility Limits
UZ1 GEO–Climate and Infiltration
UZ2 GEO–Flow Paths in the Unsaturated Zone
UZ3 GEO–Radionuclide Transport in the Unsaturated Zone
SZ1 GEO–Flow Paths in the Saturated Zone
SZ2 GEO–Radionuclide Transport in the Saturated Zone
Direct1 GEO–Volcanic Disruption of Waste Packages
Direct2 GEO–Airborne Transport of Radionuclides
Dose1 BIO–Representative Volume
Dose2 BIO–Redistribution of Radionuclides in Soil
Dose3 BIO–Biosphere Characteristics

Note: Shaded areas indicate key technical issue subissues and integrated subissues relationships.

The comments, mandated by the Nuclear Waste Policy Act of 1982 (1982), accompanied the DOE site recommendation submitted in February 2002 to the President of the United States. This report provides additional background information pertaining to the staff more recent interactions with DOE (to October 2001), particularly the technical bases for staff views presented in the public meetings with DOE August 2000 to September 2001. The report also documents the information staff considered in formulating their views, including the results of the in-depth review of DOE and contractor documents; the independent work of NRC and its contractor, the Center for Nuclear Waste Regulatory Analyses (CNWRA); published literature; and other publicly available information.

Staff intend to publish an updated Integrated Issue Resolution Status Report approximately once a year until the beginning of any licensing review. As DOE submits information in response to the agreements reached at technical exchanges, however, staff will update material in this report as soon as possible. Based on these updates, staff will determine whether the material submitted by DOE is adequate to resolve the issue or whether additional information is needed. If additional information is needed, a request for the information will be prepared and provided to DOE.

1.2 Prelicensing Issue Resolution Process

The NRC strategic plan (2000) calls for the early identification and resolution, at the staff level, of issues before the receipt of a potential license application to construct a geologic repository. The principal means for achieving this goal is through prelicensing interaction with DOE.

As previously mentioned, in August 2000, DOE and NRC agreed to hold technical exchanges focused specifically on issue resolution. The purpose of issue resolution is to assure that sufficient information is available on an issue to enable NRC to conduct a review of a proposed license application. Resolution at the staff level does not preclude an issue from being raised and considered during the licensing proceedings and does not predecide the NRC staff evaluation of that issue after its review of any license application. Issue resolution at the staff level, during prelicensing, is achieved when the staff has no further questions or comments at a point in time regarding how DOE is addressing an issue. The discussions recorded during the technical exchanges reflect the current understanding of issues most important to repository performance by the NRC staff. This understanding is based on all information available prior to the meetings and includes limited, focused, and risk-informed reviews of selected portions of recently provided DOE documents (e.g., analysis and model reports and process model reports). Additional information (e.g., changes in design parameters) could raise new questions or comments regarding a previously resolved issue.

Three categories of issue resolution are defined by the NRC: (i) closed, (ii) closed-pending, and (iii) open. Issues are closed if the DOE approach and available information acceptably address staff questions such that no information beyond what is currently available will likely be required for regulatory decision making at the time of any license application. Issues are closed-pending if the DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (through specified testing or analysis), acceptably addresses the NRC questions so that no information beyond that provided, or agreed to, will likely be required at the time of a potential license application. Issues are open if NRC has identified questions regarding the DOE approach or information and DOE has not yet acceptably addressed the questions or agreed to provide the necessary additional information in a potential license application. As a result of technical exchanges up to the October 2001 cut-off date for this document, DOE and NRC reached agreements pertaining to a subset of the nine postclosure key technical issues and their associated subissues and the preclosure issues. The status of each key technical issue subissue is presented in Table 1.1-3. The agreements reached during the technical exchanges are included in Appendix A.

NRC considers all issues open, in terms of a potential licensing decision, unless and until DOE submits a high-quality license application, the staff completes its independent safety review and issues a safety evaluation report, NRC provides an opportunity for a hearing on issues raised by the parties, and NRC makes its final determination of whether the DOE license application meets the NRC regulations. Any NRC decision will be based on all the information available at that time.

To facilitate tracking issue resolution status and to aid in future discussions, the DOE and NRC technical exchange agreements are assigned to integrated subissues (see Appendix A). Note that, in addition to the 14 integrated subissues shown in Figure 1.1-2, the assignment of agreements also includes the additional Total System Performance Assessment and Integration and Preclosure Subissues defined in Section 1.1.

1.3 Regulations Applicable to a Potential High-Level Waste Repository at Yucca Mountain

Following is a brief history of regulations and a discussion of the main principles included in the standards and regulations. Figure 1.1-3 provides a timeline for pertinent rulemaking (adapted from CRWMS M&O, 2000b).

The Nuclear Waste Policy Act of 1982 (1982) established the national policy and defined the responsibilities of various federal agencies for the safe disposal of spent nuclear fuel, high-level waste, and transuranic radioactive waste (referred to collectively as high-level waste in this report) generated mainly as a result of commercial power production and defense activities. According to the Nuclear Waste Policy Act of 1982 (1982), the DOE is responsible for siting, building, operating, and closing an underground geologic repository; the U.S. Environmental Protection Agency (EPA) has the responsibility of setting generally applicable environmental radiation protection standards based on authority established under other laws; and the NRC must implement the EPA standards by incorporating them into its regulations and must decide whether to authorize construction, operation, and closure of a repository.

In 1985, EPA established generic standards for the management, storage, and disposal of high-level waste in 40 CFR Part 191 (50 FR 38066, September 19, 1985). NRC developed its regulations in 10 CFR Part 60. These standards and regulations were intended to apply to all

Table 1.1-3. Status of Key Technical Issue Subissues Resolutions						
Key Technical Issue	**Subissue 1**	**Subissue 2**	**Subissue 3**	**Subissue 4**	**Subissue 5**	**Subissue 6**
Unsaturated and Saturated Flow Under Isothermal Conditions	Closed	Closed	Closed-Pending	Closed-Pending	Closed-Pending	Closed-Pending
Igneous Activity	Closed-Pending	Closed-Pending	N/A	N/A	N/A	N/A
Container Life and Source Term	Closed-Pending	Closed-Pending	Closed-Pending	Closed-Pending	Closed-Pending	Closed-Pending
Structural Deformation and Seismicity	Closed-Pending	Closed-Pending	Closed-Pending	Closed	N/A	N/A
Radionuclide Transport	Closed-Pending	Closed-Pending	Closed-Pending	Closed-Pending	N/A	N/A
Thermal Effects on Flow	Closed-Pending	Closed-Pending	N/A	N/A	N/A	N/A
Evolution of the Near-Field Environment	Closed-Pending	Closed-Pending	Closed-Pending	Closed-Pending	Closed-Pending	N/A
Repository Design and Thermal-Mechanical Effects	Closed	Closed-Pending	Closed-Pending	Closed	N/A	N/A
Total System Performance Assessment and Integration	Closed-Pending	Closed-Pending	Closed-Pending	Closed-Pending	N/A	N/A

appropriate facilities in the United States, including the proposed high-level waste repository in Yucca Mountain, Nevada. In 1987, the U.S. Court of Appeals for the First Circuit Court invalidated the standard and remanded it to EPA (Natural Resources Defense Council, Inc., 1987). Also in 1987, the Nuclear Waste Policy Act of 1982 (1982) was amended by, among other actions, designating Yucca Mountain, Nevada, as the only potential site to be characterized for a high-level waste repository.

In 1992, Congress directed EPA, in Section 801 of the Energy Policy Act of 1992 (1992), to contract with the U.S. National Academy of Sciences to advise EPA on the appropriate technical basis for public health and safety standards governing a potential repository at Yucca Mountain. On August 1, 1995, the U.S. National Academy of Sciences Committee on Technical Basis for Yucca Mountain Standards issued its report Technical Bases for Yucca Mountain

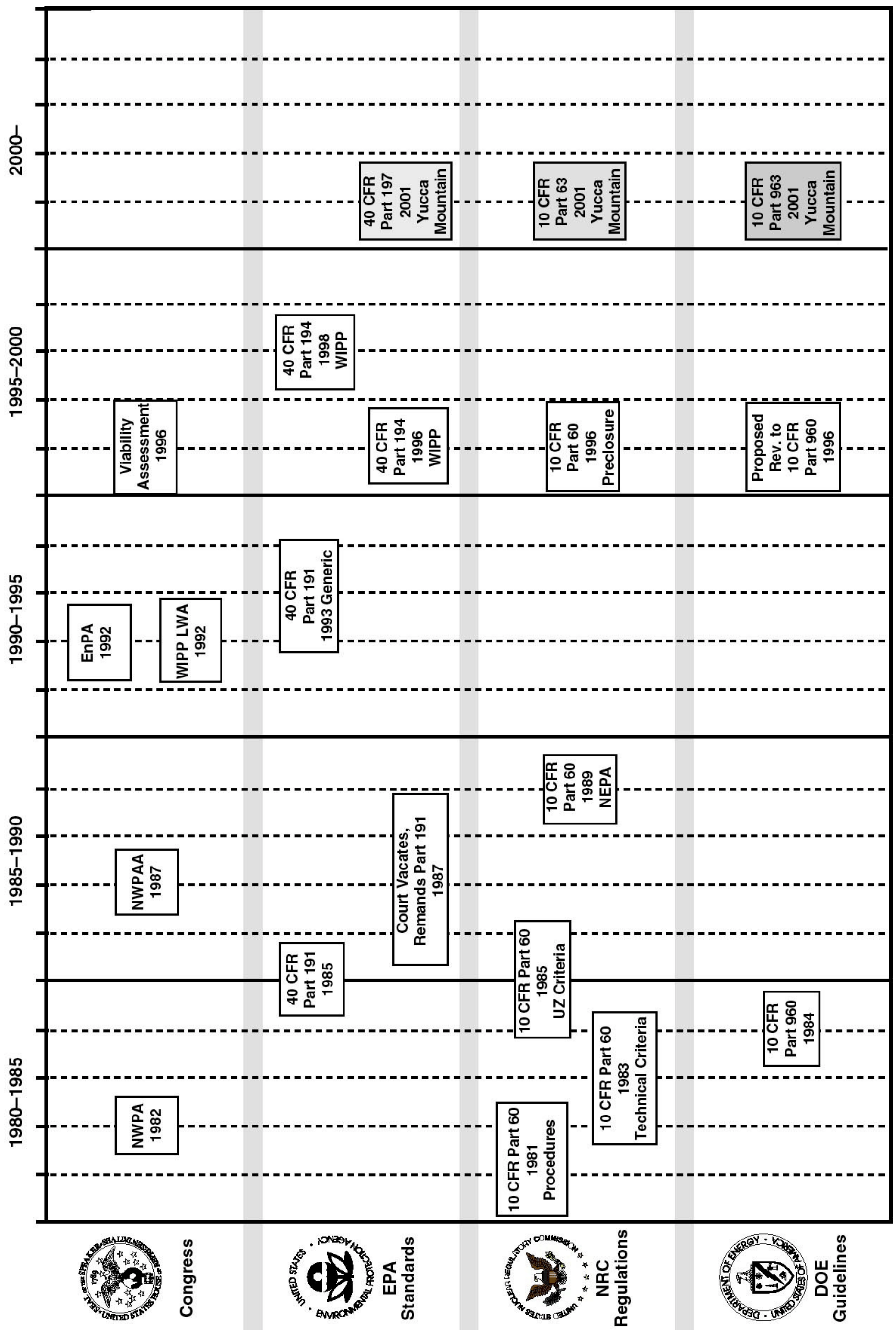

Figure 1.1-3. Timeline of Legislative and Regulatory Events, 1980–2000

Standards (National Research Council, 1995). EPA issued its final standards applicable to Yucca Mountain in a new 40 CFR Part 197 on June 13, 2001. NRC prepared its final regulations based on careful review and consideration of the public comments received on its proposed rule and the statutory direction for NRC to adapt its technical criteria to be consistent with final EPA standards. NRC published its final regulations in a new 10 CFR Part 63 on November 2, 2001. These regulations include criteria for long-term repository performance as well as licensing procedures, records and reporting, monitoring and testing programs, performance confirmation, quality assurance, personnel training and certification, and emergency planning.

EPA Standards

A brief summary of key aspects of the EPA standards is provided next.

Radiation Standards: EPA specified radiation standards for the operational phase of repository development (i.e., the period of time during which waste is brought to the site and placed in the repository) and for permanent disposal (i.e., the period of time after permanent closure or sealing of the repository). The two phases are often referred to as the preclosure and postclosure phases. The preclosure or operational phase of the repository is limited by an annual individual dose limit of 0.15 mSv/yr [15 mrem/yr] for members of the public from normal operations at the repository.

The EPA standards specify three separate standards for the disposal or postclosure phase that address individual protection, human intrusion, and groundwater protection. The individual protection standard specifies that a reasonably maximally exposed individual shall receive no more than 0.15 mSv/yr [15 mrem/yr] from all exposure pathways (e.g., internal radiation exposures from ingestion of contaminated water, crops and animal products; external exposures from contamination on the ground). Consistent with the U.S. National Academy of Sciences recommendation that the standards define the characteristics of the exposure scenario, the EPA standards specify characteristics of the reasonably maximally exposed individual for estimating doses from potential releases from the repository. The standard specifies that the reasonably maximally exposed individual lives approximately 18 km [11 mi] from the repository in the predominant direction of groundwater flow and withdraws water from the aquifer that contains the highest concentration of contamination; has a diet and living style representative of the people who now live in the Town of Amargosa Valley, Nevada; and drinks 2 L [.53 gal] of water daily. The radiation standard for human intrusion is also a dose limit of 0.15 mSv/yr [15 mrem/yr] for the reasonably maximally exposed individual, however, calculation of the consequences of human intrusion is constrained by specific assumptions. The circumstances of human intrusion assumes that exploratory drilling for groundwater results in the intruders drilling directly through a waste package to the water table directly below the repository. DOE is to determine the earliest time that an intrusion would occur, using current technology for drilling water wells, without recognition by the drillers that a waste package was penetrated. Finally, EPA specified separate standards for the protection of groundwater. The groundwater standards set concentration limits for certain Radionuclides {i.e., 0.185 Bq/l [5 pCi/l] for radium-226 and 228, and 0.556 Bq/l [15 pCi/l]} for the combined alpha emitting radionuclides excluding radon and uranium) and a dose limit for other radionuclides {i.e., 0.04 mSv/yr [4 mrem/yr]} to the whole body or any individual organ for beta and photon emitters). These postclosure standards apply over a 10,000-year compliance period. EPA considered both policy and technical reasons in selecting this compliance period.

Performance Assessments: The performance assessment is a systematic analysis that identifies the features, events, and processes (i.e., specific conditions or attributes of the geologic setting; degradation, deterioration, or alteration processes of engineered barriers; and interactions between the natural and engineered barriers) that might affect performance of the geologic repository; examines their effects on performance; and estimates the potential radiological consequences. DOE is required to show compliance with the postclosure performance objectives with a performance assessment. To ensure DOE uses meaningful and reasonable calculations, EPA specified certain limitations for the performance assessment to preclude boundless speculation. The DOE performance assessments are not to include consideration of "very unlikely" features, events, and processes, which EPA defines to be those features, events, and processes that have less than one chance in 10,000 of occurring within 10,000 years of disposal. In addition, the EPA standards direct NRC to exclude unlikely features, events, and processes, or sequences of events and processes, from the required assessments for demonstrating compliance with the human intrusion and groundwater protection standards. EPA did not define unlikely features, events, and processes in its standards, but, rather, left the specific probability of the unlikely features, events, and processes for NRC to define. The EPA standards also specify criteria that pertain to the characteristics of a reference biosphere. The standards specify that the reference biosphere used in the performance assessments needs to be consistent with present conditions in the Yucca Mountain area and speculation on changes in society, human biology, or increases or decreases in human knowledge or technology should not be considered.

NRC Regulations

On February 22, 1999, NRC proposed licensing criteria in a new, separate part of its regulations, at 10 CFR Part 63, for disposal of high-level waste in a potential geologic repository at Yucca Mountain, Nevada. After publication of the proposed 10 CFR Part 63, the NRC staff provided members of the public and other stakeholders multiple opportunities to discuss the proposed requirements. NRC published its final regulations for disposal of high-level wastes in a potential geologic repository at Yucca Mountain, Nevada, on November 2, 2001. The regulations address the performance of the repository system in addition to addressing the licensing procedures, records and reporting, monitoring and testing programs, performance confirmation, quality assurance, personnel training and certification, and emergency planning. The primary focus of the regulations is public health and safety. In particular, the regulations provide for safety evaluations, safety plans and procedures, and continued oversight of safety.

Safety Evaluations: Safety evaluations are required for compliance with both the preclosure and postclosure performance objectives. The NRC regulations contain specific requirements for the preclosure and postclosure safety analyses to ensure they consider an appropriate range of issues in sufficient detail to allow NRC to determine whether or not DOE has demonstrated compliance with the performance objectives.

The preclosure safety analysis is a systematic examination of the site; the design; and the potential hazards, initiating events, and their resulting event sequences and potential radiological exposures to workers and the public. The regulations require DOE to identify the event sequences that might lead to radiological exposures. An event sequence means a series of actions or occurrences within the natural and engineered components of a geologic repository operations area that could potentially lead to exposure of individuals to radiation. An event sequence includes one or more initiating events and associated combinations of repository system component failures, including those produced by the action or inaction of operating personnel. The regulations classify the event sequences by two broad categories called Category 1 and Category 2. Those event sequences that are expected to occur one or more times before permanent closure of the geologic repository operations area are referred to as Category 1 event sequences. Consistent with the EPA final standards, Category 1 events sequences are limited to an annual individual dose of 0.15 mSv/year [15 mrem/yr] for members of the public from normal operations at the repository. Other event sequences that have at least one chance in 10,000 of occurring before permanent closure are referred to as Category 2 event sequences. The repository operations area is to be designed such that any Category 2 event sequence (i.e., those event sequences representing off-normal or accident conditions) will not result in an individual dose larger than 0.05 Sv [5 rem]. The analysis of a specific Category 2 design basis event would include an initiating event (e.g., an earthquake) and the associated combinations of repository system or component failures that can potentially lead to exposure of individuals to radiation. An example design basis event is a postulated earthquake (the initiating event) which results in (i) the failure of a crane lifting a spent fuel waste package inside a waste handling building, (ii) damage to the building ventilation (filtration) system, (iii) the drop and breach of the waste package, (iv) damage to the spent nuclear fuel, (v) partitioning of a fraction of the radionuclide inventory to the building atmosphere, (vi) release of some radioactive material through the damaged ventilation (filtration) system, and (vii) exposure of an individual (either a worker or a member of the public) to the released radioactive material.

A primary focus of the preclosure safety analysis is the identification of the structures, systems, and components relied on to limit or prevent potential event sequences or mitigate their consequences (i.e., important to safety). To ensure that DOE performs a comprehensive evaluation of safety for both workers and the public, the NRC regulations require that DOE address specific topics in its safety assessment. Among these are: means to limit concentration of radioactive material in air; means to limit the time needed to perform work near radioactive materials; means to control access to high radiation areas or airborne radioactivity areas; means to prevent and control criticality; radiation alarms that warn of significant increases of radiation levels, concentrations of radioactive material in air, and increased radioactivity in effluents; the ability of structures, systems, and components to perform their intended safety functions, assuming the event sequences occur; explosion and fire detection and suppression systems; means to provide reliable and timely emergency power to instruments, utility service systems, and operating systems important to safety if there is a loss of primary electric power; and means to inspect, test, and maintain structures, systems, and components important to safety to ensure their continued functioning and readiness.

Introduction

The EPA final standards require that DOE show compliance with the postclosure performance objectives using a performance assessment subject to certain constraints (see previous discussion under EPA standards). Evaluation of repository performance is complicated by uncertainties because of the first-of-a-kind nature of the repository and the very long time period for the analysis (i.e., 10,000 years). NRC is confident that a scientifically credible performance assessment is the best basis on which NRC can make an informed, reasonable licensing decision. To ensure that DOE develops a sufficiently credible evaluation of postclosure performance, the NRC regulations require that (i) uncertainties inherent in any performance assessment are thoroughly explained and analyzed or addressed, (ii) the DOE performance assessment is tested (corroborated) to the extent practicable, and (iii) there are added bases that provide confidence that the postclosure performance objectives will be met (i.e., multiple barriers). For example

- DOE is required to consider uncertainty in its representation of the repository (uncertainty and variability in parameter values must be taken into account) and the events that can happen during the compliance period (consideration of potentially disruptive events with a probability of occurrence as low as one chance in 10,000 of occurring over 10,000 years). Also, DOE must provide further assurances that uncertainty in the information (e.g., evaluation of site characterization data) used to develop the performance assessment has been evaluated by consideration of alternative conceptual models of features and processes that is consistent with available data and current scientific understanding. DOE must also supply its basis for including or excluding features, events, and processes that significantly affect performance.

- DOE is required to provide the technical basis for the models used in the performance assessment. Approaches for providing the technical basis would include comparisons of these models with information relevant to the conditions of geologic disposal and time periods of the assessment (e.g., results from detailed process-level models, field investigations, and natural analogs).

- The geologic repository must include multiple barriers, consisting of both natural barriers and an engineered barrier system. The performance assessment makes use of models and parameters that represent the behavior of the natural features of the repository system (e.g., characteristics of the hydrology, geology, and chemistry of the natural setting of the repository) as well as its engineered components. Specific features that have a capability to significantly affect the amount of water that contacts waste or the movement radionuclides in the geosphere (e.g., waste package, radionuclide sorption capacity of specific hydrogeologic units) are important to isolation of waste and are termed barriers. An important focus for the performance assessment is the identification of barriers relied on to isolate radioactive waste and characterization of each barrier capabilities. Confidence that the postclosure performance objectives will be met is not solely a matter of quantitative comparison with the performance objectives. A requirement that multiple barriers make up the repository system ensures that repository performance is not wholly dependent on a single barrier. As a result, the system is more tolerant of failures and external challenges such as disruptive events.

<u>Safety Plans and Procedures</u>: Safety evaluations identify the types of situations or scenarios that might result in radiological exposures, however, requirements for safety plans and procedures are used to minimize the potential for radiological releases and to be prepared in the event of radiological releases occur. To minimize the potential for radiological releases, the regulations specify that DOE must provide programs for training of personnel, quality assurance, and performance confirmation.

The Quality Assurance program comprises all those planned and systematic actions necessary to provide adequate confidence that the geologic repository and its structures, systems, or components will perform satisfactorily in service. The Quality Assurance program is applied to all structures, systems, and components important to safety (preclosure safety) and to design and characterization of barriers important to waste isolation (postclosure safety). Thus quality assurance requirements apply to a variety of activities such as facility and equipment design and construction, facility operation and maintenance, inspecting, testing, analyses of samples and data, tests and experiments, and scientific studies.

Confidence in the safety of the repository can be increased further by a program of continued investigation of repository performance (i.e., performance confirmation program). The regulations provide for a performance confirmation program to confirm the assumptions, data, and analyses that led to the findings that permitted construction of the repository and subsequent emplacement of the wastes. The general requirements for the performance confirmation program state that the program must provide data that indicate whether (i) subsurface conditions encountered and changes in those conditions during construction and waste emplacement are within limits assumed in the licensing review; and (ii) natural and engineered systems and components required for repository operation, and that are designed or assumed to operate as barriers after permanent closure, are functioning as intended and anticipated. Thus, key geotechnical and design parameters, including any interactions between natural and engineered systems and components, will be monitored throughout site characterization, construction, emplacement, and operation to identify any significant changes in the conditions assumed in the license application that may affect compliance with the performance objectives. Given the significant amount of time (e.g., tens of years) anticipated for construction and waste emplacement operations, it is likely that significant technical uncertainties will be resolved by performance confirmation, thereby providing greater assurance that the performance objectives will be met.

The regulations also contain certain requirements for DOE to be prepared for unexpected conditions. Specifically, DOE is required to have plans to cope with radiological accidents (i.e., emergency planning) and for retrieval of waste. Emergency planning is intended to ensure that DOE is prepared to respond, both on site and off site, to accidents. The required Emergency Plan includes identification of each type of accident, description of the means of mitigating the consequences of each type of accident; prompt notification of offsite response organizations; and adequate methods, systems, and equipment for assessing and monitoring actual or potential consequences of a radiological emergency condition. Additionally, DOE is required to design and plan the repository for a potential retrieval of the radioactive waste. Waste retrieval is intended to be an unusual event only to be undertaken to protect public health and safety. For example, if information became available during the performance confirmation program that indicated that public health and safety would not be protected, the radioactive waste could be retrieved from the repository.

Continued Safety Oversight: The regulations provide for continued oversight of the safety of the repository through requirements to help preserve knowledge of the repository for future generations. The regulations specify that DOE employ both active and passive means to regulate and prevent activities that could impair the long-term isolation of radioactive waste. These measures could include construction of permanent markers to identify the site and repository; placement of records in the archives and land record systems of local, state, and Federal Government agencies to identify the location of the repository, boundaries of the site, and the nature and hazard of the waste; and a program for continued oversight to prevent any activity at the site that poses a risk of breaching the engineered barriers of the repository. Finally, the regulations require DOE to develop a program to provide long-term monitoring of the repository (i.e., after the repository has been closed).

Identification of the NRC Policy Issues

As previously mentioned, the purpose of issue resolution is to assure that sufficient information is available on an issue to enable NRC to conduct a review of a proposed license application. The NRC and DOE interactions on the key technical issues and the issue resolution process are in conformance with the NRC efforts to streamline its high-level waste program and prepare for an efficient and competent review of any license application DOE may submit. As part of the issue resolution process, the NRC staff attempt to identify, and raise to management attention, any policy issues that may need the NRC Commission guidance. These issues could include issues that may require NRC rule changes, Commission direction, or Commission interpretations of existing policies.

Since August 2000, NRC and DOE have held technical exchanges on all the key technical issues and preclosure safety. These technical exchanges focused on issue resolution. Agreements were reached between DOE and NRC on additional information needed from DOE in a possible license application. No specific NRC policy issues were identified as a result of these technical exchanges. As the issue resolution process moves forward, the NRC staff will communicate NRC policy issues to the Commission, if any are identified.

1.4 Risk-Informing NRC Reviews

The reviews documented in this report were conducted to determine the resolution status of technical issues during the prelicensing period. Therefore, these reviews were not to decide whether a license should be granted. Although the purposes of the prelicensing issue resolution reviews and the licensing reviews are different, they share a basic underlying philosophy. This basic review philosophy can be found in the NRC strategic plan (2000) in the discussion of licensee responsibility, which states

> LICENSEE RESPONSIBILITY embodies the principle that, although the U.S. Nuclear Regulatory Commission is responsible for developing and enforcing the standards governing the use of nuclear installations and materials, *it is the*

> *licensee who bears the primary responsibility for conducting those activities safely. The U.S. Nuclear Regulatory Commission's role is not to monitor all licensee activities but to oversee and audit them* [emphasis added]. This allows the agency to focus its inspection, licensing, and other activities on those areas where the need, and the likely safety and safeguards benefit, is [sic] greatest.

Consequently, the licensee is held fully responsible for the safe operation of a nuclear facility while the NRC actions (including reviews) are focused on those areas where the need and the likely safety benefit are the greatest. More formally, the risk-informed approach is defined in an NRC white paper (NRC, 1999b) as one in which risk insights are considered together with other factors that better focus licensee and regulatory attention on issues commensurate with their importance to public health and safety. The risk insights are gained from risk assessments, engineering analyses, operating experience, and evaluations of performance histories. An appropriately applied risk-informed approach can reduce unnecessary conservatism, lead to better decision making, and support economical use of resources. A risk-informed approach lies between a risk-based approach and a deterministic approach.

A risk-informed approach focuses the NRC prelicensing reviews on topics that, among other factors, are major potential contributors to safety or alternatively that are likely to contribute most to risk reduction. These topics are selected based on information presented by DOE, independent staff investigations, published information, and experience gained through attending meetings of review committees and participating in site visits. To a large extent, staff rely on information provided by DOE to risk-inform its review. Through its repository safety strategy (CRWMS M&O, 2000c), DOE proposes the main system components on which it will rely for demonstrating the safety of any repository it may propose. In its preclosure integrated safety analyses and postclosure performance assessments, DOE demonstrates the implementation of the repository safety strategy. Combined with NRC staff independent analyses, these DOE analyses provide a reasonable framework for selecting items of high importance to system safety and, therefore, that should be subjected to a more thorough NRC review. This approach of risk-informing reviews directly helps to meet two NRC strategic goals: enhance effectiveness, efficiency, and realism; and reduce unnecessary regulatory burden. The approach indirectly contributes to the other two goals: enhance safety, environment, defense, and security; and increase public confidence.

The following three principles are important in implementing the NRC regulatory mission:

- NRC does not select sites nor does it design systems, structures, and components. The Nuclear Waste Policy Act of 1982 (1982), however, requires prelicensing consultation between DOE and NRC.

- The NRC role is not to monitor all DOE repository activities but to oversee and audit them. As a part of prelicensing consultation, NRC will evaluate information provided by DOE to determine if such information is sufficient to make regulatory decisions if it is later included in a license application. Reviews of items involving new methods and assumptions may use independent calculations and limited gathering of data for verification purposes. Otherwise, the NRC staff will review the information to ensure that assumptions are justified, methods used are acceptable and applicable over the range presented, models are properly applied, and results are acceptable. Staff will conduct appropriate bounding calculations, performance assessments, and confirmatory analyses using process-level models; however, in-depth, detailed analyses can be limited to a very few applications.

- After a license application is submitted and reviewed, NRC has three options: (i) grant the license, (ii) grant the license subject to conditions, or (iii) deny the license. Other than rejecting an applicant or licensee proposal, NRC has no power to compel a licensee to come forward or to require a licensee to prepare a different proposal. The burden of proof is on the applicant to show that the proposed action is safe, to demonstrate that regulations are met, and to ensure continued compliance with the regulations.

1.5 Preclosure and Postclosure Review Processes

A geologic repository system would use both engineered and natural features to meet the preclosure and postclosure performance objectives. Mathematical modeling and computer simulations are expected to be an important part of any DOE demonstration of repository safety. Other lines of evidence (e.g., natural analogs for postclosure and empirical observations of other nuclear and nonnuclear facilities for preclosure) are also expected to be a part of the DOE safety case. Identification of issues, review of technical information, status, and suggestions on the path forward for resolving specific technical issues are presented in Chapters 2 and 3 for preclosure and postclosure topics, respectively. In this section, five generic acceptance criteria that apply to all aspects of repository safety are discussed. These generic criteria are later formulated as review methods, which are then customized for application to each review based on risk information. The questions associated with each of the following five generic criteria are those for which a review seeks answers.

(1) <u>System Description and Model Integration</u>

- Have consistent and appropriate assumptions and initial and boundary conditions been propagated throughout the DOE models and calculations?

- Are the conditions and assumptions used to generate any look-up tables or regression equations consistent with other conditions and assumptions in the preclosure and postclosure safety analyses?

- Have important design features that will set the initial and boundary conditions for models and calculations been included?

- Has DOE considered the space-time dimensionality appropriately?

- Have important physical phenomena and couplings been included in the preclosure and postclosure safety analyses?

- Has sufficient justification been provided for any excluded coupling?

(2) Data Are Sufficient for Model Justification

- Has DOE demonstrated that sufficient data exist to support the conceptual models and define relevant parameters in the DOE models and calculations?

- Is the primary source of data (field, laboratory, or natural analog) appropriately qualified from a quality assurance perspective?

- Are conceptual models and parameter values, where data are inadequate, based on other appropriate sources, such as expert elicitation conducted in accordance with NUREG–1563 (NRC, 1996)?

- Has DOE performed sensitivity and uncertainty analyses to test the need for additional data?

- Has DOE provided sound bases for the inclusion or exclusion of observed phenomena in its conceptual models?

(3) Data Uncertainty Is Characterized and Propagated through the Model Abstraction

- Are the parameter values used in the models and other calculations reasonable based on data from the Yucca Mountain region and other applicable laboratory tests, design documents, natural analogs, and applicable industry standards?

- Do parameter values, their assumed ranges, and their probability distributions (if used), reasonably account for uncertainty and variability?

- Are any bounding assumptions technically defensible?

- Are the data consistent with the design features and the assumptions of the conceptual models?

- Have any correlations between parameter values been appropriately considered?

- How do the DOE parameter values compare to those in published literature or those obtained independently by the staff?

- What is the sensitivity of the system safety measures (preclosure and postclosure) to the parameters?

(4) Model Uncertainty Is Characterized and Propagated through the Model Abstraction

- Has DOE considered plausible alternative models?

- Has DOE provided supporting information for the conceptual model(s) used in the safety case?

- Are the intermediate outputs of the engineered and natural system models produced by DOE consistent with the selected conceptual model(s)?

(5) Model Abstraction Output Is Supported by Objective Comparisons

- Has DOE demonstrated that there is a reasonable physical basis to explain the output of the models or results of other calculations t used to draw safety-related conclusions?

- Has DOE assembled other sufficient evidence to support model results?

Detailed acceptance criteria for each generic topic is presented in NRC (2002).

1.6 References

CRWMS M&O. "Performance Confirmation Plan." TDR–PCS–SE–000001. Revision 01 ICN 01. Las Vegas, Nevada: TRW Environmental Safety Systems, Inc. 2000a.

———. "Total System Performance Assessment for the Site Recommendation." TDR–WIS–PA–000001. Revision 00 ICN 01. Las Vegas, Nevada: TRW Environmental Safety Systems, Inc. 2000b.

———. "Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Consideration." TDR–WIS–RLL–000001. Revision 04 ICN 01. Las Vegas, Nevada: TRW Environmental Safety Systems, Inc. 2000c.

Energy Policy Act of 1992. Pub. L. 102–86. 106 Stat. 2776 (1992).

National Research Council. "Technical Bases For Yucca Mountain Standards." Washington, DC: National Academy Press. 1995.

Natural Resources Defense Council, Inc. v. United States EPA, 824 F.2d 1258 (1st Cir. 1987).

NRC. NUREG–1563, "Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program." Washington, DC: NRC. November 1996.

———. "Guidance Document for Streamlining the High-Level Waste Program." Washington, DC: NRC. 1999a.

———. "White Paper on Risk-Informed and Performance-Based Regulation, Announcement No. 19, March 11, 1999." Washington, DC: NRC. 1999b.

———. NUREG–1614, "U.S. Nuclear Regulatory Commission Strategic Plan: Fiscal Year 2000–Fiscal Year 2005." Washington, DC: NRC. September 2000.

———. NUREG–1804, "Yucca Mountain Review Plan—Draft Report for Comment." Revision 2. Washington, DC: NRC. March 2002.

Nuclear Waste Policy Act of 1982. Pub. L. 97–425. 96 Stat. 2201 (1982).

Sagar, B., ed. NUREG/CR–6513, No. 1, CNWRA 96–OIA, “NRC High-Level Radioactive Waste Program Annual Progress Report: Fiscal Year 1996.” Washington, DC: NRC. January 1997.

2 REPOSITORY SAFETY BEFORE PERMANENT CLOSURE

2.1 Preclosure Safety Analysis

2.1.1 Site Description As It Pertains to Preclosure Safety Analysis

2.1.1.1 Description of Issue

This section of the Integrated Issue Resolution Status Report addresses assessment of the Yucca Mountain site description as it pertains to DOE preclosure safety analysis. Site description comprises (i) site geography, (ii) regional demography, (iii) local meteorology and regional climatology, (iv) regional and local surface and groundwater hydrology, (v) site geology and seismology, (vi) igneous activity, (vii) site geomorphology, and (viii) site geochemistry. Assessment of the DOE preclosure site description is for compliance with the performance objectives in 10 CFR Part 63, which requires a preclosure safety analysis of the Geologic Repository Operations Area for the period before permanent closure. Adequacy of the site description is assessed based on information necessary for DOE to conduct its preclosure safety analysis and Geologic Repository Operations Area design. Section 1.3, Regulations Applicable to High-Level Waste Repository at Yucca Mountain, of the Integrated Issue Resolution Status Report discusses the methodology used by staff for this review.

The DOE site description is primarily documented in CRWMS M&O (2000a) and in DOE (1999a). These reports, plus additional supporting DOE documents identified in the appropriate subsections that follow, are reviewed to the extent that they contain site description information relevant to the preclosure safety analysis. Much site description information also pertains to repository safety after permanent closure and, where appropriate, this review cross-references appropriate sections of the postclosure review contained within this Integrated Issue Resolution Status Report. In addition, this preclosure review incorporates information previously evaluated within the key technical issue framework, including Key Technical Issues: (i) Igneous Activity, (ii) Structural Deformation and Seismicity, (iii) Evolution of the Near-Field Environment, (iv) Thermal Effects on Flow, (v) Repository Design and Thermal Mechanical Effects, (vi) Unsaturated and Saturated Flow Under Isothermal Conditions, and (vii) Total System Performance Assessment and Integration.

2.1.1.2 Importance to Safety

Yucca Mountain is located in Nye County, Nevada, within the Western Great Basin of the Central Basin and Range physiographic province of the North American Cordillera. Topography of the Yucca Mountain region reflects the extensional tectonics that controlled the region's geologic history throughout the past 65 million years. Regional topography is characterized by exhumed blocks of basement crust that form subparallel north-south striking ranges separating elongated and internally drained basins. The ranges are up to several hundred kilometers long with elevations up to 2 km [1 mi] above the basin floors. Much of the surface faulting took place at the base of the ranges along normal faults that dip moderately (~60°) beneath the adjacent basins (generally defined as range-front faults); although complex faulting within the basins is also common. The region remains seismically and volcanically active. Climate is arid to semiarid, and natural water flow is generally restricted to groundwater several hundred meters (500+ ft) below the surface with occasional surface runoff in washes and across alluvial fan drainages after rainstorms. Groundwater flows in several regional and local aquifers contained within alluvial valley fill sedimentary strata, volcanic rocks, and underlying carbonate strata. The repository is to be housed in the silicic volcanic rocks, mainly tufaceous strata erupted from calderas to the north and northwest of Yucca Mountain between 10 and 15 million years ago.

The Yucca Mountain site rests primarily within the westernmost parts of the Nevada Test Site. Parts of the proposed repository are also within the Beatty District of the public lands administered by the Bureau of Land Management and U.S. Air Force (Nellis Air Force Range). The nearest population centers are Beatty, Nevada {28 km [17 mi] to the west-northwest}, Amargosa, Nevada {24 km [15 mi] to the south}; Pahrump, Nevada {83 km [52 mi] to the south-southeast}, and Las Vegas, Nevada {142 km [88 mi] to the east-southeast}. The U.S. Congress selected Yucca Mountain for characterization in 1983, in part, because of its thick unsaturated zone, its arid to semiarid climate, and the existence of a rock type that would support excavation of stable openings.

Directed by the present regulatory framework of risk-informed performance-based standards (e.g., 10 CFR Part 63), review of the DOE preclosure safety analysis is restricted to information necessary to demonstrate the repository will be designed, constructed, and operated to meet the specified exposure limits (performance objectives) through the preclosure period. Site characterization, especially of the natural systems, is necessary to evaluate the ability of the site to perform within the performance objectives. The natural systems provide the framework within which the engineered systems will be expected to operate and perform.

2.1.1.3 Technical Basis

Review of the site description is organized according to the eight review methods and associated acceptance criteria identified in the Yucca Mountain Review Plan (NRC, 2002). These eight review methods and acceptance criteria are organized around eight general subsections of the site description, which are

- Site Geography
- Regional Demography
- Local Meteorology and Regional Climatology
- Regional and Local Surface and Groundwater Hydrology
- Site Geology and Seismology
- Igneous Activity
- Site Geomorphology
- Site Geochemistry

2.1.1.3.1 Site Geography

The following sections on site geography refer to the requirements of 10 CFR 63.112(c). The potential DOE license application should contain a description of the site geography adequate to permit evaluation of the preclosure safety analysis and the Geologic Repository Operations Area design.

Site Location

Yucca Mountain is located in Nye County, Nevada, approximately 142 km [88 mi] west-northwest of Las Vegas. The proposed repository site would be on land controlled by the U.S. Air Force (Nellis Air Force Range), the DOE Nevada Test Site, and the U.S. Bureau of Land Management.

The geographic location of the proposed high-level waste repository at Yucca Mountain, Nevada, is adequately identified in CRWMS M&O (2000a). However, the location of the proposed preclosure and postclosure controlled areas, as defined in CRWMS M&O (2000a), may need to be redrawn to conform with the EPA Standard for Yucca Mountain.

Significant Natural and Manmade Features

DOE describes natural features at the Yucca Mountain site in CRWMS M&O (2000a). Significant manmade features are identified and located in Tables 2.2-1 and 2.2-2 and in Figures 2.2-7 and 2.2-8 in CRWMS M&O (2000a). Table 2.2-1 and Figure 2.2-7 adequately identify and locate facilities and infrastructure outside, but near the preclosure controlled area. Table 2.2-2 and Figure 2.2-8 identify both existing and potential surface facilities in the preclosure controlled area at Yucca Mountain. Figures 2.2-9 (north portal) and 2.2-10 (south portal) in CRWMS M&O (2000a) show the facilities and infrastructure in greater detail. These figures also identify potential facilities and infrastructure within the radiologically controlled area.

The locations of 13 of the features listed in Table 2.2-2, however, have not been determined because DOE has not yet finalized all aspects of the site design:

- Security Station 2
- Utility Building
- General Parking Areas
- Transformer Yard
- Optional Tuff Crushing and Screening Plant
- Aggregate Storage Area
- Water Storage Tank
- Discharge Storage Pond
- Dispatcher House
- Diesel Fuel Storage Tank with Sump
- Truck Unloading Area
- Surface Rail Parking Area
- Security Station, Main Gate

Although locations of some of these facilities may not be critical to preclosure safety, others, such as the aggregate storage area, water storage tanks, and diesel fuel storage tanks, could impact preclosure site safety. During future meetings on preclosure safety, DOE needs to identify the locations of all manmade and natural features important to preclosure safety and document them in a potential license application.

Repository Safety Before Permanent Closure

Site Maps

CRWMS M&O (2000a) contains maps that adequately locate (i) Yucca Mountain (Figures 1.1-1, 2.2-1, 2.2-2, 2.2-3), (ii) physiography (Figures 1.2-1 and 2.2-4), (iii) facilities and infrastructure (Figures 1.3-1, 1-3.2, 2.2-7, 2.2-8, 2.2-9, and 2.2-10), (iv) preclosure controlled area (Figure 2.2-5), and (v) potential withdrawal area (Figure 2.2-6). The maps and information conveyed are adequate to identify these features with regard to preclosure safety assessment in a potential license application.

2.1.1.3.2 Regional Demography

The following sections on regional demography refer to the requirements of 10 CFR 63.112(c). The potential DOE license application should contain a description of the regional demography adequate to permit evaluation of the preclosure safety analysis and the Geologic Repository Operations Area design.

The regional demography is reviewed in CRWMS M&O (2000a) and DOE (1999a). In CRWMS M&O (2000a), population estimates are based principally on the Nevada State Demographer's reports (Nevada State Demographer, 1999a,b,c), and on estimates made by CRWMS M&O (1998a) and by the U.S. Census Bureau (1993, 1996). These data are for the estimated population in 1998. The regional demographics are inadequate as they are based on outdated population estimates. DOE estimates should take into account the most recent census data compiled in the 2000 census.

2.1.1.3.3 Local Meteorology and Regional Climatology

The following sections on local meteorology and regional climatology refer to the requirements of 10 CFR 63.112(c). The potential DOE license application should contain a description of the local meteorology and regional climatology adequate to permit evaluation of the preclosure safety analysis and the Geologic Repository Operations Area design.

Climate and Meteorological Conditions

The modern climatic and meteorological conditions at Yucca Mountain are influenced by a broad range of atmospheric mechanisms including global-scale processes, regional weather patterns, seasonal variations, and local topographically controlled weather patterns (CRWMS M&O, 2000a). Central and southern Nevada's current climate is generally arid to semiarid because of modern regional weather patterns, far-away moisture sources such as the Pacific Ocean (including the Gulf of California) or the Gulf of Mexico, and the numerous mountain ranges between Yucca Mountain and these moisture sources. The degree of aridity varies in space, mostly by elevation, and in time, seasonally and annually. Typical rainfall is less than 254 mm/yr [10 in/yr]. Temperatures are warm in the summer {often near 40 °C [104 °F]} and cool to cold in winter {as cold as 0 °C [32 °F]} (CRWMS M&O, 2000a).

Present-day climate and meteorological conditions are discussed in CRWMS M&O (2000a). Discussions on the local meteorology are based on data acquired by the onsite meteorological monitoring network operated by the Yucca Mountain Radiological and Environmental Programs

Department and selected regional National Oceanic and Atmospheric Administration meteorological stations (CRWMS M&O, 2000a). Information on the large-scale climatic factors affecting the Yucca Mountain area was obtained from textbooks and scientific literature as described in the CRWMS M&O (2000a).

Staff have not fully reviewed all aspects of the DOE summary of local meteorological and regional climatological conditions as they relate to preclosure safety. Future revisions of the Integrated Issue Resolution Status Report will provide staff assessment of these aspects of the Yucca Mountain site description.

<u>Precipitation and Flooding</u>

Precipitation is characterized in Section 6.2.3.1 of CRWMS M&O (2000a). Tables 6.2-3 and 6.2-4 summarize the precipitation statistics for five stations at and near Yucca Mountain; Tables 6.2-10 to 6.2-18 provide monthly and annual climatological summaries, including precipitation, for the local weather stations one to nine, within the Radiological and Environmental Programs Department Sites; Table 6.2-20 provides monthly climatology summaries for regional weather stations; Table 8.2-4 summarizes the annual precipitation for the National Weather Service Stations between 1921 and 1947; and Table 6.2-25 summarizes the annual precipitation for the National Weather Service Stations between 1948 and 1995. Average precipitation for Yucca Mountain ranges between 174 and 195 mm/yr [7 and 8 in/yr] compared with the 254 mm/yr [10 in/yr] average for the region with only 102–107 mm/yr [4 in/yr] in the Amargosa farms area. Average precipitation values are based on 30-year records.

Flooding is discussed in Section 7.3 of CRWMS M&O (2000a). This section summarizes local and regional flood studies in southern Nevada, as well as local studies in the Yucca Mountain region. Results of hydrologic engineering studies started in 1999 have not yet been reported by DOE or its contractors.

Staff have not fully reviewed all aspects of the DOE summary of precipitation and flooding as they relate to preclosure safety. Future revisions of the Integrated Issue Resolution Status Report will provide staff assessment of this aspect of the Yucca Mountain site description. Staff note, however, that summaries of data from nearby regional meteorological stations, including the Amargosa Farms, Jackass Flat, and Area 12 Mesa, are not included, despite their relatively long rainfall records. The relative close proximity of Site 9 (Radiological and Environmental Programs Department Site), Jackass Flat, and Amargosa Farms meteorological stations would provide additional support for meteorological data and models.

<u>Severe Weather</u>

Severe weather events include extreme precipitation event from storms, high winds, and tornadoes. Severe weather conditions at Yucca Mountain are described in Section 6.2 of CRWMS M&O (2000a). Staff have not fully reviewed all aspects of the DOE summary of severe weather as they relate to preclosure safety. Future revisions of the Integrated Issue Resolution Status Report will provide staff assessment of this aspect of the Yucca Mountain site description.

2.1.1.3.4 Regional and Local Surface and Groundwater Hydrology

The following section on regional and local surface and groundwater hydrology refer to the requirements of 10 CFR 63.112(c). The potential DOE license application should contain a description of the local and regional hydrological information to support evaluation of the preclosure safety analysis and the Geologic Repository Operations Area design.

A review of the integration of surface and groundwater characteristics into the design, construction, and operation of the repository is a necessary component of the preclosure safety analysis. The primary concerns are inundation and erosion by water and debris flows of the surface facilities and components and elevated flux of water into subsurface tunnels during the operational phase of the repository. To ensure that hydrological features relevant to preclosure safety and repository operations area design are adequately identified, descriptions of the following items will be evaluated:

- Stream locations
- Natural drainage features
- Flood potential
- Perched water
- River or stream control structures
- Depth of aquifers beneath the site and their recharge and discharge features

This section reviews the characterization and analyses of surface and groundwater interaction with the repository design. The focus is proportionately on features deemed to be high-risk-significant structures, systems, and components important to safety. Accordingly, evaluation is needed for the (i) flood potential and drainage design for the facilities, systems, and components; (ii) transportation pathways crossing wash channels in the control area; and (iii) design modification and standoff distances from known and unexpected faults crossing emplacement drifts and access tunnels. These three items are discussed in the context of Surface Waters and Groundwater.

The primary area of surface facilities is the north pad, adjacent to the north portal of the Exploratory Studies Facility. Other areas include facilities on the south pad adjacent to the south portal of the Exploratory Studies Facility, a potential onsite storage area sited on the northern portion of Midway Valley (CRWMS M&O, 1998b), the ventilation shafts for the operational period and for postclosure, the muck area in Midway Valley, and the transportation routes used to deliver the waste to the north pad facilities. The design of the potential repository and associated facilities is partially completed, with few details on some components. Aspects of the design will likely change, though the rationale for any design constraints should not change.

Documents reviewed for repository and facility design are CRWMS M&O (1998b, 1999, 2000b). Documents reviewed for characterization of the natural systems are CRWMS M&O (2000a) and DOE (1995), and Bullard (1986). Bullard (1994) was not available at the time of this review. Documents reviewed for preclosure safety are CRWMS M&O (2000c) and DOE (2001).

Surface Waters

There are no perennial streams in the Yucca Mountain area. Ephemeral streams flow, however, and drainage areas have been adequately delineated (CRWMS M&O, 2000a). Flow in the wash channels occurs as a result of large-magnitude precipitation events, either as localized, intense, summer storms or as regional, long-duration storms. Localized summer storms generally can lead to flash floods in any of the washes on and near Yucca Mountain. Flooding in Fortymile Wash is generally caused by regional, long-duration winter precipitation events. Runoff during intense precipitation can both erode the hillslopes and inundate and erode the washes. Both water and rock debris flows are known to occur in the Yucca Mountain area.

Large-magnitude precipitation events can cause three problems for repository and operational design: (i) localized drainage of water and debris flows onto facilities; (ii) drainage off facility buildings and pads, including increased loads on roofs of critical building structures; and (iii) flooding and associated debris flows in and adjacent to main wash channels. Natural drainage features and engineered drainage within facilities are discussed first, followed by a discussion of flooding along wash channels.

Multiple ventilation and exhaust shafts are part of the current repository design (CRWMS M&O, 2000b). Separate ventilation systems will be operated, one for the emplacement operations and one for the excavation operations. The number and location of shafts are not fixed in the basecase design and may also vary in the design alternatives. The shafts appear to be vertical and will intersect the ground surface somewhere between the crest of Yucca Mountain and part way down the east flank. It is not clear what the ventilation shaft design calls for: the intersection with the ground surface to avoid channels in the upper washes of the east flank of Yucca Mountain or construction of engineered structures that will route runoff away from the shaft openings. Ventilation shafts are clearly not sited over emplacement drifts. Hence, the safety concern is with operation of the ventilation systems and flooding of localized zones in the tunnels. The exhaust main is below the elevation of the emplacement drifts and the ventilation cross drifts are between emplacement drifts.

The north pad lies near the bottom of Exile Hill. Runoff or debris flow from the east side of Exile Hill could move onto the north portal pad. The elevation difference between the top of Exile Hill and the north portal is about 35 m [115 ft] and for the northern part of the pad is 50 m [164 ft]. The horizontal distance is about 110 m [361 ft] to the portal and 175 m [574 ft] to facilities on the pad. This means there is only a small catchment area above the north portal facilities, based on the design described in CRWMS M&O (2000c). Analysis of probable maximum precipitation on the Exile Hill hillslope would dictate if any hillslope modifications or engineered systems would be needed. The facilities at the south portal pad are not sited in a flood-prone area but may be at similar risk for local hillslope water and debris flows as well as drainage off the pad.

In addition to runoff from Exile Hill, direct precipitation during intense storms could lead to flooding of facilities, buildings, and components. DOE (2001) mentions the design of roofs to withstand a 100-year precipitation event. NUREG–0800 (NRC, 1987) also includes review plans for site drainage and the effects of sedimentation and erosion. Because the drainage design for the north portal pad is tied to the flood mitigation from washes in Midway Valley (part of the pad being below the 100-year flood), drainage from the north portal pad is described in the next section.

Flooding and associated debris flows are common occurrences in washes of the Yucca Mountain area and generally in the arid southwest. Flood maps can be created for any precipitation recurrence interval. The flood maps can then be used to site facilities and components or to engineer the facilities and components to withstand a flood. For drainage off facilities, local topography and modified slopes and material characteristics would be considered in designing the routing components for water runoff.

Probable maximum flood is defined as the maximum runoff condition resulting from the most severe combination of hydrologic and meteorologic conditions considered reasonably possible for the drainage basin being studied. Probable maximum flood is derived using the probable maximum precipitation. A 100-year flood is the flood derived from a precipitation event having a recurrence interval of 100 years. By definition, there is no recurrence interval for a probable maximum precipitation or flood.

Bullard’s (1986) approach for estimating a probable maximum flood using a synthetic unit hydrograph developed with the probable maximum precipitation event is in agreement with the Army Corps of Engineers approach recommended in NUREG–0800 (NRC, 1987). Bullard (1986) used the maximum possible precipitation event determined from Hydrometeorologic Report 49 to generate the synthetic unit hydrograph. Hydrometeorologic Report 49 is obtained from the National Weather Service, National Oceanic and Atmospheric Administration. The approach for determining the water level associated with the probable maximum flood at the north portal pad, which is adjacent to the Midway Valley wash, also incorporates a bulking factor of two. The bulking factor is needed because Bullard’s (1986, 1994) approach is for clear water [i.e., the sediment (e.g., cobbles, boulders) volume carried in the water is not included in the estimate of (clear) water levels in the wash].

CRWMS M&O (2000b) and DOE (1995) refer to the results of Bullard (1994) and the addition of the bulking factor by Blanton (1992) in discussing probable maximum floods that might affect repository facilities. DOE (2001, p. 5-14), however, uses the 100-year flood for design considerations. It is not clear if peak water levels and flow rates of the probable maximum flood differ significantly from the 100-year flood. The choice of the 100-year flood leaves flooding as borderline between a Category 1 or 2 design consideration (CRWMS M&O, 2000c); however, Category 2 is selected (DOE, 2001). Documentation of ongoing engineering studies in the north portal area (CRWMS M&O, 2000b) may clarify the choice of the 100-year flood for design considerations and the category designation.

A portion of the north portal pad is within the area of the probable maximum flood. CRWMS M&O (2000c) and DOE (2001) note that critical buildings and systems will be designed above the probable maximum floodline, such as the Carrier Preparation Building, the Waste Handling Building, and the Waste Treatment Building. In addition, drainage from the radiological control area will include an underground storm drainage system designed to protect this portion of the pad from a probable maximum flood. The rest of the facility buildings on the pad near the north portal will be designed to withstand the 100-year flood. More details are needed to clarify the distinction between areas designed for the probable maximum flood and those designed for the 100-year flood.

A muck pile developed during excavation of the drifts is currently sited in Midway Valley (CRWMS M&O, 1998b, 1999). Sediments in Midway Valley aggregated during the modern climate conditions. There is little incision from ephemeral stream flow off the east flank of Yucca Mountain. A muck pile extending from approximately the south portal to the north portal might lead to a focusing of stream flow from Split, Coyote, Wren, and Drill Hole Washes. Coalescing stream flow into Midway Valley could incise and possibly erode facility systems.

Siting of a potential onsite storage area in the northern extent of Midway Valley (CRWMS M&O, 1998b, 1999) may be affected by flooding of any drainages leading into the northern portion of Midway Valley (e.g., Yucca Wash). It is not clear if the potential onsite storage area is still being considered.

Transportation pathways near the north portal area do not cross currently incising wash channels. Transportation pathways farther from the north portal were not described in the reviewed documents (CRWMS M&O, 1998b, 2000a). It appears, however, that radioactive waste being transported to the north portal will cross Fortymile Wash. Significant sediment movement and its associated erosive capabilities are known to occur after large-magnitude precipitation events (CRWMS M&O, 2000c). DOE did not discuss transportation pathways crossing Fortymile Wash in the documents reviewed for this report, and hence DOE has not discussed what measures will be taken to reduce risk associated with transportation structures crossing highly erosive environments. River or stream control structures may not be the preferred method of reducing risk at the Fortymile Wash crossing point because of the erosive nature of the intermittent water and debris flows.

Groundwater

Water influx into the drifts and access tunnels during operations could occur from perched water, a rising water table, or significant surface floods leading to flow down fault or fracture zones.

Evidence of upwelling water along faults remains a controversial issue. CRWMS M&O (2000b) describes an abundance of evidence purporting to refute the theory of upwelling of deep water to the repository horizon and the ground surface. Ongoing work estimating formation temperatures of fluid inclusions in secondary minerals along faults may resolve the issue.

Opposite of the upwelling fluids flow is the possibility of focused, fast pathway, downward percolation. The chemistry of the perched water body and of the aquifer beneath Yucca Mountain suggests the likelihood of recharge by fast pathway water flowing through faults and fractures. Portions of the repository access tunnels and emplacement drifts will intersect faults or underlie faults that cut the nonwelded Paintbrush tuffs. These areas may be prone to elevated water influx. Though standard mining practices would alleviate the problems, none have been noted in the reviewed repository design documents.

The depth of the aquifers and perched water beneath the site and the recharge and discharge features have been adequately described in CRWMS M&O (2000b). Evidence of past water table positions suggests maximum elevations in the repository footprint of 120 m [394 ft] above present day elevations (CRWMS M&O, 2000b). Perched water has been found at the base of the Topopah Springs Tuff and in the Calico Hills Formation below the repository footprint, but it is unlikely to occur in the repository horizons. Though there are aspects of these recharge and discharge features that remain highly uncertain, the lack of certainty for aspects not mentioned above does not warrant changes to the current design.

Summary

CRWMS M&O (2000a) and references therein adequately describe streams, drainages, and aquifers that might affect operation of the repository. Staff have not fully reviewed all aspects of the DOE summary of regional and local surface and groundwater hydrology with respect to preclosure safety. Future revisions of the Integrated Issue Resolution Status Report will provide staff assessment of these aspects of the Yucca Mountain site description. This preliminary assessment identified eight features that warrant further clarification:

- Potential water and debris flows from hill slopes above shafts and the north and south pads
- Siting criteria or engineered barriers for ventilation and emplacement shafts
- Routing of surface water from east flank washes around or through the muck pile
- Water level and peak discharge rate differences between the probable maximum flood and the 100-year flood
- Facility buildings and components that use 100-year flood design considerations rather than probable maximum flood
- Hydrologic issues for siting of a potential onsite storage area in northern Midway Valley
- Transportation route to north pad, particularly as it crosses incising channels such as Fortymile Wash
- Criteria for addressing water influx from faults that intersect drifts

2.1.1.3.5 Site Geology and Seismology

The following sections on site geology and seismology refer to the requirements of 10 CFR 63.112(c). The potential DOE license application should contain a description of the site geology and seismology to adequately permit evaluation of the preclosure safety analysis and the Geologic Repository Operations Area design.

Site Geology

Site geology includes the regional geologic and tectonic settings, Quaternary stratigraphy and surface processes, Yucca Mountain site stratigraphy and structural geology, geoengineering properties, integrated site models, and natural resources. Each of these areas is discussed with respect to the preclosure site description.

Regional Geologic Setting

As noted by DOE (CRWMS M&O, 2000a), Yucca Mountain lies within the Central Basin and Range physiographic province of the North American Cordillera. The region is characterized by complex interactions of strike-slip and extensional deformation, active since onset of the Cenozoic (65 million years). The region remains tectonically active as indicated by numerous Quaternary faults (including evidence for Holocene activity), historic seismicity (including the 1992 Little Skull Mountain earthquake activity), and volcanism (punctuated by the most recent volcanic eruption at Lathrop Wells Cone approximately 80,000 years ago).

Geologically, the Great Basin consists of north-south fault-bounded basins and mountain ranges (including Yucca Mountain) overprinted by extensive volcanic activity. Faults are mostly normal dip-slip or dextral strike-slip faults that reflect the extensional and transtensional deformation caused by interactions between the western margin of the North American continent with the Pacific plate during approximately the past 65 million years. In its description of geologic setting (CRWMS M&O, 2000a), DOE adopts a segmented regional framework in which the region is divided into three tectonic domains. Each tectonic domain is a structurally bounded section of the Earth’s crust with relatively similar deformational characteristics within the domain compared with markedly different deformational characteristics in adjacent domains. These domains are the Walker Lane domain, which includes the site; the Basin and Range domain, which includes the areas to the north and east; and the Inyo-Mono domain, which includes regions to the west and south.

The stratigraphy of the geologic setting consists of igneous, sedimentary, and volcanic rocks that range in age from Proterozoic (2500 million years) to the present. Pre-Cenozoic rocks (before 65 million years), which constitute the basement rocks of the regional geologic setting, primarily consist of Precambrian and Early Cambrian (approximately 2500 to 500 million years) siliciclastic strata overlain by a thick Paleozoic (approximately 500–245 million years) section of limestones and dolomite. The regional carbonate aquifer is within these Paleozoic strata. Cenozoic rocks of the Yucca Mountain geologic setting fall into three general groups: (i) pre-Middle Miocene (>16.5 million years) strata (including volcaniclastics) that predate the southwestern Nevada volcanic field, (ii) Middle to Late Miocene (16.6–5.3 million years) volcanic rocks that compose the southwestern Nevada volcanic field, and (iii) Plio-Pleistocene (5.3 million years to the present) basalts and basin sediments. The Cenozoic rocks overlie complexly deformed Paleozoic and Precambrian rocks on a regional erosional unconformity, suggesting significant uplift and erosion of the pre-Cenozoic rocks associated with extensional tectonics of the Basin and Range.

Structurally, the geologic setting is characterized by two distinct structural styles. Pre-Cenozoic (older than 65 million years) rocks are folded and faulted in contractile structures indicative of a series of compressional mountain buildings that affected much of western North America in the late Paleozoic and throughout the Mesozoic (approximately 245–65 million years). Cenozoic (65 million years to the present) deformation is extensional, producing normal and strike faults and related extensional features characteristic of the Basin and Range. The fault-bound edifice of Yucca Mountain, which includes a series of north-south, dip-slip faults and northwest-southeast strike-slip faults, is a product of the Cenozoic extension of the Basin and Range.

Historic earthquakes on many Basin and Range faults indicate that active extension is ongoing. Distribution of epicenters suggests that the most active areas of extension are within the eastern California shear zone, the Central Nevada Seismic Belt, and along the Wasatch Front in Utah. Geodetic measurements of plate motions also show active extension in these same regions (e.g., Bennett, et al. 1997; Savage, et al. 1995; Dixon, et al.,1995). The integrated strain rate across the eastern California shear zone is 12.1 ± 1.2 mm/yr [0.48 ± 0.05 in/yr], and most of that strain is apparently accommodated by slip on large faults such as the Death Valley–Furnace Creek and Owens Valley fault zones (Dixon, et al., 1995). Based on the relative motions of the Pacific and North American plates, this pattern of extension has been nearly constant during the past 3–4 million years (Harbert and Cox, 1989). The driving mechanism for ongoing extension is controversial, attributed to either a mantle plume associated with the Yellowstone hot spot (Saltus and Thompson, 1995), sinking of previously subducted oceanic lithosphere beneath the Basin and Range (Bohannon and Parsons, 1995), gravitationally derived buoyancy forces (Jones, et al., 1996; England and Jackson, 1989), or external plate tectonic forces from the motion of the Pacific and Sierra Nevada north and west relative to North America (Thatcher, et al., 1999).

The regional geologic setting for Yucca Mountain comprises tectonic, stratigraphic, and structural elements and furnishes context for more detailed understanding of the natural processes currently affecting Yucca Mountain and for evaluation of the site geology. CRWMS M&O (2000a) provides a comprehensive summary of the regional geologic setting. The summary gleans information from a variety of DOE, U.S. Geological Survey, and State of Nevada reports as well as from geologic literature published in professional journals. DOE findings with respect to site geology are consistent with the regional geologic setting as described in previous staff reviews (e.g., NRC, 1999a). Thus, the DOE regional geologic setting summary provides sufficient technical bases for the descriptive and process models used to assess the ability of the natural system to help meet preclosure safety performance objectives.

Since the 1999 staff review and summary of the site description (CRWMS M&O, 2000a), new aeromagnetic data were acquired (Blakely, et al., 2000). These new data may provide additional information on the regional geologic setting, especially geologic features such as faults and volcanoes now buried within the thick accumulations of alluvial material in the basins. DOE should evaluate the new aeromagnetic results and modify existing interpretations of the geologic setting as needed.

Regional Tectonic Setting

The tectonic setting of Yucca Mountain provides a framework for descriptive and process models of the Yucca Mountain site and region within the context of the geological evolution of the Basin and Range physiographic province. Tectonic models for Yucca Mountain region explain geologic and geophysical data within the established tectonic processes. To do so, discrete data sets such as the histories of volcanism, deposition, and fault movement are integrated to develop a reasonable interpretation of the geological evolution of the region, compatible with existing data and the principles of the earth sciences. In this way, tectonic models provide a regional context within which DOE scientists evaluated attributes of the Yucca Mountain region such as seismic sources, faulting probability, structural control of groundwater flow, magmatism, and geologic stability of the natural and engineered systems. Tectonic models of the Yucca Mountain region depict large crustal features such as long faults (e.g., Solitario Canyon fault), extensive fracture systems, volcanoes, blocks of rock as big as mountain ranges, basins such as Crater Flat, and additional evidence of strains caused by plate tectonics such as detachment faults and the progressive southerly vertical axis of rotation of fault blocks.

The geological community investigating Yucca Mountain has not accepted any single explanation of these features and processes. Initial staff review of the geologic literature (e.g., McKague, et al., 1996) suggested that tectonic interpretations of the Yucca Mountain region could be organized into 11 tectonic models. Staffs from DOE, NRC, CNWRA, the U.S. Geological Survey, and the State of Nevada met in San Antonio, Texas, on May 7–8, 1996, for an Appendix 7 meeting to discuss conceptual tectonic models. In this meeting, the 11 tectonic models proposed for the Yucca Mountain region were reviewed based on the most recent geological and geophysical data.

From discussions in the meetings, it was clear that 5 out of the 11 tectonic models were supported by the existing data (NRC, 1998, 1999a, Appendix C–1). In addition, there was no general consensus among the attendees at the Appendix 7 meeting on which models are truly independent and which models may function as subsets of others. Since that meeting, staff conclude that in a broader sense, these five models can be considered within two general categories of deformation. The first three models are dominantly related to extensional deformation, and the other two are dominantly related to strike-slip deformation. Moreover, the five models are not mutually exclusive. Locally, extensional-dominated deformation (e.g., within Crater Flat) can exist within a larger region of transtensional deformation related to a pull-apart basin. Potential implications of the five viable models to repository performance subissues are summarized in NRC (1998, Appendix C–3; 1999a, Appendix C–1).

Since the 1996 Appendix 7 meeting, the classification of the tectonic models has changed [e.g., the full range of tectonic models was presented to the DOE expert elicitation panel, who then developed a suite of models to describe the alternative interpretations (CRWMS M&O, 1998c; Stepp, et al., 2001)]. In CRWMS M&O (2000a), 4 categories of tectonic models are described that incorporate elements of the originally proposed list of 11: (i) Crater Flat caldera model, (ii) detachment fault models, (iii) rift/graben (elastic-viscous) models, and (iv) lateral-shear/pull-apart basin models.

Repository Safety Before Permanent Closure

Staff reviewed the development and application of tectonics models in postclosure performance assessments (including development of the probabilistic seismic hazard assessment) and have classified the subissue as closed for prelicensing (see Section 1.2 for definition of closed) (NRC, 1998). DOE has sufficient information with regard to the postclosure aspects of seismic and faulting hazards analyses. In that assessment, staff recommended that (i) the full range of tectonic models, as presented in the probabilistic seismic hazard assessment (CRWMS M&O, 1998c; Stepp, et al., 2001), should be applied uniformly and with continuity across the entire DOE analysis of Yucca Mountain, as appropriate; (ii) classification of specific models as preferred or favored is be avoided because these terms present a negative connotation; and (iii) DOE should continue to evaluate new scientific information with regard to the regional tectonics as necessary. These recommendations also apply to the site description of regional tectonic models as it relates to preclosure safety analyses.

The DOE findings (CRWMS M&O, 2000a) about the site geology are consistent with the regional tectonic models described in previous staff reviews (e.g., NRC, 1999a). In addition, the DOE review provides a comprehensive summary of data, results, and interpretations of tectonic models similar to previous staff reviews (e.g., NRC, 1999a). Thus, the DOE regional tectonic model summary provides sufficient technical bases for the descriptive and process models used to assess the ability of the natural system to help meet preclosure safety performance objectives.

Since the 1999 staff review and summary of the site description (CRWMS M&O, 2000a), there is a newly published regional reconstruction of Basin and Range extension (Snow and Wernicke, 2000). This new paper presents a regional reconstruction that includes significant Miocene (24–5 million years) detachment faulting with vertical- and horizontal-axis rotations of many of the major ranges including Bare Mountain. DOE should evaluate the new tectonic interpretations in Snow and Wernicke (2000) and modify the existing summary of the regional tectonic models as needed.

Quaternary Stratigraphy and Surficial Processes

The Quaternary stratigraphy of the Yucca Mountain region yields geological information used to assess (i) recent faulting activity, (ii) inter-arrival times between large earthquakes on major faults, (iii) ongoing tectonic activity, (iv) recent volcanism, (v) paleoclimates, and (vi) erosion rates. Landform evolution created by surficial processes is also important to issues of land use in the vicinity of Yucca Mountain. Land use is an important consideration in the biosphere model used for performance assessment. CRWMS M&O (2000a) provides a comprehensive summary of the Quaternary stratigraphy and surficial processes. The summary gleans information from a variety of DOE, U.S. Geological Survey, and State of Nevada reports as well as from geologic literature published in professional journals. Technical work related to characterization of seismic sources (e.g., U.S. Geologic Survey, 1996) and to possible anomalous influxes of hydrothermal waters during seismic events (e.g., Taylor and Huckins, 1995) provides much of the detailed mapping and interpretations.

Eight Quaternary alluvial units were recognized within the Yucca Mountain region (U.S. Geological Survey, 1996). These alluvial units range in age from 1,650 thousand years to the present. Their stratigraphy forms the basis for many paleoseismic interpretations in which ages and amounts of fault displacements were determined from relative juxtapositions of the eight alluvial units across active fault zones. This information was used by the DOE expert elicitation panel in its construction of the Yucca Mountain probabilistic seismic hazard assessment (CRWMS M&O, 1998c; Stepp, et al., 2001). Results from the probabilistic seismic hazard assessment are used for both post and preclosure performance assessments and as input to the preclosure seismic design.

The DOE summary of the Quaternary stratigraphy and surficial processes (CRWMS M&O, 2000a) provides sufficient technical bases for the descriptive and process models used to assess the ability of the natural system to help meet preclosure safety performance objectives, with the exception of the site-specific criteria and seismic response models.

For preclosure seismic design, specific information on the Quaternary alluvium at the facility site is necessary to construct a site response model of earthquake-induced ground motions. DOE collected site information from approximately 20 test borings and several test pits and trenches, but that information has not yet been provided to the staff for review. DOE established a timetable for release of the information that includes the Seismic Design Inputs Report in September 2001 and the Seismic Topical Report 3 in fiscal year 2002.[1,2] Thus, staff consider this portion of the site description closed, pending submission of the necessary and promised information from DOE. Details of the application of DOE information on preclosure hazard assessments from natural surficial processes are provided within their respective sections of this Integrated Issue Resolution Status Report.

<u>Site Stratigraphy</u>

Site stratigraphy forms the framework for modeling and analyses of rock properties, mineral distributions, faulting, fracturing, hydrologic flow, radionuclide transport, performance assessment, and subsurface repository design. The exposed stratigraphic sequence at Yucca Mountain is composed of Middle to Late Miocene (16.6–5.3 million years) volcanic strata. These volcanic rocks consist mostly of pyroclastic flow and fallout tephra deposits with minor lava flows and reworked materials erupted from the southwestern Nevada volcanic field between 15.2 and 11.4 million years ago (Sawyer, et al., 1994).

Because of their importance for understanding geologic systems at Yucca Mountain, the volcanic rocks have been a major focus of stratigraphic studies being conducted as part of the site characterization program. Many investigations of the Yucca Mountain area have focused on mapable, lithostratigraphic, hydrogeologic, and thermal-mechanical properties of the tuffs. Each type of investigation has led to its own stratigraphic system (Scott and Bonk,1984; Buesch, et al., 1996; Flint, 1998; Ortiz, et al., 1985). Table 4.5-3 of CRWMS M&O (2000a)

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[1]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Structural Deformation and Seismicity (October 11–12, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[2]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Repository Design and Thermal-Mechanical Effects (February 6–8, 2001)." Letter (February 28) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

provides a cross-correlation of these different stratigraphic units. Different compositions of the volcanic magma, eruption types (effusive versus explosive), cooling histories, and transport and deposition mechanisms combine to produce the range of depositional features observed in the Yucca Mountain strata.

The two most critical tuff units to the preclosure safety analysis are the Paintbrush Group tuffs including Tiva Canyon and the Topopah Springs Tuff. These two units make up the bulk of exposed volcanic rocks at Yucca Mountain. The Topopah Spring Tuff includes the host rock units for the potential repository and, as such, its characteristics are of direct importance to repository design. At Yucca Mountain, the Topopah Spring Tuff has a maximum thickness of approximately 380 m [1,247 ft]. The formation is divided into a lower crystal-poor member and an upper crystal-rich member. Each member is then divided further into numerous zones, subzones, and intervals based on variations in crystal content and assemblage, size and abundance of pumice and lithic clasts, distribution of welding and crystallization zones, and fracture characteristics (Buesch, et al., 1996). The Tiva Canyon Tuff is a large-volume, regionally extensive, silica-rich tuff sequence that forms most of the rocks exposed at the surface of Yucca Mountain (Day, et al., 1997, 1998).

CRWMS M&O (2000a) and numerous references therein provide a detailed and comprehensive summary of the site stratigraphic work. The DOE regional geologic setting summary provides sufficient technical bases for the site stratigraphy used to assess the ability of the natural system to help meet preclosure safety performance objectives.

<u>Site Structural Geology</u>

Site structural geology of Yucca Mountain describes the spatial and temporal patterns of faulting and fracturing of the Miocene Age volcanic bedrock at the Yucca Mountain potential repository site. An understanding of faulting and fracturing is important to the design of a potential repository and to the evaluation of its ability to meet preclosure safety performance goals. The structural geologic setting of Yucca Mountain is used to evaluate the amount and quality of rock available for underground construction, identification, and characterization of hydrologic flow paths and the assessment of seismic and fault displacement hazards.

Yucca Mountain comprises a thick accumulation of volcanic tuff deposited on an irregular surface of eroded and deformed Paleozoic and Precambrian basement composed of highly faulted and folded sedimentary and metasedimentary rocks. These tuffs were erupted from a series of Middle to Late Miocene (15–9 million years) calderas that collectively form what has been defined as the southwestern Nevada volcanic field. Sawyer, et al. (1994) provide the most recent comprehensive regional stratigraphy of the Miocene volcanic rocks in the Yucca Mountain region. Rocks of the Paintbrush Group, principally Tiva Canyon Tuff (12.7 million years), make up the main surface exposures of Yucca Mountain, hereas the repository horizon is within the Topopah Springs Tuff (12.8 million years). The Paintbrush Group tuffs rest on a sequence of older tuffs, including the Prow Pass and Bullfrog members of the Crater Flat Group. Younger tuffs related to the Timber Mountain Group are locally exposed at Yucca Mountain in topographic lows between large block-bounding faults. This observation, along with evidence for growth faults in the Paintbrush rocks in Solitario Canyon (e.g., Carr, 1990; Day, et al., 1997), suggests that faulting and tuff deposition were synchronous at

Yucca Mountain. Trenching studies of the Solitario, Paintbrush Canyon, and Bow Ridge faults also show sufficient evidence for multiple faulting events in the Quaternary (U.S. Geological Survey, 1996, Sections 4.6 and 4.7). Thus, it appears that faulting has been active throughout the geologic history of Yucca Mountain, although present-day rates of fault movement are significantly lower than in the late Miocene, when volcanic rocks at Yucca Mountain were first deposited.

The majority of faults at Yucca Mountain are either north-trending normal faults or northwest-trending, dextral strike-slip faults. The larger faults in these two orientations bound the fault blocks that underlie Yucca Mountain. These two sets of faults are interpreted to be contemporaneous, based on mutual terminations and secondary structures between them, such as pull-apart basins (Day, et al., 1997, 1998). Some northwest-trending faults are dominantly normal faults, accommodating extension in relay ramps between overlapping normal faults (Ferrill, et al., 1999). Only four reverse faults with north-south or northeast-southwest strikes have been identified, but they are potentially key features for constraining the kinematic history of the region (Day, et al., 1998) and for identifying infiltration pathways (Levy, et al., 1997). Much of the detailed fieldwork to study faults in the central block focused on the Ghost Dance and Sundance faults, which are close to the subsurface trace of the Exploratory Studies Facility (Spengler, et al., 1994; Potter, et al., 1996).

Yucca Mountain consists of a sequence of north to north-northeast trending, fault-bound ridges crossed by occasional northwest-trending, dextral strike-slip faults. Faults dip almost uniformly to the west and separate blocks of gentle to moderate east-dipping tuff strata. From north to south, both fault displacement and dip of bedding increase and, thus, indicate progressively greater extension of the Crater Flat basin southward (Scott, 1990). This pattern is most profound on the west flank of Yucca Mountain, which is defined by a series of left-stepping and north-trending *en echelon* faults. The southward increase in fault offset is coupled with greater block rotation, both horizontal and vertical (Scott, 1990). Work by the U.S. Geological Survey suggests that this pattern of faulting, along with rotated paleomagnetic direction in the tuffs, resulted from a discrete period of extension followed by a discrete period of dextral shear, akin to an oroclinal bending model (Hudson, et al., 1994; Minor, et al., 1997).

More recent reanalyses of these data suggest an alternative explanation. The north-to-south displacement gradient and rotation of fault blocks are a result of increased rollover deformation in the hanging wall above a listric Bare Mountain fault (Ferrill, et al., 1996; Ferrill and Morris, 1997; Stamatakos and Ferrill, 1998; Morris and Ferrill, 1999).

An *en echelon* pattern of faulting is best expressed along the western edge of Yucca Crest and the fault line escarpment that follows the west-dipping Solitario Canyon, Iron Ridge, and Stagecoach Road faults (e.g., Simonds, et al., 1995). The geometry of faults and ridges defines a scallop trend composed of linear, north-trending fault segments connected by discrete curvilinear northwest-trending fault segments. For example, the ends of the northwest-trending curvilinear Iron Ridge fault bend to the northwest near its overlap with both the Stagecoach Road and Solitario Canyon faults. Yucca Mountain also contains numerous swarms of small northwest-trending faults that connect the large north-trending faults. One example is at West Ridge, which is cut by numerous small faults that connect segments of the Windy Wash and Fatigue Wash faults. This geometry strongly suggests that the entire Yucca Mountain fault system is an *en echelon* branching fault system (Ferrill, et al., 1999) in which faulting on the large block-bounding fault triggers relatively widespread, but predictable, secondary faulting on connecting and linking faults. Linkage of the *en echelon* system is either by lateral propagation of curved fault tips or formation of connecting faults that breach the relay ramps (Ferrill, et al., 1999, Figure 1; Peacock and Sanderson, 1994; Trudgill and Cartwright, 1994). More importantly, from this interpretation of *en echelon* faulting, it follows that locally developed faults and fractures were produced by local variations of the stress field (e.g., Crider and Pollard, 1998) rather than dramatic swings of the regional extension direction (Throckmorton and Verbeek, 1995). The amount, orientation, and degree of faulting directly depend on the relative position of the rock within the *en echelon* fault system, either in relay ramps that connect overlapping *en echelon* fault segments or in the hangingwall or footwall blocks of the block-bounding faults.

Fracturing of the volcanic rocks at Yucca Mountain started soon after deposition of the volcanic tuffs about 11–13 million years ago. The first fractures of the volcanic rocks were probably cooling fractures (also commonly referred to as cooling joints). Soon after deposition of the tuffs, tectonic and gravitational forces caused additional fracturing of the tuffs. Cooling, tectonic, and unloading fractures constitute the naturally occurring fracture system at Yucca Mountain. Because the region is still tectonically active with erosion, both tectonic and unloading joints continue to form. Manmade fractures in drifts at Yucca Mountain are also present, formed by excavation of the tunnels and drifts. As discussed in the preceding paragraphs, faults are also prominent features of the structural framework at Yucca Mountain. Small faults and shear joints (up to meters in length and of small displacement) grade upward in scale to large features (hundreds of meters, in the case of joints, and tens of kilometers, in the case of faults). NRC (1999a) provides a comprehensive discussion of fractures and fracture studies at Yucca Mountain.

For preclosure safety analysis, the most critical aspect of fracture characterization is the statistical representation of the various fracture sets. The statistical properties of fractures (most notably fracture intensity and orientation) are used to assess the stability of subsurface openings and potential rockfall characteristics, especially the size of rock blocks that may fall on the waste packages. Azimuthal orientation of the drifts within the proposed repository is optimized to ensure large block volumes are minimized (i.e., drifts perpendicular to the dominant fracture orientation).

Nevertheless, staff analyses (e.g., NRC, 1999a) have shown that characterization of fracture networks at Yucca Mountain is impaired by several important sampling biases common to fracture analyses. If left uncorrected, these sampling biases lead to underrepresentation of fracture intensity and misrepresentation of fracture-set orientations. For example, because of the limited diameter of the Exploratory Studies Facility {7 m [23 ft]}, the lengths of the longest fractures are often unconstrained. The ends of the fracture are simply obscured in unexposed rock. In addition, the orientation of a one-dimensional sampling line (e.g., borehole or detailed line survey scanline) or two-dimensional sampling surface (e.g., pavement, roadcut, or tunnel surface) inherently biases sampling against discontinuities parallel to the sampling line or surface and in favor of sampling discontinuities at a high angle to the sampling line or surface. Mathematical corrections (Terzaghi, 1965) can partially compensate for this sampling bias. Finally, because measuring every fracture from the microscale to megascale is impractical or impossible for large sample areas, fracture studies usually invoke a size (e.g., length) cutoff. This was commonly 1 m [3 ft] in the Yucca Mountain studies. Fractures smaller than that cutoff dimension are simply not counted. Consequently, small fractures are underrepresented in fracture characterizations. Exclusion of small fractures may skew fracture-intensity determinations.

CRWMS M&O (2000a) provides a summary of the site structural geology. The summary gleans information from a variety of DOE, U.S. Geological Survey, and State of Nevada reports as well as from geologic literature published in professional journals. Nevertheless, as discussed at the October 2000 technical exchange between DOE and NRC, several areas of the DOE site characterization, especially with regard to fractures and fracture geometry, require additional information. DOE has agreed to a plan and schedule for providing the needed information prior to license application submittal.

Of particular importance to preclosure safety and design is the potential for sampling bias of fracture orientations. For example, DOE developed a drift layout plan of the potential repository (azimuths of drifts) based on assumptions of the measured fracture orientations at Yucca Mountain. DOE wants to minimize block volumes of potential rockfalls by aligning the drifts perpendicular to the azimuth of the dominant fracture set. Staff have previously commented that the statistical representation of fracture orientations, based on the measured fractures at Yucca Mountain may contain a sampling bias such that the actual fracture orientations are different from those used in the DOE design calculation (NRC, 1999a). DOE agreed to provide that information prior to submitting a potential license application.[3] Thus, the DOE structural geology summary does not yet provide sufficient technical bases for the descriptive and process models used to assess the ability of the natural system to help meet preclosure safety performance objectives, but DOE has agreed to a plan and schedule for providing the needed information prior to license application submittal.

Site Geoengineering Properties

Staff review of the information provided by DOE on site geoengineering properties is discussed in Section 2.1.7 of this Integrated Issue Resolution Status Report.

Staff have not fully reviewed the information provided by DOE on geoengineering properties for surface-facility design. Future revisions of the Integrated Issue Resolution Status Report will provide staff assessment of this aspect of the Yucca Mountain site description.

Integrated Site Model

The Integrated Site Model of Yucca Mountain is a three-dimensional representation of the rock layers and faults, rock properties, and minerals in the subsurface at Yucca Mountain. The models provide a baseline representation of the geology of the site for use in hydrologic flow,

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[3]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Structural Deformation and Seismicity (October 11–12, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

radionuclide transport, repository design, and performance assessment modeling. The Integrated Site Model consists of three components:

- Geologic Framework Model
- Rock Properties Model (except Thermal-Mechanical Properties)
- Mineralogical Model

DOE developed the Integrated Site Model to provide a consistent volumetric portrayal of the rock layers, several rock properties, faults, and mineral distributions in the subsurface of Yucca Mountain. DOE provided detailed descriptions of the three component models of the Integrated Site Model in CRWMS M&O (2000d) with attendant analysis and model reports (CRWMS M&O, 2000e,f,g).

A DOE contractor constructed the Geological Framework Model Version 3.1 (CRWMS M&O, 2000h) using quality assurance approved EarthVision software, Version 4.0. The staff reviewed Geological Framework Model Version 3.1 (NRC, 1999a, Appendix F) and found it to be a largely credible digital three-dimensional representation of the stratigraphy, faults, fault blocks, and topography of Yucca Mountain at the site-scale. The Geological Framework Model Version 3.1 (CRWMS M&O, 2000h) adequately represents the site scale, three-dimensional geologic framework of Yucca Mountain. Though Geological Framework Model Version 3.1 (CRWMS M&O, 2000h) is deemed credible, it should not be considered the final step to develop a geologic framework model for Yucca Mountain because any additional fault data obtained or any new interpretations formulated should be incorporated into the model. This is particularly true for the outer and deeper portions of the model where subsurface data used to constrain the model are sparse. DOE clearly indicated that Geological Framework Model Version 3.1 (CRWMS M&O, 2000h) as it presently exists is not intended to represent a tectonic model. The level of detail and accuracy of stratigraphic horizon and fault representations in Geological Framework Model Version 3.1 (CRWMS M&O, 2000h) are adequate as a geologic framework for the Integrated Site Model. Presently, no major problems exist with abstracting stratigraphic horizons or fault surfaces in Geological Framework Model Version 3.1 (CRWMS M&O, 2000h) to process models. At this time, there are no major discrepancies related to representation of stratigraphic horizons or faults that would preclude DOE from using Geological Framework Model Version 3.1 (CRWMS M&O, 2000h).

Staff have not fully reviewed all aspects of the Rock Properties and Mineralogical Model components of the Integrated Site Model as they relate to preclosure safety. Future revisions of the Integrated Issue Resolution Status Report will provide staff assessment of this aspect of the Yucca Mountain site description.

<u>Natural Resources</u>

Natural resource assessments of the Yucca Mountain region by DOE have focused on an area defined as the conceptual controlled area or the natural resources site study area summarized in CRWMS M&O (2000i). The DOE assessment of natural resources focused on natural occurrences of metallic minerals, industrial rocks and minerals, hydrocarbons (petroleum, natural gas, oil shale, tar sands, and coal), and geothermal energy either already known to exist within the region that could reasonably exist based on models of natural resource occurrence or analogous regions with a similar geologic setting (i.e., other regions primarily within the southern Great Basin).

Staff have not fully reviewed all aspects of the DOE summary of the natural resources as they relate to preclosure safety. Future revisions of the Integrated Issue Resolution Status Report will provide staff assessment of this aspect of the Yucca Mountain site description.

Rock Properties

The scope of acceptance criteria on rock properties includes confirmation that site characterization data include geomechanical properties and conditions of host rock for the rock formations where major construction activities will occur. Staff review of the information provided by DOE on geoengineering properties for subsurface design has been discussed in Section 2.1.7 of the Integrated Issue Resolution Status Report.

Stability and Suitability of Subsurface Materials

The scope of acceptance criteria on stability and suitability of subsurface materials requires verification that rock mechanics testing data support the license application analyses of the stability of subsurface materials. Staff review of the information provided by DOE on geoengineering properties for subsurface design has been discussed in Section 2.1.7 of this Integrated Issue Resolution Status Report.

Soil Properties

The acceptance criteria on soil properties will be satisfied if it DOE presents sufficient soil properties information appropriate for the design of structures, systems, and components important to safety.

Staff have not reviewed the DOE information on soil properties as they relate to preclosure safety. Future revisions of the Integrated Issue Resolution Status Report will provide staff assessment of this aspect of the Yucca Mountain site description.

Stability and Suitability of Surface Materials

Staff have not reviewed the DOE information on the stability and suitability of surface materials as they relate to preclosure safety. Future revisions of the Integrated Issue Resolution Status Report will provide staff assessment of this aspect of the Yucca Mountain site description.

Seismic and Faulting Hazards

DOE calculation of seismic and fault displacements hazards for both pre and postclosure analyses was developed from a probabilistic seismic hazard analysis conducted by DOE (CRWMS M&O, 1998c; Stepp, et al., 2001). In the probabilistic seismic hazard analysis, DOE used six teams of experts. Each team consisted of three specialized geoscientists with expertise in either paleoseismology, Basin and Range structural geology, or Basin and Range seismology. To assess seismic sources, the teams mainly relied on information provided by the

U.S. Geological Survey, DOE, and related Yucca Mountain studies augmented by published literature. In addition, the teams were assembled for six workshops, held between April 1995 and June 1997, at which the experts exchanged information on seismic sources and participated in additional discussions with other external experts. Details of the workshops are given in CRWMS M&O (1998c).

In 10 CFR 100.23, NRC identified a probabilistic approach to seismic hazard analysis as an appropriate method to address uncertainties associated with earthquake-induced ground motions. DOE (1996) outlined the methodology used for its probabilistic seismic hazard analysis, which was accepted, in principle, by NRC.[4] The methodologies recommended in NRC (1996) also offer acceptable approaches for evaluating the probabilistic seismic hazard at Yucca Mountain.

Similar to the seismic hazard assessment, DOE used the same expert elicitation to develop a probabilistic fault displacement hazard assessment. The objective of fault displacement analyses was to evaluate the potential hazards of an active fault intersecting vital components of the engineered barrier subsystem, especially waste packages.

Staff assessment of the DOE probabilistic seismic and fault displacement hazard analyses is discussed in Section 3.3.2, Mechanical Disruption of Engineered Barriers, and in an NRC report (1999a). For preclosure issues, DOE has yet to provide all the information necessary for staff to complete its review. In particular, DOE has not yet established specific seismic site response models for important surface facilities. DOE agreed to provide information that includes the Seismic Design Inputs Report and the Seismic Topical Report 3.[5,6]

Seismic Design

Staff have not reviewed the DOE information on the seismic design with respect to preclosure as it relates to preclosure safety. Future revisions of the Integrated Issue Resolution Status Report will provide a staff assessment of this aspect of the Yucca Mountain site description.

Facility Stability

Staff have not reviewed the DOE information on facility stability with respect to preclosure safety. Future revisions of the Integrated Issue Resolution Status Report will provide staff assessment of this aspect of the Yucca Mountain site description.

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[4]Bell, M.J. "Issue Resolution Status Report on Methodology to Assess Fault Displacement and Vibratory Ground Motion Hazard at Yucca Mountain." Letter (July 25) to S.J. Brocoum, DOE. Washington, DC: NRC. 1996.

[5]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Structural Deformation and Seismicity (October 11–12, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[6]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Repository Design and Thermal-Mechanical Effects (February 6–8, 2001)." Letter (February 28) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

2.1.1.3.6 Igneous Activity

The following sections on igneous activity refer to the requirements of 10 CFR 63.112(c). The potential DOE license application should contain a description of the historical regional igneous activity adequate to permit evaluation of the preclosure safety analysis and the Geologic Repository Operations Area design.

Distributed basaltic volcanism is a long-lived characteristic of the Yucca Mountain region. Since the end of large-scale silicic caldera activity around 11 million years, approximately 12 igneous events are known to have occurred within 30 km [19 mi] of the proposed repository site. Each of these igneous events consisted of one to four volcanic cinder cones and multiple subsurface intrusions that extend for kilometers away from the volcano. Basaltic cinder cones form during eruptions that typically have 2–8-km [1–5-mi]-high eruption columns. These eruption columns can disperse fragments of quenched magma (i.e., tephra) tens of kilometers from the vent. Basaltic tephra-fall deposits 20 km [12 mi] from the volcano are generally 1–100 cm [0.4–39 in] thick with bulk densities of 1,200–1,700 kg/m^3 [75–106 lb/ft^3] (e.g., Hill, et al., 1998; NRC, 1999b).

In the preliminary external hazards analysis, DOE generated a potential external hazards list from a generic check list of natural phenomena. DOE selected potential natural phenomena through a screening process. These selected events have been further screened through additional analyses, and bounding natural events that could lead to potential radiological release have been identified. The DOE event preventive strategy is to design the structures, systems, and components important to safety to withstand the bounding natural design basis events. DOE should demonstrate that determination of frequencies of the events is defensible and also provide design bases and design criteria used to mitigate design basis events (DOE, 1999b). For example, the selected natural phenomena do not include volcanic tephra-fall as a design basis event.

DOE concludes that no more than 3 cm [1 in] of volcanic tephra could be deposited on repository facilities during the preclosure period (1999b). DOE thus excluded roof loading caused by tephra fall from further consideration, because the load imparted by a 3-cm [1-in]-thick tephra deposit is bounded by the minimum design load requirements specified by the Uniform Building Code. Additionally, the effects of volcanic tephra on air filters and ventilation systems are considered bounded by sandstorms (DOE, 1999b).

Available analysis or data do not support the basis for concluding that a 3-cm [1-in]-thick volcanic tephra deposit is the worst-case event. The 3-cm [1-in]-thick deposit cited in DOE (1999b) applies only for a volcanic eruption occurring 150 km [93 mi] from the proposed repository site (i.e., Perry and Crowe, 1987). Basaltic volcanic eruptions have an annual probability of occurrence that exceeds 1×10^{-6} within 10 km [6 mi] of the proposed repository site (e.g., NRC, 1999b). Tephra-fall deposits measured about 10 km [6 mi] from volcanoes analogous to those within 20 km [12 mi] of Yucca Mountain are on the order of 1–100 cm [1–39 in] thick (e.g., NRC, 1999b). These deposits increase in thickness to around 400 cm [158 in] within 1 km [1 mi] of the volcanic event. In addition, Perry and Crowe (1987) conclude that a 1-m [3-ft]-thick tephra-fall could occur approximately 3 km [2 mi] from a basaltic volcanic event. Noncompacted, dry basaltic volcanic tephra has bulk deposit densities that can range

1,200–1,700 kg/m^3 [75–106 lb/ft^3] (e.g., Hill, et al., 1998; NRC, 1999b). These deposit densities can increase by a rough factor of two when wet, depending on average grain size and sorting of the deposit. Thus, a basaltic volcanic eruption in the area around Yucca Mountain represents a Category 2 event that could deposit 1–400 cm [0.03–13 ft] of dry tephra on surface structures, resulting in dry loads between 12 and 6,800 kg/m^2 [2 and 1,390 lb/ft^2]. In addition, DOE has not provided a technical basis to determine the analogy of wind-blown sands to volcanic tephra particles. Volcanic tephra-fall deposits contain a greater range of particle sizes than wind-blown sands, which may have different effects on air filters and ventilation systems.

The DOE summary of igneous activity relevant to preclosure safety (DOE, 1999b) does not provide sufficient information to evaluate potential effects on the performance of surface facilities. DOE needs to provide additional information on the amount and character of potential tephra deposits that could fall on surface facilities from basaltic volcanic eruptions located within areas where the annual probability of a new volcano forming is $\geq 10^{-6}$. DOE should then evaluate the potential effects of these tephra-fall deposits on structures and systems important to safety.

2.1.1.3.7 Site Geomorphology

The following sections on site geomorphology refer to the requirements of 10 CFR 63.112(c). The potential DOE license application should contain a description of the site geomorphology adequate to permit evaluation of the preclosure safety analysis and the Geologic Repository Operations Area design.

For preclosure, site geomorphology refers to geologic processes of erosion and the likelihood that extreme erosion (e.g., landslides, rock avalanches, and other mass wasting and rapid fluvial degradation in channels or interfluves) might affect site structures and operations. Staff have not fully reviewed all aspects of the DOE summary of the site geomorphology as they relate to preclosure safety, although aspects of erosional hazards are addressed in Section 2.1.1.3.4, Regional and Local Surface and Groundwater Hydrology. Future revisions of the Integrated Issue Resolution Status Report will provide staff assessment of this aspect of the Yucca Mountain site description.

2.1.1.3.8 Site Geochemistry

The following sections on site geochemistry refer to the requirements of 10 CFR 63.112(c). The potential DOE license application should contain sufficient site geochemical information to support evaluation of the preclosure safety analysis and the Geologic Repository Operations Area design.

Geochemistry of Subsurface Waters

The unsaturated zone at Yucca Mountain contains pore waters, fracture waters, and isolated perched water (CRWMS M&O, 2000a). Yang, et al. (1996, 1998) measured chemical compositions of ambient pore water and perched water from Yucca Mountain and vicinity. Perched waters were sampled from boreholes using plastic bailers, and pore waters were extracted from borehole core samples using high-pressure uniaxial compression techniques.

Perched water and pore water compositions were measured using inductively coupled plasma spectroscopy and ion chromatography. Stratigraphic units penetrated by the boreholes are (in descending order) the Paintbrush Group (composed of Tiva Canyon Tuff, Yucca Mountain Tuff, Pah Canyon Tuff, and Topopah Spring Tuff), the Calico Hills Formation, and the Prow Pass Tuff. However, no ambient pore water compositions were reported from the Topapah Spring Tuff, because extraction techniques were apparently unable to produce an adequate volume of water from this tuff. There are also no measured fracture water compositions from Yucca Mountain because of the difficulty of collecting fracture water samples. However, fracture water has been collected from Rainier Mesa (White, et al., 1980) and appears to be similar in composition to perched and saturated zone waters collected at Yucca Mountain. Staff consider that the problems DOE experienced in collecting and analyzing pore water samples from the Topapah Spring Tuff and fracture water samples at Yucca Mountain were unavoidable, given the current state of extraction technologies.

The pore water analyses of Yang, et al. (1996, 1998) provide valuable characterizations of groundwater chemistry at Yucca Mountain, but there are indications that aspects of these data are unreliable. Yang, et al. (1996, 1998) noted charge imbalances in the chemical analyses. In addition, Apps (1997) concluded that measured pH values are inaccurate, based on inconsistencies of pH measurements of water from the J–13 Well. Browning, et al. (2000) noted that the range of analytical pH for pore waters extracted from similar depths within individual boreholes appears unreasonably wide, suggesting that measured pH values are unreliable. Browning, et al. (2000) noted similar abrupt variations in some reported major aqueous species concentrations. Potassium occurs in primary and secondary phases at Yucca Mountain and is an important component of Yucca Mountain waters, but Yang, et al. (1996, 1998) did not always report potassium concentrations. Finally, particulate aluminum in filtered samples resulted in unreliable aluminum concentrations (Yang, et al., 1996). Clearly, there are significant uncertainties in the pore water analyses of Yang, et al. (1996, 1998) that compromise the utility of these data. Apps (1997) and Browning, et al. (2000) propose different sets of assumptions for revising/improving these data using aqueous speciation calculations. DOE used little or none of the groundwater compositional data provided by Yang, et al. (1996, 1998); Apps (1997); or Browning, et al. (2000) in any process-level models providing input into the Total System Performance Assessment–Site Recommendation. DOE provided adequate information on ambient groundwater chemistry at Yucca Mountain, with the exception of some minor and trace components (see Section 3.3.3, Quality and Chemistry of Water Contacting Waste Packages and Waste Form, of this report). However, DOE sufficiently evaluated the preclosure and postclosure (see Section 3.3.3, Quality and Chemistry of Water Contacting Waste Packages and Waste Form, of this report) performance implications of the data.

Geochemistry of Rock Strata

CRWMS M&O (2000a) provides a summary of data provided by DOE on geochemical composition of the rock strata at Yucca Mountain. X-ray diffraction techniques were used to characterize the mineralogy of core samples from boreholes in the vicinity of Yucca Mountain. These data were combined with information from stratigraphic and potentiometric surfaces and incorporated into the three-dimensional Mineralogic Model part of the Geologic Framework model. The Mineralogic Model was designed as a resource to interpolate information about mineral assemblages between boreholes where measurements were made, and this model has been a useful effort. Although DOE provided sufficient information on matrix mineralogy via developing the Mineralogic Model, staff judge that more work is needed to characterize the mineralogy of fractures and lithophysal cavities for numerical modeling efforts, such as reactive transport modeling. DOE should provide additional information on the types of minerals present in fractures at Yucca Mountain and vicinity and quantify the relative abundances of these types of minerals.

<u>Geochemical Alterations</u>

The chemical compositions of ambient groundwater from Yucca Mountain are expected to evolve significantly before contacting drip shields and waste packages. Several different factors will control the composition of water as it percolates through the overlying rock toward the drift, including temperature, the types of materials that interact chemically with the water along the flow pathway, and flow velocity versus reaction rate. Thermal-hydrological models suggest that temperatures at the drift crown will remain above nominal boiling for approximately 1,000 years (CRWMS M&O, 2000j). These models suggest that ambient groundwater compositions should adequately characterize seepage compositions for the majority of the 10,000-year compliance period, but this is probably not true. It is unlikely that ambient pore water will ever drip in significant volumes from the drift crown at the Yucca Mountain repository because fractures are expected to be the predominant flow pathway to the drift. Even if ambient pore water drips in significant volumes, the effects likely would be unimportant to the lifetime of the drip shield/waste package because corrosion is enhanced in higher temperature, more saline solutions. After water seeps out of the porous rock, its chemical composition continues to evolve through evaporation and salt formation processes in the engineered barrier subsystem. Thus, ambient groundwater above the proposed repository will be subjected to thermal perturbations in several different environments that will change its chemical compositions during time. Predictions of the quantity and chemistry of water contacting the drip shields and waste packages throughout the 10,000-year compliance period for the proposed Yucca Mountain repository are thus difficult and must be accomplished by considering both analytical data and numerical models.

Section 3.3.3, Quality and Chemistry of Water Contacting Waste Packages and Waste Form, of this report presents staff concerns regarding the DOE approach to characterizing compositions of seepage water at the drift crown and evaporated water in the engineered barrier subsystem. Of these, the two most significant concerns for preclosure involve the DOE approach toward model validation and the treatment of data and model uncertainties.

2.1.1.4 Status and Path Forward

DOE and NRC have not yet held a technical exchange to outline prelicensing agreements related to the sufficiency of the DOE preclosure site description. Table 2.1.5-1 provides a summary of the preclosure items related to the site description with cross-references to related agreements in the postclosure key technical issues. The table forms the basis for pending discussion with DOE regarding preclosure site description. Sufficient is meant to indicate that DOE presented enough information for staff to conduct a license review, if DOE were to submit a license application. Those items considered pending require either additional review by staff or additional information from DOE.

Table 2.1.1-1. Summary of Resolution Status of Site Description Preclosure Topic

Preclosure Items	Status	Related Agreements	Comments
Site Geography	Pending	None	Current information sufficient, but site location information may need updates given proposed EPA Standard and design for an expanded repository (DOE, 2001).* Location of 13 surface facility features not yet provided in DOE designs. Current information sufficient, but site map may need updates given proposed EPA Standard and alternative design for expanded repository (DOE, 2001).*
Regional Demography	Pending	None	Demographic information needs to be updated to include fiscal year 2000 census data.
Local Meteorology and Regional Climatology	Pending	None	Staff review incomplete.
Regional and Local Surface and Groundwater Hydrology	Pending	None	Additional information needed to evaluate potential water and debris flows, siting criteria or ventilation shafts, maximum versus 100-year flood, 100-year flood design considerations, storage in Midway Valley, transportation across active drainages, and water influx along faults. Additional information also necessary for proposed alternative design for expanded repository (DOE, 2001).*

Table 2.1.1-1. Summary of Resolution Status of Site Description Preclosure Topic (continued)			
Preclosure Items	**Status**	**Related Agreements**	**Comments**
Site Geology and Seismology	Pending	RDTME.2.01 RDTME.2.02 RDTME.3.03 RDTME.3.04 SDS.1.02 SDS.2.01 SDS.2.02 SDS.2.03	Current information on regional geologic and tectonic setting as well as site stratigraphy is sufficient. Additional information may be necessary for proposed alternative design for expanded repository (DOE, 2001).* Site soil data necessary for seismic response models and site design. DOE agreed to provide information by time of license application.† DOE agreed to provide additional information on rock properties.† Expanded repository in alternative design (DOE, 2001)* requires additional DOE characterization. DOE agreed to provide additional information on probabilistic seismic and fault displacement hazard assessments.†
Igneous Activity	Pending	None	Inadequate technical bases for DOE evaluation of tephra deposition at the site.
Site Geomorphology	Pending	None	Staff review incomplete.
Site Geochemistry	Pending	None	DOE has not yet fully used available information for preclosure performance assessment. Additional information on types of minerals present in fractures necessary for reactive transport modeling. Inadequate treatment of model validation, data, and model uncertainties in the DOE approach.

*DOE. "Yucca Mountain Science and Engineering Report." DOE/RW–0539. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management, Yucca Mountain Site Characterization Project. 2001.
†Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Structural Deformation and Seismicity (October 11–12, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

2.1.1.5 References

Apps, J.A. "Hydrochemical Analysis." *The Site-Scale Unsaturated Zone Model of Yucca Mountain, Nevada for the Viability Assessment*. G.S. Bodvarsson, T.M. Bandurraga, and Y.S. Wu, eds. LBNL–40376. UC–814. Berkeley, California: Lawrence Berkeley National Laboratory. 1997.

Bennett, R.A., J.L. Davis, P. Elósegui, B.P. Wernicke, J.K. Snow, M.J. Abolins, M.A. House, G.L. Stirewalt, and D.A. Ferrill. "Global Positioning System Constraints on Fault Slip Rates in the Death Valley Region, California and Nevada." *Geophysical Research Letters.* Vol. 24. pp. 3,037–3,076. 1997.

Blakely, R.J., V.E. Langenheim, D.A. Ponce, and G.L. Dixon. "Aeromagnetic Survey of the Amargosa Desert, Nevada and California: A Tool for Understanding Near-Surface Geology and Hydrology." U.S. Geological Survey Open-File Report 00-188. 2000.

Blanton, J.O. "Nevada Test Site Flood Inundation Study, Part of a Geological Survey Flood Potential and Debris Hazard Study, Yucca Mountain Site." Denver, Colorado: U.S. Bureau of Reclamation. 1992.

Bohannon, R.G. and T. Parsons. "Tectonic Implications of Post-30 Ma Pacific and North American Relative Plate Motions." *Geological Society of America Bulletin*. Vol. 107. pp. 937–959. 1995.

Browning, L., W.M. Murphy, B.W. Leslie, and W.L. Dam. "Thermodynamic Interpretations of Chemical Analyses of Unsaturated Zone Water from Yucca Mountain, Nevada." Scientific Basis for Nuclear Waste Management XXIII. R.W. Smith and D.W. Shoesmith, eds. Symposium Proceedings 608. Pittsburgh, Pennsylvania: Materials Research Society. pp. 237–242. 2000.

Buesch, D.C., R.W. Spengler, T.C. Moyer, and J.K. Geslin. "Proposed Stratigraphic Nomenclature and Macroscopic Identification of Lithostratigraphic Units of the Paintbrush Group Exposed at Yucca Mountain, Nevada." U.S. Geological Survey Open-File Report 94-469. 1996.

Bullard, K.L. "PMF (Probable Maximum Flood) Study for Nevada Nuclear Waste Storage Investigation Project, GR–87–8." Denver, Colorado: U.S. Bureau of Reclamation. 1986.

Bullard, K.L. "Nevada Test Site Probable Maximum Flood Study, Part of a Geological Survey Flood Potential and Debris Hazard Study, Yucca Mountain Site for U.S. Department of Energy, Office of Civilian Radioactive Waste Management." Denver, Colorado: U.S. Bureau of Reclamation. 1994.

Carr, W.J. "Styles of Extension in the Nevada Test Site Region, Southern Walker Lane Belt: An Integrated Volcano-Tectonic and Detachment Fault Model." *Geological Society of America Memoir 176*. Boulder, Colorado: Geological Society of America. pp. 283–303. 1990.

CRWMS M&O. "Yucca Mountain Site Characterization Project: Summary of Socioeconomic Data Analyses Conducted in Support of the Radiological Monitoring Program, April 1997 to April 1998." MOL. 19980803.0064. Las Vegas, Nevada: CRWMS M&O. 1998a.

———. "Repository Surface Design Layout Analysis." BCB000000–01717–0200. Revision 02. Las Vegas, Nevada: CRWMS M&O. 1998b.

———. "Probabilistic Seismic Hazard Analyses for Fault Displacement and Vibratory Ground Motion at Yucca Mountain, Nevada, Final Report." WBS 1.2.3.2.8.3.6. Las Vegas, Nevada: CRWMS M&O. 1998c.

———. "Repository Surface Design Engineering Files Report." BCB000000–01717–5705–00009. Revision 03. Las Vegas, Nevada: CRWMS M&O. 1999.

———. "Yucca Mountain Site Description Document." TDR–CRW–GS–000001. Revision 01 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000a.

———. "Reference Design Description for a Geologic Repository." TDR–MGR–SE–000008. Revision 03 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000b.

———. "Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation." TDR–MGR–SE–000009. Revision 00 ICN 02. Las Vegas, Nevada: CRWMS M&O. 2000c.

———. "Geologic Framework Model (GFM3.1)." MDL–NBS–GS–000002. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000d.

———. "Rock Properties Model (RPM3.1)." MDL–NBS–GS–000004. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000e.

———. "Mineralogical Model (MM3.0)." MDL–NBS–GS–000003. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000f.

———. "Integrated Site Model Process Model Report." TDR–NBS–GS–000002. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000g.

———. "Geologic Framework Model (GFM3.1)." MDL–NBS–GS–000002. Revision 00 ICN 01. North Las Vegas, Nevada: CRWMS M&O. 2000h.

———. "Natural Resources Assessment." Input Transmittal 00313.T. Las Vegas, Nevada: CRWMS M&O. 2000i.

———. "Multiscale Thermohydrologic Model." ANL–EBS–MD–000049. Revision 00. North Las Vegas, Nevada. CRWMS M&O. 2000j.

Crider, J.G. and D.D. Pollard. “Fault Linkage: Three-Dimensional Mechanical Interaction Between Echelon Normal Faults.” *Journal of Geophysical Research*. Vol. 103. pp. 4,373–24,391. 1998.

Day, W.C., C.J. Potter, D.S. Sweetkind, R.P. Dickerson, and C.A. San Juan. “Bedrock Geologic Map of the Central Block Area, Yucca Mountain, Nye County, Nevada.” U.S. Geological Survey Miscellaneous Investigations Series, Map I–2601. Scale 1:6000. 1997.

Day, W.C., R.P. Dickerson, C.J. Potter, D.S. Sweetkind, C.A. San Juan, R.M. Drake, II, and C.J. Fridrich. “Geologic Map of the Yucca Mountain Area, Nye County, Nevada.” U.S. Geological Survey Miscellaneous Investigations Series, Map I–2627. Scale 1:24,000. 1998.

Dixon, T.H., S. Robaudo, J. Lee, and M.C. Reheis. “Constraints on Present-Day Basin and Range Deformation from Space Geodesy.” *Tectonics.* Vol. 14. pp. 755–772. 1995.

DOE. “Technical Basis Report for Surface Characteristics, Preclosure Hydrology, and Erosion.” YMP/TBR–0001. Revision 0. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management, Yucca Mountain Site Characterization Project. 1995.

———. “Preclosure Seismic Design Methodology for a Geologic Repository at Yucca Mountain.” Topical Report YMP/TR–003–NP. Revision 1. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 1996.

———. “Draft Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada.” DOE/EIS–0250D. Washington, DC: DOE, Office of Civilian Radioactive Waste Management. 1999a.

———. “MGR External Events Hazards Analysis.” ANL–MGR–SE–000004. Revision 00. Las Vegas, Nevada: CRWMS M&O. 1999b.

———. “Yucca Mountain Science and Engineering Report.” DOE/RW–0539. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management, Yucca Mountain Site Characterization Project. 2001.

England, P. and J. Jackson. “Active Deformation of the Continents.” *Annual Reviews of Earth and Planetary Sciences*. Vol. 17. pp. 197–226. 1989.

Ferrill, D.A. and A.P. Morris. “Geometric Considerations of Deformation Above Curved Normal Faults and Salt Evacuation Surfaces.” *The Leading Edge*. Vol. 16. pp. 1,129–1,133. 1997.

Ferrill, D.A., G.L. Stirewalt, D.B. Henderson, J.A. Stamatakos, K.H. Spivey, and B.P. Wernicke. NUREG/CR–6401, “Faulting in the Yucca Mountain Region, Critical Review and Analyses of Tectonic Data from the Central Basin and Range.” Washington, DC: NRC. 1996.

Ferrill, D.A., J.A. Stamatakos, and D. Sims. “Normal Fault Corrugation: Implications for Growth and Seismicity of Active Normal Faults.” *Journal of Structural Geology*. Vol. 21. pp. 1,027–1,038. 1999.

Flint, L.E. “Characterization of Hydrogeologic Units Using Matrix Properties, Yucca Mountain Nevada.” U.S. Geological Survey Water Resources Investigations Report 97-4243. 1998.

Harbert, W. and A. Cox. “Late Neogene Motion of the Pacific Plate." *Journal of Geophysical Research.* Vol. 94. pp. 3,052–3,064. 1989.

Hill, B.E., C.B. Connor, M.S. Jarzemba, P.C. La Femina, M. Navarro, and W. Strauch. “1995 Eruptions of Cerro Negro Volcano, Nicaragua, and Risk Assessment for Future Eruptions.” *Geological Society of America Bulletin*. Vol. 10: pp. 1,231–1,241. 1998.

Hudson, M.R., D.A. Sawyer, and R.G. Warren. “Paleomagnetism and Rotation Constraints for the Miocene Southwestern Nevada Volcanic Field.” *Tectonics.* Vol. 13. pp. 258–277. 1994.

Jones, C.H., J.R. Unruh, and L.J. Sonder. “The Role of Gravitational Potential Energy in Active Deformation in the Southwestern United States.” *Nature*. Vol. 381. pp. 37–41. 1996.

Levy, S.S., D.S. Sweetkind, J. Fabryka-Martin, P.R. Dixon, J.L. Roach, L.E. Wolfsberg, D. Elmore, and P. Sharma. “Investigations of Structural Controls and Mineralogic Associations of Chlorine-36 Fast Pathways in the ESF.” Los Alamos Milestone SP2301M4. Los Alamos, New Mexico: Los Alamos National Laboratory. 1997.

McKague, H.L, J.A. Stamatakos, and D.A. Ferrill. “Type 1 Faults in the Yucca Mountain Region.” CNWRA 96-007. Revision 1. San Antonio, Texas: CNWRA. 1996.

Minor, S.A., M.R. Hudson, and C.J. Fridrich. “Fault-Slip Data, Paleomagnetic Data, and Paleostress Analyses Bearing on the Neogene Tectonic Evolution of Northern Crater Flat Basin, Nevada.” U.S. Geological Survey Open-File Report 97-285. 1997.

Morris, A.P. and D.A. Ferrill. “Constant-Thickness Deformation Above Curved Normal Faults.” *Journal of Structural Geology.* Vol. 21. pp. 67–83. 1999.

Nevada State Demographer. “Nevada Population Estimates by County 1990 through 1998.” Reno, Nevada: Nevada Department of Taxation. 1999a. <http://www.state.nv.us/budget/sapop.html> (14 June 1999).

———. “Population of Nevada's Incorporated Cities 1995–1998.” Reno, Nevada: Nevada State Demographer. 1999b. <http://www.state.nv.us/budget/sapop.html> (11 June 1999).

———. “Population of Nevada's Unincorporated Towns—Historical Governor's Certified Series.” Reno, Nevada: Nevada State Demographer. 1999c. <http://www.scs.unr.edu/demographer> (11 June 1999).

NRC. NUREG–0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants.” Section 2.4.3: Probable Maximum Flood (PMF) on Streams and Rivers. LWR Edition. Washington, DC: NRC. 1987. Updated April 1989.

———. “Issue Resolution Status Report, Key Technical Issue: Structural Deformation and Seismicity.” Revision 1. Washington, DC: NRC, Division of Waste Management. 1998.

———. “Issue Resolution Status Report, Key Technical Issue: Structural Deformation and Seismicity.” Revision 2. Washington, DC: NRC, Division of Waste Management. 1999a.

———. “Issue Resolution Status Report, Key Technical Issue: Igneous Activity.” Revision 2. Washington, DC: NRC, Division of Waste Management. 1999b.

———. NUREG–1804, "Yucca Mountain Review Plan—Draft Report for Comment." Revision 2. Washington, DC: NRC. March 2002.

Ortiz, T.S., R.L. Williams, F.B. Nimick, B.C. Whittet, and D.L. South. “A Three-Dimensional Model of Reference Thermal/Mechanical and Hydrological Stratigraphy at Yucca Mountain, Southern Nevada.” SAND84–1076. Albuquerque, New Mexico: Sandia National Laboratories. 1985.

Peacock, D.C.P. and D.J. Sanderson. “Geometry and Development of Relay Ramps in Normal Fault Systems.” *American Association of Petroleum Geologists Bulletin*. Vol. 78. pp. 147–165. 1994.

Perry, F.V. and B.M. Crowe. “Preclosure Volcanic Effects: Evaluations for a Potential Repository Site at Yucca Mountain, Nevada.” Los Alamos, New Mexico: Los Alamos National Laboratory. 1987.

Potter, C.J., W.C. Day, D.S. Sweetkind, and R.P. Dickerson. “Fault Styles and Strain Accommodation in the Tiva Canyon Tuff, Yucca Mountain, Nevada.” *EOS, Transactions, American Geophysical Union*. Vol. 77, Supplement. p. S265. 1996.

Saltus, R.W. and G.A. Thompson. “Why Is It Downhill from Tonopah to Las Vegas?: A Case for Mantle Plume Support of the High Northern Basin and Range.” *Tectonics.* Vol. 14. pp. 1,235–1,244. 1995.

Savage, J.C., M. Lisowski, J.L. Svarc, and W.K. Gross. “Strain Accumulation Across the Central Nevada Seismic Zone, 1973–1994.” *Journal of Geophysical Research.* Vol. 100. pp. 20,257–20,269. 1995.

Sawyer, D.R., R.J. Fleck, M.A. Lanphere, R.G. Warren, D.E. Broxton, and M.R. Hudson. “Episodic Caldera Volcanism in the Miocene Southwestern Nevada Volcanic Field: Revised Stratigraphic Framework, $^{40}Ar/^{39}Ar$ Geochronology, and Implications for Magmatism and Extension.” *Geological Society of America Bulletin*. Vol. 106. pp. 1,304–1,318. 1994.

Scott, R.B. "Tectonic Setting of the Yucca Mountain Region, Southwest Nevada." *Geological Society of America Memoir 176*. Boulder, Colorado: Geological Society of America. pp. 251–282. 1990.

Scott, R.B. and J. Bonk. "Preliminary Geologic Map of Yucca Mountain, Nye County, Nevada with Geologic Sections." U.S. Geological Survey Open-File Report 84-494. Scale 1:12,000. 1984.

Simonds, W.F., J.W. Whitney, K. Fox, A. Ramelli, J.C. Yount, M.D. Carr, C.D. Menges, R. Dickerson, and R.B. Scott. "Map of Fault Activity of the Yucca Mountain Area, Nye County, Nevada." U.S. Geological Survey Miscellaneous Investigations Series, Map I–2520. Scale 1:24,000. 1995.

Snow, J.K. and B.P. Wernicke. "Cenozoic Tectonism in the Central Basin and Range: Magnitude, Rate, and Distribution of Upper Crustal Strain." *American Journal of Science*. Vol. 300. pp. 659–719. 2000.

Spengler, R.W., C.A. Braun, L.G. Martin, and C.W. Weisenberg. "The Sundance Fault—A Newly Recognized Shear Zone at Yucca Mountain, Nevada." U.S. Geological Survey Open-File Report 94-49. 1994.

Stamatakos, J.A. and D.A. Ferrill. "Comment on Strike-Slip Fault System in Amargosa Valley and Yucca Mountain, Nevada by R.A. Schweickert and M.M. Lahren." *Tectonophysics.* Vol. 294. pp. 151–160. 1998.

Stepp, J.C., I. Wong, J. Whitney, R. Quittmeyer, N. Abrahamson, G. Toro, R. Youngs, K. Coppersmith, J. Savy, T. Sullivan, and Yucca Mountain Probabilistic Seismic Hazard Analysis Project Members. "Probabilistic Seismic Hazard Analyses for Ground Motions and Fault Displacements at Yucca Mountain, Nevada." *Earthquake Spectra*. Vol. 17. pp. 113–151. 2001.

Taylor, E.M. and H.E. Huckins. "Lithology, Fault Displacement, and Origin of Secondary Calcium Carbonate and Opaline Silica at Trenches 14 and 14d on the Bow Ridge Fault at Exile Hill, Nye County, Nevada." U.S. Geologic Survey Open-File Report 93-477. 1995.

Terzaghi, R. "Source of Error in Joint Surveys." *Geotechnique*. Vol. 15. p. 287. 1965.

Thatcher, W., G.R. Foulger, B.R. Julian, J. Svarc, E. Quilty, and G.W. Bawden. "Present-Day Deformation Across the Basin and Range Province, Western United States." *Science.* Vol. 283. pp. 1,714–1,718. 1999.

Throckmorton, C.K. and E.R. Verbeek. "Joint Networks in the Tiva Canyon and Topopah Spring Tuffs of the Paintbrush Group, Southwestern Nevada." U.S. Geological Survey Open-File Report 95-2. 1995.

Trudgill, B. and J. Cartwright. “Relay-Ramp Forms and Normal-Fault Linkages, Canyonlands National Park, Utah.” *Geological Society of America Bulletin*. Vol. 106. pp. 1,143–1,157. 1994.

U.S. Census Bureau. “1990 Census of Population and Housing, Population and Housing Unit Counts, Nevada.” Table 9. Population and Housing Units, 1970 to 1990; Area Measurements and Density: 1990. 1990 CHP–2–30. Washington, DC: U.S. Census Bureau. 1993.

———. “Land Area, Population and Density for States and Counties: 1990.” Summary Tape File 1C. Washington, DC: U.S. Census Bureau. 1996.

U.S. Geological Survey. “Seismotectonic Framework and Characterization of Faulting at Yucca Mountain, Nevada.” WBS 1.2.3.2.8.3.6. 1996.

White A.F., H.C. Claassen, and L.V. Benson. “The Effect of Dissolution of Volcanic Glass on the Water Chemistry in a Tufaceous Aquifer.” U.S. Geological Survey Open-File Report 1535–Q. 1980.

Yang, I.C., G.W. Rattray, and P. Yu. “Interpretation of Chemical and Isotopic Data from Boreholes in the Unsaturated Zone at Yucca Mountain, Nevada.” U.S. Geologic Survey Water Resources Investigations Report 96-4058. 1996.

Yang, I.C., P. Yu, G.W. Rattray, J.S. Ferarese, and R.N. Ryan. “Hydrochemical Investigations in Characterizing the Unsaturated Zone at Yucca Mountain, Nevada.” U.S. Geologic Survey Water Resources Investigations Report 98-4132. 1998.

2.1.2 Description of Structures, Systems, Components, Equipment, and Operational Process Activities

2.1.2.1 Description of Issue

This section on Description of Structures, Systems, Components, Equipment and Operational Process Activities addresses assessment of the DOE description of structures, systems, components, equipment, and operational process activities for the surface and subsurface facilities of the proposed geologic repository. 10 CFR 63.112 requires a license application for construction authorization of a geologic repository to include a preclosure safety analysis. A preclosure safety analysis is required to demonstrate the safety of the proposed design and operations in the geologic repository operations area with regard to the overall preclosure performance objectives through a systematic examination of the site information, the design, the potential hazards, initiating events and resulting event sequences, and potential radiological exposures to workers and the public. This analysis should lead to the identification of structures, systems, components important to safety, and safety measures that are relied on to limit or prevent the potential consequences of the hazards and event sequences identified. To conduct a meaningful preclosure safety analysis on the design and operations such that the needed structures, systems, components, and safety measure can be determined; the structures, systems, components, equipment, process activities, and sources of hazardous materials involved in the safety analysis need to be sufficiently described. The extent of description should be consistent with the level of the preclosure safety analysis performed.

Furthermore, 10 CFR 63.112(a) requires that, in the license application, the DOE preclosure safety analysis must include a general description of the structures, systems, components, equipment, and operational process activities at the geologic repository operations area. Also in 10 CFR 63.21, the regulatory requirement stipulates that a license application should include (i) information relative to materials of construction of the geologic repository operations area (including geologic media, general arrangement, and approximate dimensions) and codes and standards that DOE proposes to apply to the design and construction of the geologic repository operations area [10 CFR 62.21(c)(2)]; (ii) a description and discussion of the design of the various components of the geologic repository operations area and the engineered barrier subsystem (including dimensions, material properties, specifications, and analytical and design methods used) along with any applicable codes and standards [10 CFR 63.21(c)(3)(i)]; and (iii) a description (of the kind, amount, and specifications) of the radioactive material proposed to be received and possessed at the geologic repository operations area at the Yucca Mountain site [10 CFR 63.21(c)(4)].

2.1.2.2 Importance to Safety

A sufficient description of the structures, systems, components, equipment, operational process activities, and sources of hazardous materials consistent with the nature of the preclosure safety analysis is of paramount importance to ensure the success of the safety analysis. Without an adequate description in the license application, the outcome of the safety analysis is not likely to lead to an appropriate identification of the structures, systems, and components important to safety, and safety measures that are necessary to limit or prevent the potential dose consequences. As a result, reasonable assurance of the design and operations in the geologic repository operations area to meet the preclosure performance objectives may not be obtained.

2.1.2.3 Technical Basis

DOE has not yet finalized the design of structures, systems, components, equipment, and operational process activities in the geologic repository operations area. The DOE descriptions of these items are preliminary, and, therefore, the staff evaluation is preliminary.

Approximately 70,000 metric tons of high-level waste will be received, processed, and emplaced during the proposed operational period of 24 years (CRWMS M&O, 1999a). This high-level waste includes the spent nuclear fuel and the defense high-level waste. The geologic repository operations area may be conveniently categorized into surface and subsurface facilities. The surface facilities will be used to receive spent nuclear fuel and defense high-level waste shipments, temporarily store them, and prepare and package the wastes for underground emplacement (DOE, 1998). The surface facilities will house radiological protection, utilities, and ventilation for the underground facilities and also provide other supporting functions. The surface facilities consist of three primary functional areas: (i) the waste receiving and inspection area, where incoming trucks and rail cars arrive and are inspected; (ii) the surface portion of the waste operations area, which includes all buildings where radioactive material is handled for packaging and temporary storage; and (iii) the general support facilities, consisting of administrative buildings, security stations, and warehouses (DOE, 2001).

The restricted-access area for waste handling and packaging facilities will include buildings and equipment for receiving, packaging, and temporary storing of all incoming wastes. The surface plant also will include a waste treatment facility for processing all the radioactive wastes generated by on-site operations (e.g., protective clothing, decontamination fluids, and ventilation filters). Support facilities for the repository will include offices for administrative, management, and engineering staff; a firehouse; medical, training, and computer centers; a vehicle maintenance and repair shop; security buildings; a machine and sheet metal shop; and an electrical shop. Warehouses will be needed to store bulk materials, equipment, spare parts, and supplies.

Facilities for environmental measurements and instrument laboratories will also be required. Surface facilities to support the underground operations include staff changing rooms and showers, as well as space to store mining equipment and vehicles. Electric transmission lines will be extended to the repository facilities from existing local utility lines, and a new substation will be provided at the site. Utilities that support the repository will include an electric power building with emergency electrical generating equipment, steam-generating equipment, compressor and chiller systems, and cooling towers with water treatment equipment. A system for treating and distributing potable water and water for fire protection will also be required. New wells or storage tanks may be needed to supply the water required for construction and operation of the repository. Finally, stations for dispensing gasoline and diesel fuel will be required at the site. Various DOE reports provide further descriptions of the repository surface facilities (DOE, 1998, 2001; CRWMS M&O, 1999a).

The repository subsurface facilities consist of portals and access ramps, access mains, emplacement drifts, openings to support the subsurface ventilation, and openings to support monitoring and performance confirmation testing (CRWMS M&O, 1998). The waste packages will be emplaced in the repository siting volume (DOE, 1998). The repository host horizon is located above the water table in the unsaturated zone. The repository emplacement drifts and perimeter main drifts will be located entirely within this siting volume. The physical location and general arrangement of the subsurface facility in the unsaturated zone above the water table take advantage of the mountain's natural geologic barriers and other attributes as part of the overall waste containment strategy. Another design consideration was locating the emplacement drifts away from major faults. A detailed description of the repository subsurface facilities is available in various reports (DOE, 1998, 2001; CRWMS M&O, 2000a,b).

The portal and access ramps (north portal, south portal, north ramp, and south ramp) of the existing exploratory studies facility will be integrated into the proposed repository and would connect the surface and subsurface facilities through the access mains. The access mains are a network of tunnels that define the perimeter of, and provide access to, the proposed emplacement area. The access mains comprise the north-south trending east main and west main, which are interconnected through other shorter tunnels, such as the north and south mains, and to the surface facility through the access ramps (CRWMS M&O, 2000b). The access mains have a nominal diameter of 7.62 m [25 ft] and are provided with rail lines to support the transportation of the waste packages to and from the emplacement area. The east and west mains will also serve to conduct intake ventilation air to the emplacement area (CRWMS M&O, 2000c). The emplacement drifts will be an array of horizontal tunnels trending approximately east-northeast-west-southwest (252 azimuth) between the east and west mains. Each drift will have a diameter of 5.5 m [18.5 ft] and will be separated from the adjacent drifts by a center-to-center distance of 81 m [265.7 ft]. The transition from the east and west mains to the emplacement drifts (which are nearly perpendicular to the mains) will be provided through the emplacement-drift turnouts (CRWMS M&O, 2000a). A pair of isolation doors located near the emplacement drift and access main ends of each turnout will help control airflow into the emplacement drifts and to protect the access mains from radiation that emanates from the waste packages in the emplacement drifts. The ground-support system for the emplacement drifts will consist of steel sets and wire mesh, with occasional rock bolts installed in the roof area if considered necessary during construction. The ground support will be of carbon-steel material and will be designed for an operational life of up to 175 years, with possible extension to 300 years (CRWMS M&O, 2000a,d).

Other openings that constitute the underground facility include the north-south trending exhaust main located below the emplacement drifts; the ventilation raises (i.e., shafts excavated from the floor of the emplacement drifts to the roof of the exhaust main), and the intake and exhaust shafts and other drifts within the emplacement block that will be used for various purposes other than waste emplacement. The ground-support system for the nonemplacement openings (including the access mains) will initially consist of pattern rock bolts and welded wire fabric and, where necessary, shotcrete or steel sets. A final ground support consisting of a cast-in-place concrete lining will be installed to provide long-term support for such openings during the preclosure period.

Contingent on NRC granting a construction authorization, construction will begin on the initial portions of the surface and subsurface facilities that include additions to the existing surface facilities; retrofitting the north and south portals, north and south ramps, and east main drift; muck handling excavation; and installation of the subsurface ventilation systems. After this initial construction, underground openings will be developed concurrently with waste emplacement operations (DOE, 1998; CRWMS M&O, 1999b). Development of underground openings will take place without interference with waste emplacement operations. The repository openings are constructed to serve a variety of functions. Main access (shafts and ramps) provides facilities for ventilating the subsurface, emplacing waste, removing excavated material, performing maintenance, and transporting staff and materials. A conveyor belt will transport excavated rock (muck) from the subsurface to the surface. A tunnel boring machine will be used for most underground excavations. Mechanical methods, such as road-header machines or the drill-and-blast excavation method, may be used where tunnel boring machine operation is not feasible. Other construction-related activities will include installation of ground supports and transportation of excavated rock from the subsurface to the surface. A general description of the construction of the repository surface and subsurface facilities has been provided in various reports (DOE, 1998, 2001; CRWMS M&O, 1999a).

As discussed earlier, the repository will have the capability to receive and emplace approximately 70,000 metric tons (77,162 tons) of uranium waste. The waste will arrive at the repository by rail or truck and be received at the radiologically controlled area 24 hours a day. The rail shipment will arrive at the site as a unit train consisting of one or two locomotives, three to five rail cars carrying one cask per rail car, and buffer rail cars between rail cars with casks. The truck shipment will arrive in legal-weight trucks. DOE developed a schedule of receipt based on a reference design (CRWMS M&O, 1999a). The reference design is based on an approximated annual receipt rate of 3,000 metric tons (3.307 tons) of uranium waste for an operational period of 24 years. Annual rate of receipt and handling of casks, canisters, fuel assemblies, and disposal canisters in the facility will vary. In the preclosure safety analysis, however, it is important to know the maximum handling rate because 10 CFR 63.21(c)(5) requires that the preclosure safety analysis is carried out at maximum capacity and rate of receipt of waste.

The waste handling and emplacement operations have been discussed in DOE (1998). North portal surface facilities constitute the primary surface facilities to receive spent nuclear fuel and high-level waste shipments and prepare and package the wastes for underground emplacement (DOE, 1998). All waste shipments will be received at a security station where they will be inspected. Casks mounted on a carrier will be transported within the controlled area by a site prime mover. Waste shipments will be transported to the carrier preparation building or to a parking area to wait for a bay in the carrier preparation building. The prepared carrier will be transported from the carrier preparation building to the waste handling building, where the shipping casks are sent to one of two waste handling systems: a wet assembly transfer system that includes a pool or a dry canister transfer system.

The wet assembly transfer system will receive casks containing individual fuel assemblies that have either been loaded into the cask directly or are contained in a nondisposable canister that must be removed from the cask and opened before the assemblies can be removed. Some nondisposable canisters may have been welded closed and will need to be cut open. The assemblies will be removed from the casks or canisters in a pool environment, after which they will be transferred to and dried in a fuel assembly transfer cell before being loaded into a disposal container (DOE, 1998). The dry canister transfer system will receive spent nuclear fuel, vitrified defense high-level waste, and special defense waste forms, including immobilized plutonium, in canisters designed for direct insertion into disposal containers.

The disposal canister handling system will receive loaded containers from both wet assembly transfer and dry canister transfer systems. After the disposal canister has been loaded, sealed, and tested, it is referred to as a waste package. The waste packages will be placed in the horizontal position and loaded into a subsurface transporter, which takes them to an emplacement drift. The subsurface transporter is a shielded cask mounted on a rail car. A locomotive will be coupled to each end of the transporter at the waste handling building loading facility. The two locomotives will move the transporter into and down the north ramp and into the east or west drift. At the selected emplacement drift, one locomotive will be uncoupled. The remaining locomotive will push the transporter against the transfer dock at the emplacement drift entrance. After the waste package transporter is positioned at the transfer dock in front of the emplacement drift isolation door and the drift isolation door is opened, the transporter door will be opened and rail continuity with the emplacement drift track will be established. The transporter is equipped with a self-contained mechanism that will push the rail car through the emplacement drift door and position it for unloading. A self-propelled, remotely operated emplacement gantry, which is stationed in the emplacement drift during active emplacement operations, will move into position over the rail car. The gantry will then engage the waste package and lift it from the rail car by the skirt flanges on both ends. The emplacement gantry will lift the waste package clear of the rail car and shadow shield and carry it through the emplacement drift to its preselected emplacement location. The gantry will then lower the waste package onto the v-shaped steel supports, disengage from the waste package, and return to a position near the emplacement drift door. If the waste package has to be moved during or after emplacement, it will be removed from the emplacement drift by following the emplacement operations in reverse order.

The staff review of the description of structures, systems, components, equipment, and operational process activities is currently ongoing. This review is in coordination with the review of preclosure safety analysis. The review will focus on the following areas:

- Descriptions of location of surface facilities and their functions including structures, systems, components, and equipment

- Descriptions of and design details for structures, systems, components, equipment, and utility systems of surface facilities

- Descriptions of and design details for structures, systems, components, equipment, and utility systems of the subsurface facility

- Description of high-level waste characteristics

- Descriptions and design details of engineered barrier system components (e.g., waste package, drip shield, and backfill, if any)

- Description of geologic repository operations area processes activities and procedures including human interactions and interfaces and interactions between structures, systems, and components.

2.1.2.4 Status and Path Forward

As discussed earlier, to conduct a meaningful preclosure safety analysis on the design and operations to determine the structures, systems, and components important to safety and the safety measures, the structures, systems, components, equipment, process activities, and sources of hazardous materials involved in the safety analysis need to be sufficiently described. The extent of description should be consistent with the level of the preclosure safety analysis performed. Consequently, the adequacy of this subsection has to be evaluated in conjunction with other subsections relevant to the preclosure safety analysis including repository design. The review and evaluation activities on the description of structures, systems, components, equipment, and operational process activities will continue as the DOE design and preclosure safety analysis progress.

2.1.2.5 References

CRWMS M&O. "Controlled Design Assumptions Document." B00000000–01717–4600–00032. Revision 5. Las Vegas, Nevada: CRWMS M&O. 1998.

———. "Repository Surface Design Engineering Files Report." BCB000000–01717–5705–0009. Revision 03. Las Vegas, Nevada: CRWMS M&O. 1999a.

———. "Monitored Geologic Repository Internal Hazards Analysis." ANL–MGR–SE–000003. Revision 00. Las Vegas, Nevada: CRWMS M&O. 1999b.

———. "Subsurface Facility System Description Document." SDD–SF–SSE–000001. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000a.

———. "Waste Emplacement/Retrieval System Description Document." SDD–WES–SE–000001. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000b.

———. "Subsurface Ventilation System Description Document." SDD–SVS–SE–000001. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000c.

———. "Ground Control System Description Document." SDD–GCS–SE–000001. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000d.

DOE. “Viability Assessment of a Repository at Yucca Mountain. Vol. 2: Preliminary Design Concept for the Repository and Waste Package.” DOE/RW–0508/V2. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 1998.

———. “Yucca Mountain Science and Engineering Report Technical Information Site Recommendation Consideration.” DOE/RW–0539. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 2001.

2.1.3 Identification of Hazards and Initiating Events

2.1.3.1 Description of Issue

DOE, as a part of its license application for the proposed geologic repository at Yucca Mountain, must present a safety analysis of the repository operations area for the preclosure period. This analysis is necessary to demonstrate compliance with the preclosure performance objectives of 10 CFR 63.111 that meet the requirements specified in 10 CFR 63.112. A preclosure safety analysis requires a systematic examination of the site; design; potential hazards, initiating events, and event sequences; and radiological dose consequences to the public and workers. This section deals with identification of hazards and initiating events for the preclosure safety analysis. Both natural hazards and human-induced initiating events in addition to operational hazards may lead to an event sequence with the potential for radiological release.

DOE developed a generic list of natural hazards and initiating events that need to be considered for potential radiological release from the proposed repository during the preclosure period (CRWMS M&O, 1999a,b; DOE, 2001a). Additionally, DOE developed a preliminary list of operational hazards associated with the preclosure operations (CRWMS M&O, 1999c; DOE, 2001a). These generic lists serve as the starting point to develop a comprehensive list of site-specific hazards that have a potential to initiate event sequences with radiological consequences. The NRC and CNWRA staffs have not completed reviewing the generic lists of hazards given in these and other associated documents for completeness and appropriateness for the proposed repository. The staff will be reviewing the lists according to NRC and other guidances for other nuclear-related facilities.

This section presents an initial review of the hazards and initiating events listed in the DOE documents. In addition to CRWMS M&O (1999a,b,c) and DOE (2001a), parts of additional documents were reviewed to the extent that they contain data, analyses, or both to support the identification of hazards and initiating events.

2.1.3.2 Importance to Safety

One aspect of a risk-informed NRC review is to determine how the issue of identification of hazards and initiating events is related to that portion of the DOE repository safety strategy addressing compliance with performance objectives during the preclosure period. Identification of hazards and initiating events is critical for demonstrating compliance with the preclosure performance objectives during operations, as identified in 10 CFR 63.21(c)(5).

2.1.3.3 Technical Basis

A review of the DOE identification of hazards and initiating events during the preclosure period is provided in the following subsections. The review is organized according to the five acceptance criteria consistent with the associated review methods and acceptance criteria in NRC (2002). The acceptance criteria are based on meeting the requirements of 10 CFR 63.112(b) and (d), relating to identification of hazards and initiating events.

DOE developed a preliminary list of operational hazards and initiating events that have the potential for a radiological release during the preclosure period (CRWMS M&O, 1999a) based on the facility design and operations and the functions of the structures, systems, and components described in several system description documents. The preclosure hazards and initiating events are associated with receiving, preparing, packaging, transporting, and emplacement operations at the surface and subsurface facility of the proposed repository (DOE, 2001a). In the operational hazard analysis, DOE identifies the operational hazards and initiating events by applying a checklist of generic events (e.g., collision/crushing, chemical contamination/internal flooding, explosion/implosion, fire/thermal, and radiation/fissile materials) to the functional areas within the proposed repository. DOE divided the surface and subsurface facilities in the proposed geologic repository operations area into nine functional areas defined by specific function, physical boundary, or both (CRWMS M&O, 1999a). A preliminary review of operational hazard analysis suggests that the DOE identification of hazards is incomplete. For example, DOE does not address reliability of human actions in the preclosure operations as a potential hazard. In addition, DOE does not consider the reliability of the hardware and software used in remote operations involved in preclosure operations in some functional areas.

Status for the DOE identification of operational hazards and initiating events from surface and subsurface operations in each of the functional areas is compiled in Table 2.1.3-1, including those hazard categories not considered or addressed by DOE. The table also includes natural and human-induced hazards that may become potential initiating events during facility operations. DOE stated it plans to design the facility to withstand initiating events resulting from such hazards and, therefore, eliminated the impact of natural and human-induced hazards on facility operations from further consideration in the preclosure safety analysis (CRWMS M&O, 1999b).

In the preliminary natural and human-induced hazards analysis, DOE generated a potential external hazards list from a generic checklist of 53 human-induced and natural phenomena (CRWMS M&O, 1999b; DOE, 2001a). The events from a generic checklist were screened for potential design basis events within a 100-year preclosure period on the basis of applicability to the proposed repository. This screening was accomplished by a five-step process, as described next. DOE stated the structures, systems, and components important to safety will be designed to withstand natural and human-induced hazards that can become potential initiating events. The complete list of natural and human-induced hazards considered by DOE is shown in Tables 2.1.3-2 and 2.1.3-3.

Table 2.1.3-1. Status of DOE Operational Hazard Analysis (CRWMS M&O, 1999a)			
No.	**Functional Areas**	**Generic Events**	**DOE Preliminary Events**
1	Waste Receipt and Carrier/Cask Transport	Collision/Crushing	Cask collision, railcar derailment, overturning of truck trailer involving cask
		Chemical Contamination/Internal Flooding	Not identified
		Explosion/Implosion	Not identified
		Fire/Thermal	Diesel fuel fire
		Radiation/Fissile Materials	Radiation exposure to facility worker Criticality associated with cask collision, railcar derailment, overturned truck trailer and rearrangement of cask internals
		Human Reliability	Not addressed
		Natural and Human-Induced Events	Structures, systems, and components designed to withstand events
2	Carrier/Cask Preparation	Collision/Crushing	Cask collision, handling equipment drop on cask
		Chemical Contamination/Internal Flooding	Not identified
		Explosion/Implosion	Not identified
		Fire/Thermal	Diesel fuel fire
		Radiation/Fissile Materials	Radiation exposure to facility worker Criticality associated with cask collision, rearrangement of cask internals
		Human Reliability	Not addressed
		Natural and Human-Induced Events	Structures, systems, and components designed to withstand events

Table 2.1.3-1. Status of DOE Operational Hazard Analysis (CRWMS M&O, 1999a) (continued)			
No.	**Functional Areas**	**Generic Events**	**DOE Preliminary Events**
3	Carrier Bay	Chemical Contamination/Internal Flooding	Not identified
		Explosion/Implosion	Not identified
		Fire/Thermal	Diesel fuel fire
		Radiation/Fissile Materials	Radiation exposure to facility worker Criticality associated with cask collision/drop, rearrangement of cask internals
		Human Reliability	Not addressed
		Natural and Human-Induced Events	Structures, systems, and components designed to withstand events
4	Waste Handling–Canister Transfer	Collision/Crushing	Cask: slap down, handling equipment drop on cask Canister: drop, slap down, collision, canister drop on to disposal container, canister drop on sharp object, canister drop onto another canister in staging rack, shield door close on cask, shield door close on disposal container: slap down, and collision
		Chemical Contamination/Internal Flooding	Not identified
		Explosion/Implosion	Not identified
		Fire/Thermal	Not identified
		Radiation/Fissile Materials	Exposure to facility worker Criticality associated with small canister staging rack, collision/drop of cask/canister, rearrangement of container internals
		Human Reliability	Not addressed
		Remote Operations/Software-Hardware Reliability	Not addressed

Table 2.1.3-1. Status of DOE Operational Hazard Analysis (CRWMS M&O, 1999a) (continued)			
No.	**Functional Areas**	**Generic Events**	**DOE Preliminary Events**
		Natural and Human-Induced events	Structures, systems, and components designed to withstand events
5	Waste Handling–Assembly Transfer	Collision/Crushing	Cask: drop, slap down, collision, handling equipment drop on cask Spent nuclear fuel assembly: drop on floor, slap down, collision, spent nuclear fuel assembly staging rack, drop onto assembly dryer, and drop onto disposal container Loaded spent nuclear fuel assembly basket: drop onto spent nuclear fuel assembly staging rack, drop onto assembly cell floor, drop onto assembly dryer, collision, uncontrolled descent of incline basket transfer cart
		Chemical Contamination/Internal Flooding	Flood due to uncontrolled pool water drain-down/fill
		Explosion/Implosion	Not identified
		Fire/Thermal	Spent nuclear fuel overheating resulting in excessive clad temperature and zircalloy cladding fire in assembly transfer basket or dryer and in pool because of loss of pool water
		Radiation/Fissile Materials	Uncontrolled pool water drain-down/fill resulting in flooding and radioactive contamination of adjoining Waste Handling Building areas, increased radiation levels in assembly transfer area, potential uncovering of fuel assemblies, exposure of facility worker Criticality associated with cask collision/drop, rearrangement of cask internals, spent nuclear fuel assembly staging rack, misload of assembly dryer, misload of disposal container
		Remote Operations/Software-Hardware Reliability	Not addressed

Table 2.1.3-1. Status of DOE Operational Hazard Analysis (CRWMS M&O, 1999a) (continued)

No.	Functional Areas	Generic Events	DOE Preliminary Events
		Human Reliability	Not addressed
		Natural and Human-Induced Events	Structures, systems, and components designed to withstand events
6	Waste Handling–Disposal Container and Waste Package Remediation	Collision/Crushing	Waste package: drop, slap down, drop onto sharp object, collision, handling equipment drop Disposal container: drop, slap down, drop onto sharp object, collision, handling equipment drop
		Chemical Contamination/Internal Flooding	Not identified
		Explosion/Implosion	Not identified
		Fire/Thermal	Fuel damage by burn-through during welding process, spent nuclear fuel overheating in disposal container resulting in excessive clad temperature and possible zircalloy cladding fire
		Radiation/Fissile Materials	Exposure of facility worker Criticality associated with cask collision/drop, rearrangement of cask internals, spent nuclear fuel assembly staging rack, misload of assembly dryer, misload of disposal container
		Remote Operations/Software-Hardware Reliability	Not addressed
		Human Reliability	Not addressed
		Natural and Human-Induced Events	Structures, systems, and components designed to withstand events

Table 2.1.3-1. Status of DOE Operational Hazard Analysis (CRWMS M&O, 1999a) (continued)			
No.	**Functional Areas**	**Generic Events**	**DOE Preliminary Events**
7	Subsurface Transport, Emplacement, and Monitoring	Collision/Crushing	Transporter: derailment outdoors, derailment in ramp or main drift, collision with stationary or moving equipment, runaway, waste package reusable rail car rolls out, rockfall Emplacement gantry: derailment Waste package: drop from emplacement gantry, rockfall, steel set drop, waste package/emplacement gantry collision with equipment or another waste package, failure of isolation air lock due to rockfall
		Chemical Contamination/Internal Flooding	Flooding from water pipe break
		Explosion/Implosion	Not identified
		Fire/Thermal	Fire associated with waste package transporter/locomotive or development equipment
		Radiation/Fissile Materials	Exposure of facility worker, early or juvenile failure, and resultant release of radioactive waste Criticality associated with collision/drop of waste package and rearrangement of waste package internals
		Human Reliability	Not addressed
		Remote Operations/Software-Hardware Reliability	Not addressed
		Natural and Human-Induced Events	Structures, systems, and components designed to withstand events
8	Waste Treatment (Liquid Low Level)	Collision/Crushing	Handling equipment drop on liquid low-level waste
		Chemical Contamination/Internal Flooding	Uncontrolled release of liquid low-level waste
		Explosion/Implosion	Not identified

Table 2.1.3-1. Status of DOE Operational Hazard Analysis (CRWMS M&O, 1999a) (continued)			
No.	**Functional Areas**	**Generic Events**	**DOE Preliminary Events**
		Fire/Thermal	Not identified
		Radiation/Fissile Materials	Operator exposure to radioactive material
		Human Reliability	Not addressed
		Natural and Human-Induced Events	Structures, systems, and components designed to withstand events
9	Waste Treatment (Solid Low Level)	Collision/Crushing	Solid low-level waste drop, handling equipment drop on solid low-level waste
		Chemical Contamination/Internal Flooding	Not identified
		Explosion/Implosion	Not identified
		Fire/Thermal	Fire involving combustible low-level waste
		Radiation/Fissile Materials	Operator exposure to radioactive material
		Human Reliability	Not considered
		Natural and Human-Induced Events	Structures, systems, and components designed to withstand events

Table 2.1.3-2. List of Natural Hazards with DOE Assessment (after CRWMS M&O, 1999a; DOE, 2001a)

No.	Hazard	Hazard Definition	DOE Assessment
1	Avalanche	A large mass of snow, ice, soil, or rock or mixtures of these materials, falling, sliding, or flowing under gravity	Not applicable to the hazards list • High mountain ranges do not exist at Yucca Mountain
2	Coastal Erosion	Wearing away of soil and rock by waves and tidal action	Not applicable to the hazards list • Coastline does not exist at Yucca Mountain
3	Dam Failure	Failure of a large man-made barrier that creates and restrains a large body of water	Not applicable to the hazards list • No dam of sufficient size exits in proximity to Yucca Mountain
4	Debris Avalanche	Sudden and rapid movement of debris down steep slopes resulting from intensive rainfall	Applicable to the hazards list • Potential exists • Rate of process is sufficient to affect 100-year preclosure period • Consequence of process is significant • Annual event frequency $\geq 10^{-6}$ • Not included in another analysis
5	Denudation	Sum of processes that result in wearing away or progressive lowering of Earth's surface by weathering, mass wasting, and transportation	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is low enough for 100-year preclosure period
6	Dissolution	Processes of chemical weathering by which mineral and rock material passes into solution	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficiently high to affect 100-year preclosure period and may create rockfall • Consequence is indeterminant; assumed to be equivalent to significant enough to affect 100-year preclosure period • Annual event frequency is indeterminant; assumed $\geq 10^{-6}$ • Key Block Analysis Report will address rockfall issue

Table 2.1.3-2. List of Natural Hazards with DOE Assessment (after CRWMS M&O, 1999a; DOE, 2001a) (continued)

No.	Hazard	Hazard Definition	DOE Assessment
7	Eperogenic Displacement	Geomorphic processes of uplift and subsidence that produced broader features of continents and oceans	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is not sufficient to pose credible hazard during 100-year preclosure period
8	Erosion	Slow wearing of soil and rock by weathering, mass wasting, and action of streams	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process not sufficient to pose credible hazard during 100-year preclosure period
9	Extreme Weather Fluctuations	Various types of weather fluctuations that pose unusual design challenges	Not applicable to the hazards list • No potential exists at Yucca Mountain
10	Extreme Wind	Fastest mile of wind with 100-year return period	Applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficient during 100-year preclosure period • Potential consequence is indeterminant; assumed equivalent to true • Annual event frequency $\geq 10^{-6}$ • Not included in another analysis
11	Flood (Storm, River Diversion)	Area covered with water from storm or river diversion caused by inadequate drainage	Applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficiently high during 100-year preclosure period • Consequences of process are sufficiently high • Annual event frequency $\geq 10^{-6}$ • Not included in another analysis
12	Fungus, Bacteria, and Algae	General class of microorganisms that may be present in subsurface environment	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficiently high during 100-year preclosure period • Consequence of process not significant to affect 100-year preclosure period
13	Glacial Erosion	Lowering of Earth's surface due to grinding and scouring by glacier ice armed with rock fragments	Not applicable to the hazards list • No potential exists at Yucca Mountain for a glacier
14	Glaciation	Formation, movement, and recession of glaciers or ice sheets	Not applicable to the hazards list • No potential exists at Yucca Mountain for a glacier and associated climate change

Table 2.1.3-2. List of Natural Hazards with DOE Assessment (after CRWMS M&O, 1999a; DOE, 2001a) (continued)

No.	Hazard	Hazard Definition	DOE Assessment
15	High Lake Level	Potential overflow or flooding of lake	Not applicable to the hazards list • No potential exits at Yucca Mountain because there is no lake nearby
16	High Tide	High tide in water connected with ocean having potential for flooding inland areas	Not applicable to the hazards list • No potential exits at Yucca Mountain because there is no ocean or coastal area
17	High River Stage	Potential flooding of river or natural permanent or seasonal surface stream with considerable volume	Not applicable to the hazards list • No potential exists at Yucca Mountain because there is no river nearby
18	Hurricane	Intense cyclone that forms over tropical oceans	Not applicable to the hazards list • No potential exists at Yucca Mountain because it is located approximately 360 km [225 mi] inland from nearest ocean, northeast of Santa Monica Bay near Los Angeles; based on American National Standards Institute/American Nuclear Society 2.8–92 (1992)*, site needs to be within 160 to 320 km [100 to 200 mi] from ocean for hurricane to be potential natural hazard
19	Landslides	Wide variety of mass movement of land forms and processes involving downslope transport with gravitational influence	Applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficiently high to affect 100-year preclosure period • Consequence is indeterminant; assumed equivalent to true • Annual event frequency $\geq 10^{-6}$ • Not part of another analysis
20	Lightning	Flashing of light produced by discharge of atmospheric electricity between charged cloud and Earth	Applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficiently high during 100-year preclosure period • Consequence is indeterminant; assumed equivalent to true • Annual event frequency $\geq 10^{-6}$ • Not part of another analysis
21	Low Lake Level	Low level of lake water used for cooling	Not applicable to the hazards list • No potential exists at Yucca Mountain because there is no lake nearby

Table 2.1.3-2. List of Natural Hazards with DOE Assessment (after CRWMS M&O, 1999a; DOE, 2001a) (continued)

No.	Hazard	Hazard Definition	DOE Assessment
22	Low River Level	Low level of river water used for cooling	Not applicable to the hazards list • No potential exists at Yucca Mountain because there is no river nearby
23	Meteorite Impact	Impact of meteoroid reaching Earth's surface without completely vaporizing	Not applicable to the hazards list. • Potential exists at Yucca Mountain • Rate of process is sufficiently high during 100-year preclosure period • Consequence is indeterminant; assumed equivalent to true • Annual event frequency $\leq 10^{-6}$
24	Orogenic Diastrophism	Movement of Earth's crust produced by tectonic processes where structures within fold-belt mountain areas formed, including thrusting, folding, and faulting	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is too low to affect 100-year preclosure period
25	Rainstorm	Storm that produces 100-year or greater maximum rainfall rate occurring for one day	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficiently high during 100-year preclosure period • Consequence is indeterminant; assumed significant • Annual event frequency $\geq 10^{-6}$ • Bounded by debris avalanche, flooding, and landslide events for which this is initiator
26	Range Fire	Combustion of natural vegetation external to repository that propagates to combustible materials within operations area	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficiently high during 100-year operational period • Consequences are significant • Annual event frequency $\geq 10^{-6}$ • Will be addressed in fire hazard analyses

Table 2.1.3-2. List of Natural Hazards with DOE Assessment (after CRWMS M&O, 1999a; DOE, 2001a) (continued)

No.	Hazard	Hazard Definition	DOE Assessment
27	Sandstorm	Extreme wind capable of transporting sand and other unconsolidated surficial materials	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficient during 100-year preclosure period • Consequence is indeterminant; assumed significant • Annual event frequency $\geq 10^{-6}$ • Bounded by extreme wind and tornadoes events • Potential filter clogging is screened out from further consideration because of capability for orderly facility shutdown through technical specification—a to-be-verified item
28	Sedimentation	Process of forming or accumulating sediment in layers	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is too low in 100-year preclosure period
29	Seiche	Free or standing wave oscillation of water surface in enclosed or semienclosed basin	Not applicable to the hazards list • No potential exists at Yucca Mountain because there is no large body of water nearby
30	Seismic Activity (Uplifting)	Structurally high area in the crust, produced by positive movements over long time periods resulting in faults giving rise to upthrust of rocks	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is too slow in 100-year preclosure period
31	Seismic Activity (Earthquake)	Earthquakes including those artificially induced	Applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficiently high during 100-year preclosure period • Consequence is significant • Mean annual probabilities of Frequency Categories 1 and 2 design-basis ground motions are 1×10^{-3} and 1×10^{-4}; structures, systems, and components important to safety will be designed to withstand design-basis earthquake (Frequency Categories 1 and 2), as appropriate • Not bounded by another analysis

Table 2.1.3-2. List of Natural Hazards with DOE Assessment (after CRWMS M&O, 1999a; DOE, 2001a) (continued)

No.	Hazard	Hazard Definition	DOE Assessment
32	Seismic Activity (Surface Fault Displacement)	Fracture or zone of fractures along which there is potential for displacement of sides relative to each other parallel to fracture	Applicable to the hazards list • Potential exits at Yucca Mountain • Rate of process is sufficiently high during 100-year preclosure period • Mean annual probabilities of Frequency Categories 1 and 2 design-basis ground motions are 1×10^{-3} and 1×10^{-4}; structures, systems, and components important to safety will be designed to withstand fault displacements from design-basis earthquake (Frequency Categories 1 and 2), as appropriate • Not bounded by another analysis
33	Seismic Activity (Subsurface Fault Displacement)	Fracture or zone of fractures along which there is potential for displacement of sides relative to each other parallel to fracture	Applicable to the hazards list • Potential exits at Yucca Mountain • Rate of process is sufficiently high during 100-year preclosure period • Mean annual probabilities of Frequency Categories 1 and 2 design-basis ground motions are 1×10^{-3} and 1×10^{-4}; structures, systems, and components important to safety will be designed to withstand fault displacements from design-basis earthquake (Frequency Categories 1 and 2), as appropriate • Not bounded by another analysis
34	Static Fracturing	Break in rock due to mechanical failure by stress	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficiently high to affect 100-year preclosure period • Consequence is indeterminant; assumed significant • Annual event frequency $\geq 10^{-6}$ • Will be addressed in Key Block Analysis Report
35	Stream Erosion	Progressive removal of bedrock, overburden, soil, or other exposed matters from stream channel surface	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is too slow to affect 100-year preclosure period

Table 2.1.3-2. List of Natural Hazards with DOE Assessment (after CRWMS M&O, 1999a; DOE, 2001a) (continued)

No.	Hazard	Hazard Definition	DOE Assessment
36	Subsidence	Sudden sinking or gradual downward settling of Earth’s surface with little or no horizontal motion	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficiently high to affect 100-year preclosure period • Consequence is indeterminant; assumed significant • Annual event frequency $\geq 10^{-6}$ • Screened out because subsurface fault displacement will be only natural phenomenon that would result in collapse of underground excavations leading to subsidence; emplacement levels would be at least 200 m [656 ft] below the directly overlying ground surface; emplacement drifts will be supported by rock bolts, steel mesh, and steel sets; no surface-handling facilities will be directly over emplacement drifts
37	Tornado	Small cyclone generally less than 500 m [1,650 ft] in diameter with extremely strong winds	Applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficient to affect 100-year preclosure period • Consequence is indeterminant; hence assumed significant • Annual event frequency $\geq 10^{-6}$ • Not bounded by another analysis
38	Tsunami	Gravitational sea wave produced by large-scale, short-duration disturbance on ocean floor	Not applicable to the hazards list • No potential exists at Yucca Mountain because there is no coastal region
39	Undetected Geologic Features	Geologic features of concern to the 100-year preclosure period include natural events such as faults and volcanoes	Not applicable to the hazards list • No potential exists at Yucca Mountain; site characterization provided sufficient assurance that these types of activities would have been detected
40	Undetected Geologic Processes	Geologic processes of concern to the 100-year preclosure period include events such as erosion, tectonic, and seismic processes	Not applicable to the hazards list • No potential exists at Yucca Mountain; site characterization provided sufficient assurance that these types of activities would have been detected

Table 2.1.3-2. List of Natural Hazards with DOE Assessment (after CRWMS M&O, 1999a; DOE, 2001a) (continued)

No.	Hazard	Hazard Definition	DOE Assessment
41	Volcanic Eruption	Magma and associated gases rise into the crust and are extruded onto Earth's surface and into atmosphere	Not applicable to the hazards list • No potential exists at Yucca Mountain because there is no potential for volcanic center at the site
42	Volcanism (Intrusive Magmatic Activity)	Development and subsurface movement of magma and mobile rock materials	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficiently high to affect 100-year preclosure period • Consequence is indeterminant; assumed significant • Annual event frequency $\leq 10^{-6}$
43	Volcanism (Ash Flow, Extrusive Magmatic Activity)	Highly heated mixture of volcanic gases, magma, mobile rock material, and ash traveling down the flank of a volcano or along ground surface	Not applicable to the hazards list • No potential exists at Yucca Mountain for silicic volcanism
44	Volcanism (Ash Fall)	Airborne volcanic ash falling from eruption cloud	Not applicable to the hazards list • Potential exists for ash fall within 100-year preclosure period at Yucca Mountain • Rate of process is indeterminant; hence assumed to be significant • Consequence not significant to affect 100-year preclosure period because —worst-case ash fall depth is 3 cm [1.2 in] —worst-case live load on flat roof is 868.5 Pa [18.14 lb/ft^2], which is less than minimum 1997 Uniform Building Code requirements • Filter clogging due to ash fall is bounded by filter clogging by sandstorm event
45	Waves	Oscillatory movement of water manifested by alternate rise and fall of water surface	Not applicable to the hazards list • No potential exists at Yucca Mountain because there is no large body of water nearby

*American National Standards Institute/American Nuclear Society. "Determining Design Basis Flooding at Power Reactor Sites, An American National Standard." ANSI/ANS 2.8-92. La Grange, Illinois: American Nuclear Society. 1992.

Table 2.1.3-3. List of Human-Induced Events with DOE Assessment (after CRWMS M&O, 1999a; DOE, 2001a)			
No.	**Hazard**	**Hazard Definition**	**DOE Assessment**
1	Aircraft Crash	Accidental impact of aircraft on the site facilities	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process (i.e., impact of the crash) is immediate • Consequence is significant • Event frequency $\leq 10^{-6}$ per year
2	Inadvertent Future Intrusions (Human-Induced)	Human-induced inadvertent future intrusions with regard to 100-year preclosure period involve undetected surface access into proposed repository facilities	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficient to affect 100-year preclosure period • Consequence is indeterminant; hence assumed significant • Annual event frequency is indeterminant; hence assumed significant • Will be considered in future safeguards and security analyses—a to-be-verified item
3	Intentional Future Intrusions (Human-Induced)	Human-induced intentional future intrusions with regard to 100-year preclosure period involve undetected surface access, sabotage, or both to the proposed repository facilities	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficient to affect 100-year preclosure period • Consequence is indeterminant; hence assumed significant • Annual event frequency is indeterminant, hence assumed significant • Will be considered in future safeguards and security analyses—a to-be-verified item
4	Industrial Activity-Induced Accidents	Accidents resulting from industrial or transportation activities unrelated to proposed repository	Applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficient to affect 100-year preclosure period • Consequence is indeterminant; hence assumed significant • Annual event frequency is indeterminant at this time; hence assumed significant • Not bounded by another analysis

Table 2.1.3-3. List of Human-Induced Events with DOE Assessment (after CRWMS M&O, 1999a; DOE, 2001a) (continued)

No.	Hazard	Hazard Definition	DOE Assessment
5	Loss of Off-site/On-site Power	Loss of electric power either generated or controlled by persons outside repository system or loss of power within repository	Applicable to the hazards list • Potential exists at Yucca Mountain • Rate of the process is indeterminant at this time, hence assumed significant • Consequence is indeterminant; hence assumed significant • Annual event frequency is indeterminant at this time; hence assumed significant • Not bounded by another analysis
6	Military Activity-Induced Accidents	Accidents resulting from military activities Nevada Test Site or Nellis Air Force Range	Applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is indeterminant at this time; hence assumed significant • Consequence of the process is indeterminant at this time; hence assumed significant • Annual event frequency is indeterminant at this time; hence assumed significant • Not bounded by another analysis
7	Pipeline Accidents	Industrial pipeline transporting hazardous materials	Not applicable to the hazards list • No potential exists at Yucca Mountain; no industrial activities requiring pipelines containing hazardous materials exist or are planned to be located near the site
8	Undetected Past Intrusions	Past intrusions involve mining activities where deep shafts, drill holes, or tunnels may have been excavated	Not applicable to the hazards list • No potential exists at Yucca Mountain; site characterization provided sufficient assurance that these types of activities would have been detected

2.1.3.3.1 Hazards and Initiating Events Consideration

As shown in Tables 2.1.3-2 and 2.1.3-3, DOE included in the generic hazard list 45 natural events and 8 human-induced events that may have potentials for initiating event sequences leading to a radiological release during the preclosure period (CRWMS M&O, 1999b; DOE, 2001a). The events from the generic list were screened for potentials of becoming initiating events during a 100-year preclosure period taking into consideration the following five screening criteria (CRWMS M&O,1999b; DOE, 2001a):

- Potential exists for this event to be applicable to the proposed repository site at Yucca Mountain. Additional and separate analysis may be needed to establish the potential.

- Rate of the process is high enough to affect the potential repository during the 100-year preclosure period. If additional analysis can justify that the process occurs at too slow a rate to pose any potential hazard to the proposed repository during the 100-year period, the event will be screened out from further consideration.

- Consequence of the event is significantly high to affect the potential repository during the 100-year preclosure period.

- Event frequency is greater than or equal to 10^{-6} per year. Any event with a probability of occurring at least once in 10,000 during the 100-year preclosure period is included for further consideration.

- Event is not bounded by analysis of another event.

If all screening criteria are determined true for any natural event, the event is included in the hazard list for the proposed repository. If any statement or screening criterion cannot be evaluated appropriately at this time because of lack of specific information, the outcome of the screening criterion is assumed to be true.

It should be noted that some potential hazards are bounded by the analysis carried out for another hazard. For example, potential effects of rainstorm are bounded by the analysis for potential flooding and its associated effects. Sandstorm effects are included with extreme wind and tornado wind. Effects of subsidence are included in seismic activity—surface and subsurface fault displacement. As a result of the noted screening process and bounding analyses, DOE reduced the potential list of natural hazards to the proposed repository during preclosure period to nine events: (i) debris avalanche; (ii) extreme wind, including sandstorms; (iii) flooding, including rainstorm and river diversion; (iv) landslide; (v) lightning; (vi) seismic activity, earthquake; (vii) seismic activity, surface fault displacement; (viii) seismic activity, subsurface fault displacement, including subsidence; and (ix) tornado winds and tornado missiles.

DOE is committed to address both range fires and fires within the facility (DOE, 2001a). Appropriate prevention and mitigation controls will be provided in the design of the facility. DOE proposed to install a lightning protection system at the Waste Handling Building to prevent any direct lightning strikes on that building. Additionally, DOE concluded that waste packages would be able to withstand a direct lightning strike. Consequently, lightning has been excluded from the hazard list (DOE, 2001a).

DOE (2001a) stated that the site for surface facilities and the North Portal will be stabilized against debris avalanche and landslide. For preclosure safety analysis, these events have been grouped with flooding. Additionally, DOE grouped tornado wind loading with the extreme wind event and classified it as a tornado wind event. Tornado missile has been separately classified as a potential hazard.

As mentioned before, the staff initial review of the DOE identification of hazards and initiating events is ongoing. Following is a summary of the staff reviews of potential Aircraft Crash, Tornado Missiles, Volcanic Ash fall, and Operational hazards.

2.1.3.3.1.1 Aircraft Crash Hazard

DOE conducted an analysis to estimate hazards to the proposed repository at Yucca Mountain from potential aircraft crashes (CRWMS M&O, 1999d). DOE (CRWMS M&O, 1999d) used the suggested methodology of NUREG–0800 (NRC, 1981a) to estimate the probability of crash of an aircraft onto the proposed high-level waste repository. Additionally, CRWMS M&O (1999d) used the methodology suggested in DOE–STD–3014–96 (DOE, 1996) to estimate the effective area of a particular structure and the crash rate data for different aircraft developed by Kimura, et al. (1996). All these guidances are commonly used for estimating the aircraft crash hazard to a facility and are acceptable to NRC.

NRC (1981a) specifies that the probability of aircraft crash is considered to be less than approximately 10^{-7} per year by inspection if the distance from the facility (e.g., a nuclear power plant) meets all the following requirements:

(a) The facility-to-airport distance D is between 8 and 16 statute kilometers [5 and 10 statute miles] and the projected annual number of operations is less than $500 \times D^2$, or the facility-to-airport distance D is greater than 16 statute kilometers [10 statute miles] and the projected annual number of operations is less than $1000 \times D^2$.

(b) The facility is at least 8 statute kilometers [5 statute miles] from the edge of military training routes, including low-level training routes, except for those associated with a usage greater than 1,000 flights per year, or where activities (such as practice bombing) may create an unusual stress situation.

(c) The facility is at least 3.2 statute kilometers [2 statute miles] beyond the nearest edge of a federal airway, holding pattern, or approach pattern.

If the above proximity criteria are not satisfied or if sufficiently hazardous military activities are identified, a detailed review of aircraft crash hazards must be performed (NRC, 1981a).

CRWMS M&O (1999d) concluded that proximity criteria (a) and (c) are satisfied for commercial aircraft, private aircraft, DOE aircraft, and aircraft chartered by the DOE. Proximity criterion (b) is not applicable for these types of aircraft. Proximity criteria (a) and (b) are also satisfied for military aircraft. Only criterion (c) is not satisfied for military aviation in the vicinity of the proposed site and, therefore, an analysis estimating the annual crash frequency of military aviation is provided in CRWMS M&O (1999d).

The NRC staff disagree with the conclusion that criterion (b) of NUREG–0800, Section 3.5.1.6, Aircraft Hazards, has been met for the proposed repository site. The number of flights per year, as considered in CRWMS M&O (1999d), exceeds 1,000 flights per year by a significant margin (at least 12 to 15 times), and these flights create unusual stress situations as they fly in the restricted airspaces. It also should be noted that the above screening criteria are for nuclear power plants, none of which are located under a restricted military airspace. Therefore, criterion (b) has not been satisfied, and, consequently, a detailed analysis is necessary, as per NUREG–0800, Section 3.5.1.6, for every type of aircraft flying in the vicinity of the proposed site. The annual aircraft crash probability at the proposed facility will be the summation of probabilities from all types of aircraft engaged in different operations. Staff communicated this issue to DOE.[1] DOE agreed to develop a detailed analysis of the aircraft crash hazard using all types of aircraft flying in the vicinity of the proposed site.

Additionally, CRWMS M&O (1999d) assumed that considering the Waste Handling Building alone would be the best estimate case for estimating the aircraft crash hazard. The staff disagree with this assumption. The site plan shows that both the Waste Handling Building and the Waste Treatment Building are adjacent. Therefore, for estimating the effective area of the buildings, these two structures should be considered as one, as suggested in DOE (1996). Any crash of an aircraft on the Waste Treatment Building has the potential to affect the Waste Handling Building and any operations being conducted therein at the time of the crash. Staff communicated this issue to DOE[2] and DOE agreed to develop a revised analysis of the aircraft crash hazard at the proposed site.

DOE is also considering the option of a lower-temperature operational mode for the proposed repository (DOE, 2001a, Appendix A). One of the scenarios considered is extended surface aging of the commercial spent nuclear fuel on a pad located on the surface. This scenario will increase the effective area of the surface facilities that need to be considered for aircraft crash hazard analysis. This issue has not been previously raised with the DOE.

2.1.3.3.1.2 Tornado Missiles Hazard

DOE (CRWMS M&O, 1999e) used Section 3.5.1.4 of NUREG–0800 (NRC, 1981b) to identify the tornado missile characteristics, along with the expected impact velocity, appropriate for the proposed Yucca Mountain repository site. Additionally, DOE (CRWMS M&O, 1999e) identified the preliminary list of Quality Level 1 systems that need to be protected against the postulated tornado missiles impacts: (i) Assembly Transfer, (ii) Canistered Spent Nuclear Fuel Disposal Container, (iii) Canister Transfer, (iv) Defense High-Level Waste Disposal Container, (v) DOE Spent Nuclear Fuel Disposal Container, (vi) Waste Handling Building, (vii) Nonfuel Components Disposal Container, (viii) Uncanistered Spent Nuclear Fuel Disposal Container, (ix) Naval Spent Nuclear Fuel Disposal Container, (x) Waste Emplacement, and (xi) Waste Retrieval. Section 3.5.1.4 of NUREG–0800 (NRC, 1981b) provides an acceptable methodology for demonstrating compliance with the design of structures, systems, and components that need to withstand a postulated impact of tornado missiles and is acceptable to the NRC staff.

2.1.3.3.1.3 Volcanic Ash Fall Hazard

DOE concluded that no more than 3 cm [1.2 in] of volcanic tephra could be deposited on repository facilities during the preclosure period (CRWMS M&O, 1999b). DOE has thus excluded roof loading due to tephra fall from further consideration because the load imparted by a 3-cm-[1.2-in-] thick tephra deposit is bounded by the minimum design load requirements

[1]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[2]Ibid.

specified by the Uniform Building Code (International Conference of Building Officials, 1997). The NRC staff agree with the methodology of excluding volcanic tephra fall as a hazard; however, the NRC staff do not agree with the conclusion that a 3-cm- [1.2-in-] thick volcanic tephra deposit is the worst-case event to be expected at the proposed repository site. This issue is discussed in the next section.

2.1.3.3.1.4 Operational Hazards

The DOE operational hazard analysis methodology is documented in CRWMS M&O (1999a). This methodology, based on hazard analysis techniques described in System Safety Society (1997), consists of a generic checklist of events to identify the energy sources contained in a system (e.g., kinetic mechanical energy, electrical energy, chemical energy, thermal energy, and such) that can interact with the waste and potentially cause a radiological dose consequence to the public and facility workers. DOE used three safety analysis methodologies: Energy Analysis, Energy Trace and Barrier Analysis, and Energy Trace Checklist (System Safety Society, 1997), to develop the generic checklist of hazards applicable to the preclosure operations. The operational hazards have been classified into the following main hazard categories: (i) Collision/Crushing, (ii) Chemical/Contamination/Flooding, (iii) Explosion/Implosion, (iv) Fire/Thermal, and (v) Radiation/Magnetic/Electrical/Fissile Materials. The screening criteria, consisting of generic questions, were developed for each hazard category and applied to all the surface and subsurface operational areas of the geologic repository operations area to identify operational hazards and initiating events. DOE divided the surface and subsurface facilities into several functional areas for hazard analysis, as shown in Table 2.1.3-1. Although DOE methodology to identify hazards and initiating events is based on standard hazard analyses techniques, appropriateness and capability of the hazard analysis methodology for comprehensive identification of potential hazards at the proposed repository facility is being reviewed by staff. Preliminary review of the methodology suggests that the DOE method has a potential weakness. For example, hazards arising from incorrect actions because of human error have not been detected by the hazard analysis methodology. Numerous probabilistic risk assessment studies have shown that human errors can be important contributors to the risk associated with the operations of a nuclear facility (Swain and Guttman, 1983). It is expected that human error also will be a significant contributor to risk in the operations of the proposed repository (Eisenberg, 2001a). The DOE consideration of human factors, in the preliminary preclosure safety assessment, is confined to limited fault tree models to estimate the probability of events, such as a yoke drop from a bridge crane onto the fuel assemblies in the assembly transfer system (CRWMS M&O, 2000a), a runaway transporter carrying waste packages down the North ramp (CRWMS M&O, 1999f), or heating, ventilation, and air conditioning system unavailability (CRWMS M&O, 1999g). DOE should identify hazards and initiating events associated with human reliability in preclosure safety analysis in a consistent and unified manner in all the functional areas. The methodology proposed by DOE also does not identify potential hazards resulting from failure of the software and hardware systems used in the remote operations. During the preclosure period, surface and subsurface facility operations are expected to be remotely controlled for various equipment (e.g., overhead bridge cranes, trolleys, waste-container transporters, and gantries to move casks, canisters, bare-fuel assemblies, or waste packages) (DOE, 2001b). Software reliability may be a significant factor in the safe operation of the proposed Yucca Mountain repository (Eisenberg, 2001b). DOE should identify hazards and initiating events associated with reliability of hardware and software used in the operations in preclosure safety analysis. The preclosure topic concerning identification of operational hazards and initiating events was not discussed with DOE in the first DOE and NRC technical exchange and management meeting;[3] it will be discussed in a future technical exchange.

2.1.3.3.2 Site Data

As mentioned before, the staff review of DOE identification of hazards and initiating events is ongoing. Following is a summary of staff reviews of potential Aircraft Crash, Tornado Missiles, and Volcanic Ash fall hazards.

2.1.3.3.2.1 Aircraft Crash Hazard

Commercial and limited chartered aircraft use both McCarran International and North Las Vegas Airports. Chartered aircraft also use Tonopah Airport (CRWMS M&O,1999d). All three airports are more than 48 km [30 mi] from the proposed repository site. Commercial aircraft flying in the vicinity of the site use the federal airway V105–V135 (CRWMS M&O, 1999d). The airway V105–V135 is for air traffic below 5,400 m [18,000 ft] mean sea level. Jet Route J–92 overlies V105 and is used by air traffic above 5,400 m [18,000 ft] mean sea level (CRWMS M&O, 2000b). These airways are used by commercial air traffic between Las Vegas and Reno and other airports in the southwestern and northwestern United States. CRWMS M&O (2000b) states that the commercial air traffic is generally jet liners that fly above 5,400 m [18,000 ft] mean sea level through J–92. The proposed repository surface facilities are 17.6 statute kilometers [11 statute miles] away from the nearest edge of this 16-km [10-mi] wide airway. DOE has not provided information on the annual commercial air traffic through these airways for estimating the probability of crash onto the proposed facility. As DOE prepares detailed aircraft crash hazard analysis, commercial aircraft flying in these airways should be considered. Staff communicated this issue to DOE[4] and DOE agreed to develop a detailed analysis of the aircraft crash hazard using all types of aircraft flying in the vicinity of the proposed site.

General aviation aircraft flying under visual flight rules occasionally use U.S. Highway 95 for navigation and fly below 5,400 m [18,000 ft] mean sea level (CRWMS M&O, 2000b). CRWMS M&O (1999d) also indicated that private aircraft primarily use McCarran International, North Las Vegas, Beatty, Frans Star, and Jackass airports. It is not clear what is meant by private aircraft. DOE needs to clarify whether these private aircraft include general aviation aircraft and business jets. DOE has not provided any information regarding the flight pattern of these private aircraft in the vicinity of the proposed facility. DOE needs to provide detailed information on the number of annual flights, type(s) of aircraft, and any flight activity of these aircraft within the restricted airspace. This information should be based on historical record.

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[3]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[4]Ibid.

Staff communicated this issue to DOE[5] and DOE agreed to develop a detailed analysis of the aircraft crash hazard using all types of aircraft flying in the vicinity of the proposed site.

DOE aircraft and aircraft chartered by DOE also use the federal airways near the proposed site. These aircraft can use any airfield or landing strip within the Nevada Test Site (CRWMS M&O, 1999d). Airports controlled by DOE within 48 km [30 mi] of the proposed repository site are Desert Rock, Yucca, and Pahute Mesa airfields. Aircraft chartered by DOE for flying between Desert Rock airfield and laboratories in California and New Mexico use the federal airway V105–V135. The approach pattern to the Desert Rock airfield is outside the restricted area and at least 16 km [10 mi] away from the proposed repository site (CRWMS M&O, 1999d). Airway V105–V135 is 16 km [10 mi] wide. The nearest edge of this airway is 17.6 statute kilometers [11 statute miles] away from the proposed repository surface facilities. A total of 54,000 operations take place annually at Desert Rock, Yucca, and Pahute Mesa airfields (CRWMS M&O, 1999d). DOE has neither identified the number of annual operations at each of these airfields nor indicated the year in which 54,000 operations took place. Additionally, DOE has not indicated the type(s) of aircraft that use the airfields and the flight path(s) taken to reach the airfields. In addition, there are other federal airways near the proposed site. Staff communicated this issue to DOE[6] and DOE agreed to develop a detailed analysis of the aircraft crash hazard using all types of aircraft flying in the vicinity of the proposed site.

Helicopters routinely fly in most areas within the restricted airspace of the Nevada Test Site. Based on the information provided by CRWMS M&O (1999d), at least 1,440 helicopter flights take place annually within 3.2 km [2 mi] of the proposed repository surface facilities. These helicopters fly along Fortymile Wash, located 2.4 km [1.5 mi] from the proposed repository site. It is not clear what fraction of any of these helicopter flights overfly the proposed repository surface facilities. Assumption 4.3.4 of CRWMS M&O (1999d) states that the DOE Nevada Operations will adjust the helicopter routes to maintain a separation distance of 3.2 km [2 mi] from the surface facilities of the proposed repository. This is a to-be-verified item.

Military aircraft use Nellis Air Force Base, Tonopah Test Range, and Indian Springs Air Force Auxiliary Base airports located at distances greater than 48 km [30 mi] from the proposed site. Military aircraft, along with DOE aircraft and aircraft chartered by DOE, fly through the R–4808 restricted airspace. A classified memorandum of understanding exists between the U.S. Air Force and the DOE Nevada Operations that allows military aircraft to fly through the restricted airspace R–4808 for transitioning the 60-and 70-series ranges of the Nellis Air Force Base Range (CRWMS M&O, 1999d). The entire area is available for an aircraft to transit. No prior approval from DOE is needed unless specifically notified to the contrary by the DOE (Kimura, et al., 1998).

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[5]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001).” Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[6]Ibid.

Restricted airspace R–4808N is controlled by DOE for activities in the Nevada Test Site. R–4808S is jointly used by the Nevada Test Site, Nellis Air Force Base, and the Federal Aviation Administration, Los Angeles Air Traffic Route Traffic Control Center, for overflight of civilian aircraft. Southwestern and western parts of these restricted airspaces are used by military aircraft transiting to and from R–4807A and R–4807B. R–4808B is also used by DOE for flights to Pahute Mesa area as an extension of the Nevada Test Site. Additionally, there are 21 Military Training Routes within the Nellis Range Complex (U.S. Air Force, 1999); some are located close to the proposed repository site. Information about potential aircraft traffic in these restricted airspaces and military training routes is necessary to estimate the potential hazards to the proposed facility.

Based on the preceding discussion, CRWMS M&O (1999d) has not provided sufficient information on the flight activities by military aircraft while transitioning the restricted airspace R–4808 or in other nearby restricted airspaces. No information that may affect the safety of the proposed repository during the preclosure period has been provided on ordnance carried onboard the aircraft, flight path(s) taken by an aircraft with hung ordnance, or nearby areas where any training activities, such as air-to-air and air-to-ground combat training, are conducted by the U.S. Air Force. Information currently provided lacks sufficient details to develop an understanding of different activities conducted by the United States military near the proposed repository that may have an impact on proposed repository operations. Staff communicated this issue to DOE[7] and DOE agreed to develop a detailed map of activities by all types of aircraft flying in the vicinity of the proposed site. This map would be used to develop the revised aircraft crash hazard analysis, including information from federal and local agencies concerning how such activities may reasonably change in the future.

Estimation of aircraft crash probability requires reliable information on the parameters used in the estimation process. In addition, as discussed before, justifiable information on types of aircraft and flight activities is required for military aviation, especially when a facility is beneath a restricted military airspace. This information should be based on historical records with appropriate projections to the future to assess the hazard during the preclosure period of the proposed facility. Because the probability of aircraft crash to the proposed facility is directly proportional to the number of aircraft flying nearby, it is necessary to get a better estimate of the number of aircraft overflights than that given in CRWMS M&O (1999d). Kimura, et al. (1998) carried out a crash frequency analysis of aircraft overflying the Device Assembly Facility, located in Area 6 of the Nevada Test Site underneath the restricted airspace R–4808. They identified the number of overflights by military aircraft as one of the major sources of uncertainty in estimating aircraft crash frequency. They reported estimates that vary from 13,000 to 73,000 overflights per year. Estimates through the years vary as the mission of Nellis Air Force Base Range evolves. In CRWMS M&O (1999d), only 6 months of flight data through the R–4808N restricted airspace were presented. The number of flights per year, N, has been estimated by fitting a normal distribution to the 6 months (also to 5 months of flight information, because data for September 1996 were determined to be suspicious) of data using

[7]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

the Bestfit program of Palisade Corporation. Both 90- and 95-percent confidence levels were estimated from the fitted distribution. It was concluded that the fitted distribution is conservative. The number of flights per year, N, has been estimated to be (i) 12,716 (mean); (ii) 17,542 (90-percent confidence); and (iii) 18,910 (95-percent confidence) from the normal distribution fitted to the 6-month data. The staff disagree with this approach. Fitting a normal distribution to five or six data points leaves too few degrees of freedom to carry out any meaningful statistical analysis. As discussed in the manual of the Bestfit program, the Goodness-of-Fit tests are very sensitive to the number of data points. For a small number of data points, the tests will measure only a large difference between the input data and the distribution function. Consequently, the null hypothesis that the data were generated by a process that follows a particular distribution (in this case, normal distribution) will be accepted more often than in reality. Standard textbooks in statistics (e.g., Scheaffer and McClave, 1982) suggest that a sample size of less than 20 does not discriminate among different distributions. Many different distributions apparently may fit equally well to the data, as can be seen in the results for the Bestfit program. No single distribution produced the best fit using all three Goodness-of-Fit tests. Staff communicated this issue to DOE.[8] DOE stated that the Yucca Mountain Site Characterization Project Office is collecting overflight information by military aircraft in the vicinity of the proposed monitored geologic repository site. Recent information (Bechtel SAIC Company, LLC, 2001) shows that the average number of annual overflight increased approximately 37 percent, from 12,716 to 17,394, during the period of monitoring. DOE[9] agreed to develop a new aircraft crash hazard analysis taking into consideration aircraft overflight data appropriate to the proposed site.

No justification has been provided for classifying all the inflight mode flights by all military aircraft in the vicinity of the potential repository surface facilities as normal inflight mode. Normal inflight mode, as defined by Kimura, et al. (1996), includes “climb to cruise, cruise between an originating airfield and an operations area, if applicable, and cruise descent portions.” Special inflight mode includes “low-level and maneuvering operations in restricted area.” The proposed site lies underneath a restricted airspace and close to other restricted airspaces and military training routes. Staff communicated this issue to DOE[10] and DOE agreed to provide the mode of flight information of all types of aircraft in the vicinity of the proposed site, which would be used to develop the revised aircraft crash hazard analysis.

CRWMS M&O (1999d) assumed 29 percent of all aircraft will be F–16s, 63 percent will be F–15s, and 7 percent will be A–10s. No justification has been provided, however, why particular fractions of F–16, F–15, and A–10 aircraft were assumed in the analysis. Data from Nellis Air Force Base, presented in Table 7.2-3 of CRWMS M&O (1999d), do not indicate that the assumed distribution of these aircraft into these three types is reasonable. Moreover, a reasonable change in this distribution of the aircraft types, even with 12,716 flights in a year and

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[8]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001).” Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[9]Ibid.

[10]Ibid.

normal inflight mode, may raise the crash probability to more than 10^{-6} per year. Staff communicated this issue to DOE[11] and DOE agreed to provide details of types of military aircraft flying in the vicinity of the proposed site, which would be used to develop the revised aircraft crash hazard analysis.

It is not clear why the bounding case estimates in Tables III-3 and IV-3 of CRWMS M&O (1999d) use the crash rate of small aircraft (all types of fighter, trainer, and attack aircraft), instead of the F–16 which has the highest crash rate in normal and special inflight modes and would provide a bounding estimate. Trainer aircraft have much lower crash rates than fighters and attack aircraft (Kimura, et al., 1996). Staff communicated this issue to DOE[12] and DOE has agreed to provide justification or revise the aircraft crash hazard analysis.

CRWMS M&O (1999d) assumed F–16, F–15, and A–10 aircraft are representative for all types of aircraft flying near the proposed repository site. No justification has been provided why the analysis assumed only F–16, F–15, and A–10 aircraft when Tullman (1997) stated that "any aircraft in the Department of Defense inventory, or other NATO country, could fly these routes." A typical red flag exercise includes attack, fighter, bomber, air superiority, and reconnaissance aircraft; electric countermeasures suppression aircraft; aerial refueling aircraft; and search and rescue aircraft (U.S. Air Force, 1999). Staff communicated this issue to DOE[13] and DOE agreed to provide justification or revise the aircraft crash hazard analysis.

CRWMS M&O (1999d) does not provide any information on the ordnance carried on these aircraft. The pilot of an aircraft about to crash will attempt to jettison the ordnance first to gain altitude so more time is available to take corrective measures. The jettisoned ordnance could pose significant hazards to the proposed repository depending on the type and number of weapons. Additionally, live ordnance could pose additional hazards from flying fragments and air overpressure. Therefore, jettisoning of ordnance is also a concern for the site and should be investigated. Staff communicated this issue to DOE[14] and DOE agreed to provide the necessary information in the revised aircraft crash hazard analysis.

It should be noted that some information from the military regarding potential activities near the proposed repository site may be sensitive in nature and should be handled accordingly.

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[11]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[12]Ibid.

[13]Ibid.

[14]Ibid.

2.1.3.3.2.2 Tornado Missiles Hazard

DOE[15] proposed to screen out any effects of tornado missiles impacting a transporter carrying waste packages between the surface and subsurface facilities during the preclosure period. The rationale is that the waste package would be exposed to any potential tornado missile impact approximately 225 hours in a year. Assuming an annual frequency of missile-generating design-basis tornado to be 1×10^{-6}, the effective frequency of transporters exposed to a tornado missile would be of the order of 10^{-8} per year. The NRC staff disagreed with the approach. DOE needs to demonstrate that any impact from missiles generated by tornadoes with an annual frequency higher than 10^{-6} and with lower speed than the design-basis tornado would not cause unacceptable radiological release. An agreement with DOE was reached on this issue. DOE proposed to consider any administrative procedures as defense-in-depth measures when tornadoes would be predicted in the vicinity of the proposed site. Additionally, the current DOE tornado analysis does not address the scenario factored into the option of retrieval of waste packages. DOE[16] also proposed to update the analysis to include any potential effects of tornado missiles if retrieval of waste packages becomes necessary.

2.1.3.3.2.3 Volcanic Ash Fall Hazard

DOE analyzed potential hazards of volcanic ash to the proposed repository and concluded that a maximum 3-cm- [1.2-in-] thick volcanic tephra may be deposited at the proposed repository site. The 3-cm- [1.2-in-] thick deposit is from a volcanic eruption occurring 150 km [94 mi] from the proposed repository site [i.e., Perry and Crowe (1987)]. The basis for this conclusion is not supported by available analysis or data. Basaltic volcanic eruptions have an annual probability of occurrence that exceeds 1×10^{-6} per year at distances of approximately 10 km [6.3 mi] to 20 km [12.5 mi] southwest of the proposed repository site (e.g., NRC, 1999). Tephra-fall deposits measured approximately 10 km [6.3 mi] from volcanoes analogous to those within 20 km [12.5 mi] of Yucca Mountain are on the order of 1–100 cm [0.4–39 in] thick (e.g., Sagar, 1997). This issue was not discussed at the first Technical Exchange and Management Meeting for Preclosure Safety.[17]

2.1.3.3.3 Probability of Occurrence Determination

As mentioned before, the staff review of the DOE identification of hazards and initiating events is ongoing. Following is a summary of staff reviews of potential Aircraft Crash, Tornado Missiles, and Volcanic Ash Fall hazards.

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[15]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[16]Ibid.

[17]Ibid.

2.1.3.3.3.1 Aircraft Crash Hazard

Commercial aircraft use both McCarran International and North Las Vegas Airports. Limited chartered aircraft use Tonopah Airport (CRWMS M&O, 1999d). All three airports are more than 48 km [30 mi] from the proposed site. Consequently, more than 900,000 annual takeoff and landing operations would be necessary at these airports to have a crash probability of 10^{-7} per year to the proposed repository site. The number of commercial and general aviation aircraft currently taking off and landing at these airports is small and less than $1{,}000D^2$, where D is the distance between an airport and the site (NRC, 1981a). Therefore, current operations (landings and takeoffs) at these airports may be assumed to be negligible contributors to the overall aircraft crash hazard probability at the proposed site. DOE estimated that the crash probability at the proposed site from aircraft takeoff and landing at these three airports would be negligible. If the projected traffic growth at any of these airports increases significantly during the preclosure/operational life of the proposed facility to violate the $1{,}000D^2$ criterion, however, a detailed analysis will be necessary.

CRWMS M&O (1999d) indicated that private aircraft primarily use McCarran International, North Las Vegas, Beatty, Frans Star, and Jackass airports. Staff assume private aircraft are general aviation aircraft and include business jets. Other airports in the vicinity are small with low traffic. Only Beatty, Frans Star, and Jackass airports are within 32 km [20 mi] of the proposed site. Similarly, DOE aircraft and aircraft chartered by DOE use Desert Rock, Yucca, and Pahute Mesa airfields (CRWMS M&O, 1999d). The number of annual operations at each of these airports is significantly small to pose a credible hazard to the proposed site based on the distance and number of operations criterion of NRC (1981a). DOE stated that flights taking off and landing at these airports will have negligible contributions to the estimated aircraft crash hazard probability of the proposed site. Any projected traffic increase during the preclosure period should also be considered in the analysis.

Commercial aircraft flying in the vicinity of the proposed repository site use the federal airway V105–V135 (CRWMS M&O, 1999d). The distance from the nearest edge of this 16-km [10-mi] wide airway to the proposed site is 17.6 statute kilometers [11 statute miles]. The estimated crash probability of aircraft flying route V105–V135 will be a component of total aircraft crash probability onto the proposed site. DOE has not estimated the probability of crashes of aircraft flying this airway. Staff communicated this issue to DOE[18] and DOE agreed to provide an estimate of the crash hazard from aircraft flying the airway V105–V135 in the revised aircraft crash hazard analysis.

DOE aircraft and aircraft chartered by DOE fly between Desert Rock airfield at the Nevada Test Site and DOE laboratories and use the airway V105–V135. Some DOE aircraft and aircraft chartered by DOE also fly to Yucca and Pahute Mesa airfields within the Nevada Test Site (CRWMS M&O, 1999d). DOE (CRWMS M&O, 1999d) has not estimated the potential crash probability of DOE aircraft and aircraft chartered by DOE while flying to Desert Rock, Yucca,

[18]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

and Pahute Mesa airfields. The revised analysis of aircraft crash hazard should include these crash probability estimates. Staff performed a preliminary analysis to estimate the crash probability of DOE aircraft and aircraft chartered by DOE onto the proposed facility while transiting the airway V105–V135 as an example (Ghosh and Sagar, 2001). The details follow.

Because many of the flights to Desert Rock, Yucca, and Pahute Mesa airfields use charter aircraft (CRWMS M&O, 1999d), staff carried out a preliminary estimate assuming the aircraft would be similar to commercial aircraft in crash statistics. Therefore, Air Carrier characteristics in DOE–STD–3014–96 (DOE, 1996) will be applicable. Specific information on the type(s) of aircraft used by DOE, however, should be used to verify this assumption. Crash rate, C, for commercial aircraft is assumed to be 4×10^{-10} per flight mile (NRC, 1981a) for lack of information on specific aircraft type(s). As V105–V135 is a heavily traveled air corridor (more than 100 daily flights), the revised analysis to be carried out by the DOE may also require a more accurate estimate of the crash rate of the aircraft flying this airway (NRC, 1981a).

Approximately 54,000 annual flights of DOE aircraft use Desert Rock, Yucca, and Pahute Mesa airfields (CRWMS M&O, 1999d). Information is not available, however, about the number of annual flights to each of these airfields. Staff assumed, in one scenario, that all 54,000 flights use Desert Rock airfield. Staff also made another estimate assuming one-third of the 54,000 flights use each airport, which, by nature of the runway surface, is not a valid assumption. The effective area, A_{eff} of the surface facilities at the proposed repository has been calculated as the sum of the effective areas of each of the five structures where radioactive materials potentially can be located (CRWMS M&O, 1999d) and is equal to 0.641 km^2 [0.251 mi^2] (Ghosh and Sagar, 2001). The effective width of the airway, *W*, is 16 + 2 × 17.6, or 51.2 km [32 mi], because the airway V105–V135 is 16 km [10 mi] wide and at a distance of 17.6 statute miles [11 statute miles] from the proposed site (CRWMS M&O, 1999d). Therefore, the annual probability of crash, *P*, from DOE aircraft and aircraft chartered by DOE, based on NRC (1981a), is

$$P = N \times C \times \frac{A_{eff}}{W} = 54000 \times 4 \times 10^{-10} \times \frac{0.251}{32} = 1.7 \times 10^{-7} \qquad (2.1.3\text{-}1)$$

Assuming only one-third of the aircraft use Desert Rock airfield, the annual crash probability is 6×10^{-8}, which, as discussed before, may not be representative of the actual situation. Estimating the crash hazard of aircraft specifically flying to Yucca and Pahute Mesa airfields requires information of flight path(s) in addition to the previous information. Hence, the staff estimation was limited by lack of information. This analysis brings out the effects of lack of specific information on flight activities, as discussed in the previous section, on the estimated crash probability. Lack of specific information introduces significant uncertainty in the estimated crash probability. Several different scenarios seem equally probable. Developing a bounding scenario becomes quite difficult due to lack of defensible information. Staff communicated this issue to DOE[19] and DOE agreed to provide the necessary information and annual crash hazard estimation in the revised aircraft crash hazard analysis.

As discussed previously, DOE has not provided justification for the proportion of F–16, F–15, and A–10 aircraft assumed in the analysis (CRWMS M&O, 1999d). The staff carried out a preliminary sensitivity analysis to estimate the crash probability of military aircraft onto the proposed facility using several different scenarios (Ghosh and Sagar, 2001). The effective areas of the surface facilities were estimated for each of the three aircraft types assumed in the analysis (same types as used in CRWMS M&O, 1999d) using DOE–STD–3014–96 (DOE, 1996). Using both normal and special in-flight crash rates for the F–16, F–15, and A–10 aircraft from Kimura, et al. (1996), the estimated probabilities of a crash are given in Table 2.1.3-4. This sensitivity analysis shows the importance of having justifiable and specific information on the number of military aircraft flights with the associated activities by different aircraft types. Staff communicated this issue to DOE[20] and DOE agreed to provide justifiable information on aircraft types, numbers of flights, proportions of flights conducted by each aircraft type, and associated flight activities with appropriate future projections during the preclosure period in the revised aircraft crash hazard analysis.

Table 2.1.3-4. Estimated Probabilities of Crash, P, for Military Aircraft for Different Scenarios

Number of Aircraft Flights	F–16 (percent)	F–15 (percent)	A–10 (percent)	Flight Mode	Annual Crash Probability
12,716	29	63.9	7.1	Special	3.8×10^{-6}
17,542	29	63.9	7.1	Special	5.2×10^{-6}
18,910	29	63.9	7.1	Special	5.6×10^{-6}
12,716	100	0	0	Special	4.5×10^{-6}
18,910	100	0	0	Special	6.7×10^{-6}
12,716	100	0	0	Normal	1.5×10^{-6}
18,910	100	0	0	Normal	2.3×10^{-6}
12,716	50	40	10	Special	4.0×10^{-6}
18,910	50	40	10	Special	5.9×10^{-6}
12,716	50	40	10	Normal	1.0×10^{-6}
18,910	50	40	10	Normal	1.5×10^{-6}

[19]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[20]Ibid.

CRWMS M&O (1999d) erroneously used the formulas to calculate the effective area of a structure to estimate the aircraft crash hazard probability specified in the DOE standard (DOE, 1996, Appendix B). As a consequence of the erroneous use of these formulas, the estimated effective area determined is smaller and, hence, nonconservative. The difference is more pronounced for structures more square in shape, such as the Waste Handling Building. Staff communicated this issue to DOE[21] and DOE agreed to revise the analysis of the aircraft crash hazard at the proposed site applying the formulas as recommended in the DOE standard.

CRWMS M&O (1999d) assumed that information provided by the Nellis Air Force Base staff on expected air traffic and types of aircraft currently flying through the restricted airspace R–4808N is representative of those flying at the time of repository operation. This information was transmitted to DOE in 1997. In the aircraft hazard analysis, DOE (CRWMS M&O, 1999d) has not considered any reasonable changes in flight activities in the vicinity of the proposed repository site into account. Staff communicated this issue to DOE[22] and DOE agreed to consider information from federal and local agencies concerning how such activities may reasonably change in the future.

2.1.3.3.3.2 Tornado Missiles Hazard

DOE estimated that the frequency of transporters exposed to a tornado missile would be on the order of 10^{-8} per year. The NRC staff questioned the basis for assuming the annual frequency of a missile-generating tornado at the proposed site to be equal to 10^{-6}. DOE needs to demonstrate that tornadoes with higher annual frequency (larger than 10^{-6}) with lower wind speed, as analyzed, would not impact any structures, systems, and components causing unacceptable radiological release. Staff communicated this issue to DOE[23] and DOE agreed to provide an analysis, including (i) selection of the design basis tornado together with the supporting technical basis; (ii) selection of credible tornado missile characteristics for the waste package and other structures, systems, and components together with the technical bases; and (iii) analysis of the effects of impact of the design basis tornado missiles or justification for excluding such tornado missiles as credible hazards.

2.1.3.3.3.3 Volcanic Ash Fall Hazard

DOE concluded, in analyzing potential natural hazards to the proposed repository, that a 3-cm-[1.2-in-] thick volcanic tephra deposit is the worst-case event; however, the basis for this conclusion is not supported by available analysis or data. The 3-cm-[1.2-in-] thick deposit cited by CRWMS M&O (1999b) applies only for a volcanic eruption occurring 150 km [94 mi] from the proposed repository site (i.e., Perry and Crowe, 1987). Basaltic volcanic eruptions have an annual probability of occurrence that exceeds 1×10^{-6} per year at distances of approximately

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[21]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001).” Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[22]Ibid.

[23]Ibid.

10 km [6.3 mi] to 20 km [12.5 mi] southwest of the proposed repository site (e.g., NRC, 1999). Tephra-fall deposits measured approximately 10 km [6.25 mi] from volcanoes analogous to those within 20 km [12.5 mi] of Yucca Mountain are on the order of 1–100 cm [0.4–39 in] thick (e.g., NRC, 1997). These deposits increase in thickness to approximately 400 cm [157 in] within 1 km [0.63 mi] of the volcanic vent. In addition, Perry and Crowe (1987) conclude that a 1-m-[3.3-ft-] thick tephra deposit could occur approximately 3 km [1.9 mi] from a basaltic volcanic vent. Because the volcanic event may take place anywhere within 10 km [6.3 mi] of the proposed repository site, a tephra fall deposit with a thickness of 100–400 cm [39–157 in] on the surface facilities is a potential hazard that needs to be considered. Noncompacted, dry basaltic volcanic tephra has a bulk deposit density that can range 1,200–1,700 kg/m^3 [75–106 lb/ft^3] (e.g., Hill, et al, 1998; NRC, 1999). The density of these deposits can increase by roughly a factor of two when wet, depending on average grain size and sorting of the deposit. Thus, a basaltic volcanic eruption in the area around Yucca Mountain represents a Category 2 event that could deposit 100–400 cm [39–157 in] of tephra on surface structures. These deposits could result in loads greater than 115 kPa [240 lb/ft^2], significantly larger than that assumed to screen out this event as a potential natural hazard to the proposed repository. This issue was outside the scope of the first Technical Exchange and Management Meeting for Preclosure Safety.[24]

2.1.3.3.4 Exclusion or Inclusion of Hazards and Initiating Events

As discussed before, staff review of the DOE identification of hazards and initiating events is ongoing. Following is a summary of the staff review of potential Aircraft Crash, Tornado Missiles, and Volcanic Ash fall hazards.

2.1.3.3.4.1 Aircraft Crash Hazard

DOE excluded the aircraft crash hazard from the credible hazard list (CRWMS M&O, 1999d, 2000a; DOE, 2001a; Bechtel SAIC Company, LLC, 2001). Based on the preceding review, however, the NRC staff conclude that exclusion of aircraft crash hazard during the preclosure period is premature. There is a significant lack of specific information about the potential aircraft activities in the vicinity of the proposed site. Explicit and inherent assumptions taken and the technical bases were not adequately justified. Additionally, uncertainties in the data, compounded by lack of specific information, were not adequately characterized. Staff communicated this issue to DOE[25] and DOE agreed that exclusion of this hazard is premature. DOE has agreed to provide justifiable information on aircraft types, number of flights, proportion of flights conducted by each aircraft type, and associated flight activities with appropriate future projections during the preclosure period in the revised aircraft crash hazard analysis.

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[24]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[25]Ibid.

2.1.3.3.4.2 Tornado Missiles Hazard

Based on the discussion given in previous sections, NRC staff consider elimination of the potential tornado missiles hazard from further consideration is not supported by acceptable data, analysis, and technical bases. Staff communicated this issue to DOE[26] and DOE agreed to carry out an analysis to include the potential effects of tornado missiles or to justify exclusion of this hazard from further consideration.

2.1.3.3.4.3 Volcanic Ash Fall

DOE eliminated the potentially adverse effects of volcanic eruptions characteristic of the Yucca Mountain region from the list of Category 2 event sequences during preclosure without adequate justification for assuming the distance of nearby volcanic event sequences and the thicknesses of associated tephra fall deposit. Adequate rationale is needed to justify exclusion of this event from the Category 2 event sequences list. This issue was outside the scope of the first Technical Exchange and Management Meeting for Preclosure Safety.[27]

DOE eliminated the potential effects of volcanic tephra particles on high-efficiency particulate air filters and heating, ventilation, and air conditioning system systems based on the analogy of the effects of wind-blown sand particles during a sandstorm. DOE assumed the effects of volcanic tephra on high-efficiency particulate air filters and heating, ventilation, and air conditioning system systems are bounded by sandstorms (CRWMS M&O, 1999b) without providing information about the particle sizes in both events. Volcanic tephra fall deposits contain a greater range of particle sizes than wind-blown sands, which may have different effects on high-efficiency particulate air filters and heating, ventilation, and air conditioning systems. This issue was not discussed at the first Technical Exchange and Management Meeting for Preclosure Safety.[28]

2.1.3.3.4.4 List of Hazards and Initiating Events

Staff currently are reviewing the DOE list of hazards and initiating events. Issues will be developed in a future revision of this document.

2.1.3.4 Status and Path Forward

Identification of hazards and initiating events during the preclosure period is considered pending by the NRC staff. Further information will be required at the time of any license application.

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[26]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[27]Ibid.

[28]Ibid.

At the first Technical Exchange and Management Meeting for Preclosure Safety,[29] the NRC staff discussed only Aircraft Crash Hazard and Tornado Missiles Hazards with the DOE. Because the meeting focused on general methodologies, many specific comments were not raised at that meeting. The status of issue closure in the preclosure safety area was not discussed. Table 2.1.3-5 provides the status of the preclosure identification of hazards and initiating events.

Table 2.1.3-5. Summary of Resolution Status Hazard and Initiating Events Identification Preclosure Topic

Preclosure Items	Status	Related Agreements*	Comments
Hazards and Initiating Events Consideration	Pending	PRE.03.01	Staff Review Incomplete
Site Data	Pending	PRE.03.01 PRE.03.02	Staff Review Incomplete
Exclusion or Inclusion of Hazards and Initiating Events	Pending	PRE.03.01 PRE.03.02	Staff Review Incomplete
List of Hazards and Initiating Events	Pending	None at this time	Staff Review Incomplete

*The first Technical Exchange and Management Meeting for Preclosure Safety focused only on Aircraft Crash and Tornado Missiles Hazards. No agreements on other hazards and initiating events were reached.

2.1.3.5 References

Bechtel SAIC Company, LLC. “Technical Update Impact Letter Report.” MIS–MGR–RL–000001. Revision 00. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2001.

CRWMS M&O. “Monitored Geologic Repository Internal Hazards Analysis.” ANL–MGR–SE–000003. Revision 00. Las Vegas, Nevada: CRWMS M&O. 1999a.

———. “MGR External Events Analysis.” ANL–MGR–SE–000004. Revision 00. Las Vegas, Nevada: CRWMS M&O. 1999b.

———. “Preliminary Selection of MGR Design Basis Events.” ANL–WHS–SE–000003. Revision 00. Las Vegas, Nevada: CRWMS M&O. 1999c.

———. “MGR Aircraft Crash Frequency Analysis.” ANL–WHS–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 1999d.

[29]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001).” Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

———. “MGR Design Basis Extreme Wind/Tornado Analysis.” ANL–MGR–SE–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 1999e.

———. “Subsurface Transporter Safety Systems Analysis.” ANL–WER–ME–000001. Revision 01. Las Vegas, Nevada: CRWMS M&O. 1999f.

———. “Reliability Assessment of Waste Handling Building HVAC System.” BCBD00000–01717–0210–00008. Revision 00. Las Vegas, Nevada: CRWMS M&O. 1999g.

———. “Design Basis Event Frequency and Dose Calculation for Site Recommendation.” CAL–WHS–SE–000001. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000a.

———. “Yucca Mountain Site Description.” TDR–CRW–GS–000001. Revision 01 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000b.

DOE. “DOE Standard: Accident Analysis for Aircraft Crash Into Hazardous Facilities.” DOE–STD–3014–96. Washington, DC: DOE. 1996.

———. “Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation.” TDR–MGR–SE–000009. Revision 00 ICN 03. Las Vegas, Nevada: DOE. 2001a.

———. “Yucca Mountain Science and Engineering Report Technical Information Site Recommendation Consideration.” DOE/RW-0539. Las Vegas, Nevada: DOE. 2001b.

Eisenberg, N. “Human Reliability Analysis and Repository Preclosure Safety—An Initial Evaluation.” San Antonio, Texas: CNWRA. 2001a.

———. “Software Reliability Analysis and Repository Preclosure Safety—An Initial Survey.” San Antonio, Texas: CNWRA. 2001b.

Ghosh, A. and B. Sagar. “Preliminary Aircraft Crash Hazard Assessment at Proposed Yucca Mountain Repository.” Proceedings of the Ninth International Conference on High-Level Radioactive Waste Management, Las Vegas, Nevada, April 29–May 3, 2001. La Grange Park, Illinois: American Nuclear Society. 2001.

Hill, B.E., C.B. Connor, M.S. Jarzemba, P.C. La Femina, M. Navarro, and W. Strauch. “1995 Eruptions of Cerro Negro Volcano, Nicaragua, and Risk Assessment for Future Eruptions.” *Geological Society of America Bulletin.* Vol. 10. pp. 1,231–1,241. 1998.

International Conference of Building Officials. “Uniform Building Code.” Whittier, California: International Conference of Building Officials. 1997.

Kimura, C.Y., D.L. Sanzo, and M. Sharirli. “Crash Hit Frequency Analysis of Aircraft Overflights of the Nevada Test Site and the Device Assembly Facility.” UCRL–ID–131259. Revision 1. Livermore, California: Lawrence Livermore National Laboratory. 1998.

Kimura, C.Y., R.E. Glaser, R.W. Mensing, T. Lin, T.A. Haley, A.B. Barto, and M.A. Stutzke. "Data Development Technical Support Documentation for the Aircraft Crash Risk Analysis Methodology (ACRAM) Standard." UCRL–ID–124837. Livermore, California: Lawrence Livermore National Laboratory. 1996.

NRC. NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants. Section 3.5.1.6—Aircraft Hazards." Washington, DC: NRC. 1981a.

———. NUREG–0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants. Section 3.5.1.4—Missiles Generated by Natural Phenomena." Washington, DC: NRC. 1981b.

———. "Issue Resolution Status Report, Key Technical Issue: Igneous Activity." Revision 2. Washington, DC: NRC. 1999.

———. NUREG–1804, "Yucca Mountain Review Plan—Draft Report for Comments." Revision 2. Washington, DC: NRC. March 2002.

Perry, F.V. and B.M. Crowe. "Preclosure Volcanic Effects: Evaluations for a Potential Repository Site at Yucca Mountain, Nevada." Los Alamos, New Mexico: Los Alamos National Laboratory. 1987.

Sagar, B., ed. NUREG/CR–6513, No. 1, CNWRA 96–OIA, "NRC High-Level Radioactive Waste Program Annual Progress Report: Fiscal Year 1996." Washington, DC: NRC. January 1997.

Scheaffer, R.L. and J.T. McClave. *Statistics for Engineers*. Boston, Massachusetts: Duxbury Press. 1982.

Swain, A.D. and H.E. Guttman. NUREG/CR–1278, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications." Albuquerque, New Mexico: Sandia National Laboratories. 1983.

System Safety Society. *System Safety Analysis Handbook—A Source Book for Safety Practitioners*. 2nd Edition. Albuquerque, New Mexico: System Safety Society. 1997.

Tullman, E.J. "Nellis Airspace and Crash Data for Yucca Mountain Hazard Analysis." Letter (June 5) with enclosure to W.E. Barnes, Yucca Mountain Site Characterization Office. U.S. Department of Energy, Las Vegas, Nevada: U.S. Air Force, USAF/DOE Liaison Office. 1997.

U.S. Air Force. "Renewal of the Nellis Air Force Range Land Withdrawal: Legislative Environmental Impact Statement." Nellis Air Force Base. Las Vegas, Nevada: U.S. Air Force, Air Combat Command. 1999.

2.1.4 Identification of Event Sequences

2.1.4.1 Description of Issue

This section of the Integrated Issue Resolution Status Report addresses assessment of the DOE identification of event sequences and categorization of event sequences. 10 CFR 63.112(b) requires that, in the license application, the DOE preclosure safety analysis of the geologic repository operations area must include comprehensive identification of potential event sequences. An event sequence is defined in 10 CFR 63.2 as a series of actions and/or occurrences within the natural and engineered components of a geologic repository operations area that could potentially lead to exposure of individuals to radiation. All identified event sequences are categorized based on their frequencies of occurrence. According to 10 CFR 63.2, those event sequences expected to occur one or more times before permanent closure of the geologic repository operations area are referred to as Category 1 event sequences. Other event sequences that have at least 1 chance in 10,000 of occurring before the permanent closure are referred to as Category 2 event sequences. DOE is required to demonstrate that Category 1 and Category 2 event sequences meet the preclosure performance requirements stated in 10 CFR 63.111.

Event sequence analyses are based on development of event scenarios that include an initiating event and the subsequent sequence of events associated with the failure of structures, systems, or components, including those produced by human actions. The scenario development process results in a series of event sequences, each having a specific frequency of occurrence. The scenarios are analyzed for event sequence frequencies using event tree and fault tree analysis techniques. DOE should ensure that all possible event scenarios are considered and that all event trees and fault trees are analyzed accounting for uncertainty and variability in the estimated frequency and probability data. Inaccurate evaluation of the frequency of occurrence can lead to potential miscategorization of event sequences and erroneous safety assessment.

Based on the preliminary design of the proposed repository, DOE identified some event sequences reported in DOE (2001a) and associated reports (CRWMS M&O; 1997a, 1998, 2000a). This section of the Integrated Issue Resolution Status Report has been prepared based on the limited review of a selected number of these reports and the discussion at the first DOE and NRC preclosure technical exchange,[1] which concentrated primarily on the methodology of event sequence identification. No agreements have been reached on specific issues concerning identification of event sequences. It is expected that the staff will continue to review additional reports and develop a comprehensive list of issues relating to the preclosure safety analysis.

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[1]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

2.1.4.2 Importance to Safety

Identification of event sequences and their categorization is an integral part of the preclosure safety analysis. 10 CFR 63.2 defines the preclosure safety analysis as a systematic examination of the site, design, potential hazards, initiating events, and event sequences and their dose consequences. The objectives of the preclosure safety analysis are to ensure the facility design complies with the performance requirements and to identify the structures, systems, and components relied on for safe functioning of the facility. Additionally, DOE intends to further classify the structures, systems, and components in a graded fashion in accordance with its classification procedure (DOE, 2001a,b).

The DOE identification of structures, systems, and components important to safety and the DOE classification process are based on the capability of the structures, systems, and components to function without potential for exceeding the dose limits specified in the performance requirements of Category 1 event sequences and to prevent or mitigate the dose consequence of Category 2 event sequences. The preclosure safety analysis of the repository requires appropriate identification and categorization of the event sequences. A comprehensive safety analysis will depend on an accurate accounting and characterization of event sequences.

2.1.4.3 Technical Basis

The complexity associated with the preclosure operations develop from the (i) large inventory of radioactive wastes received at the site; (ii) large number of surface processing operations that will be performed, many in parallel, to repackage waste; and (iii) subsurface operations involving transportation and emplacement of waste packages in the underground drifts. The proposed repository will have the capability to receive and emplace approximately 70,000 MTU of spent nuclear fuel and high-level waste (CRWMS M&O, 1999a). The reference design is based on an annual receipt rate of 3,000 MTU for an operational period of 24 years (CRWMS M&O, 1999b). The annual rate of receipt and handling of casks, canisters, fuel assemblies, and waste packages in the proposed facility will vary from year to year. 10 CFR 63.21(c)(5) requires that, for the purpose of the preclosure safety analysis, it should be assumed that the operations at the proposed facility will be accomplished at the maximum capacity and rate of receipt of waste. The schedule for annual receipt and handling of casks, canisters, and waste packages in different areas of the facility is shown in Table 2-2 of the CRWMS M&O report (1999b). The peak annual handling operations given in this table indicate that the waste will undergo substantial handling operations in the proposed facility.

The DOE identification of event sequences that could potentially release radioactive material to the members of the public and facility workers is presented in DOE (2001a) and in other DOE documents (2001b,c). The DOE preliminary hazards analysis identified nine natural and human-induced initiating events that could potentially cause radiological release (DOE, 2001a,Table 5-4). DOE did not develop event scenarios from these initiating events because DOE proposed to design, construct, and operate the proposed repository to withstand these events so that no scenarios resulting in release of radioactive material are initiated (DOE, 2001c). In the future, when DOE submits the design, the staff will review and evaluate the adequacy of the DOE design, construction, and operations to withstand these initiating events.

DOE developed lists of potential event sequences from the events generated only from the facility operations. The potential event sequences have been classified into three groups: internal event sequences with potential release, internal event sequences with no release, and beyond design basis events. Staff comments in this version of the Integrated Issue Resolution Status Report are limited only to the operational hazards.

The event sequences resulting from the proposed facility operations of a geologic repository operations area that could potentially release radioactive material were further categorized as Category 1 and Category 2 based on the frequency of occurrences from the event sequence analyses (DOE, 2001a, Tables 5-5 and 5-6). DOE identified 14 Category 1 event sequences and 12 Category 2 event sequences (CRWMS M&O, 2000a). Using the bounding consequence argument for some of the event sequences, the number of Category 2 event sequences were further reduced to nine (DOE, 2001a,b,c).

DOE identified 35 event sequences not expected to result in radiological release (DOE, 2001a, Table 5-7). The event sequences in this group have been determined credible (i.e., expected to occur during the geologic repository operations area operational period), however, DOE excluded these event sequences from repository preclosure safety analysis. DOE plans to design the facility such that structures, systems, and components will either prevent these event sequences from occurring or prevent a release should the event occur. Event sequences identified in this group are primarily related to waste package drops during surface and subsurface operations (CRWMS M&O, 1997b, 2000b).

DOE also generated a list of beyond design basis events containing approximately 22 event sequences (DOE, 2001a, Table 5-12). The frequency of occurrence of these event sequences is less than 1 chance in 10,000 of occurring during preclosure period and based on specific facility design features, physical barriers, and administrative controls or a combination of these factors. DOE has excluded these event sequences from further analyses (e.g., consequence analyses) because, for event sequences with less than 1 chance in 10,000 of occurring before permanent closure,10 CFR Part 63 does not require their consideration in the repository safety analysis. DOE, however, observes that these event sequences may become credible if the prevention and mitigation features are altered because of changes in the facility design (DOE, 2001a).

This review is organized according to the two acceptance criteria consistent with the associated review methods and acceptance criteria in the Yucca Mountain Review Plan (NRC, 2002). The following acceptance criteria are based on meeting the requirements of 10 CFR 63.112(b), relating to the identification of event sequences.

2.1.4.3.1 Justification for Methodology and Assumptions

The DOE event sequence analysis using the event tree technique is acceptable because it is universally applicable to systems of all kinds and is widely used in probabilistic risk analysis for nuclear powerplants (NRC, 1983). DOE identification of operational event sequences has been reported in CRWMS M&O (2000a). DOE scenario development and event sequence analyses, which are based on preliminary facility design, simulate a simple three branch event tree analysis that includes an initiating event and two event sequences consisting of failure of a structure, system, or component associated with the scenario and the availability/nonavailability of a heating, ventilation, and air conditioning system with high-efficiency particulate air filtration (DOE, 2001a; CRWMS M&O, 2000a). Although the event tree technique is exhaustively thorough, the success of the technique is based on three basic presumptions (NRC, 1983; System Safety Society, 1997): (i) that all system events have been anticipated, (ii) all end states of these events have been explored, and (iii) the probabilities of failure for all the events have been correctly assumed. The staff tentatively agree with overall DOE approach. Staff expect DOE to provide a detailed rationale for its scenario development. The presentation of the detailed event sequence and the determination of the probability and frequency values used in the event tree analysis should be transparent and traceable to enable a staff review.

DOE has not provided adequate justification for the appropriateness of the data used to estimate probability of failure for the equipment and components used in the surface and subsurface operations event sequence analyses. For example, data used by the DOE to determine probability of drop events for assemblies and shipping casks are based on analyses of the drop events of the cranes obtained from the industry (CRWMS M&O, 1997b,1998, 2000a). DOE should provide justification that the data used from the industry to estimate failure probability are appropriate for use in repository operations. Staff concern on this issue was discussed with DOE staff at the DOE and NRC Technical Exchange on Pre-Closure Safety.[2] Although no agreement was formulated at the meeting, DOE concurred with the NRC position that the appropriateness of the failure probabilities must be justified sufficiently to support the event sequence categorization process.

DOE has presented event sequence analyses with only point estimates of probability of failure of different components (CRWMS M&O, 2000a). It is not clear whether the probability estimate DOE used in its analysis represents mean, median, or some other point estimate. Frequency of component failure is, however, highly uncertain. By ignoring the uncertainty and variability associated with each frequency or probability estimate, there is a distinct possibility of incorrectly classifying an event sequence with associated consequences. DOE should assign distribution to component failures and consider uncertainty and sensitivity analyses to estimate event sequence frequency. NRC stated its position that if DOE obtains a probability distribution for the frequency of a preclosure event sequence, the mean value of that distribution can be used to categorize the event sequence, provided that the probability distributions of the component failures are valid and account appropriately for uncertainty and variability. Staff concern on this issue of not considering uncertainty and variability of probability data used in event sequence analysis was discussed with DOE at the DOE and NRC Technical Exchange on Pre-Closure Safety.[3] Although no agreements were formulated on this issue, DOE stated that it would, as appropriate, assign probability distribution to component failure rate estimates. DOE also agreed with NRC to render appropriate attention to the event sequences near the thresholds of Category 1 and Category 2 frequency limits and to ensure that the technical basis supports the event categorization or that the event sequences are conservatively categorized.

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[2]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[3]Ibid.

2.1.4.3.2 Identification of Category 1 and 2 Event Sequences

DOE has not demonstrated continuity and traceability in its preclosure safety analysis. It identified potential hazards and initiating events from the surface and subsurface operations in CRWMS M&O (1999b). DOE also developed a generic events checklist containing a series of questions for each postulated generic hazard germane to the proposed repository operations. The checklist questionnaires were applied to each functional area of the repository to identify possible initiating events. The initiating events were further analyzed for their frequency of occurrences in several CRWMS M&O reports (1997a,b,1998, 1999c, 2000a,b). The credible initiating events were used in the event scenario development and event tree analysis (CRWMS M&O, 1998, 2000b). DOE should provide a roadmap linking the operational hazards and initiating events identified in the original hazards analysis to all the reports where this information is subsequently used.

The DOE approach to categorization of event sequences for the high-temperature facility design is acceptable. Using the assumption of a 100-year operational period, the expected frequency of occurrence is greater than or equal to 10^{-2} per year for Category 1 event sequences, and it is less than 10^{-2} per year but greater than or equal to 10^{-6} per year for Category 2 event sequences. Those event sequences with an expected frequency of occurrence less than 10^{-6} per year are excluded from the safety evaluation, and DOE defines these classes of event sequences as beyond design basis events (DOE, 2001a).

The DOE approach to categorization of event sequences in low-temperature facility design is inconsistent and unclear. For the high-and low-temperature facility design, DOE plans that handling and emplacements of waste in the facility are expected to occur for approximately a 24-year operational period. The preliminary preclosure safety evaluation and safety analysis (DOE, 2001a) use an assumption of a 100-year preclosure period, which DOE argues bounds the duration of facility operations and conservatively classifies Category 1 and Category 2 event sequences (DOE, 2001c). DOE contends that the extension of the preclosure period to 325 years for low-temperature facility design does not significantly change the operational period and, therefore, does not potentially impact the screening of events arising from surface and subsurface facility operations. Contrary to this argument, DOE calculates different categorization of the frequency thresholds of 3.1×10^{-3} per year for Category 1 event sequences, and the frequency threshold is 3.1×10^{-7} per year for Category 2 event sequences (DOE, 2001a) for the low-temperature facility design; that includes an implicit assumption of a 325-year preclosure period that is inconsistent with the bounding assumptions of a 100-year preclosure period. DOE should clearly present information on the categorization of the event sequences for the low-temperature facility design in a form consistent with the event sequence definition in 10 CFR 63.2 presented in Section 2.1.4.1.

DOE has not provided adequate technical justification that the screening of event sequences on the basis of design is consistent with the 10 CFR Part 63 requirements. DOE has identified event sequences for the geologic repository operations area operations not expected to result in radiological release (DOE, 2001a, Table 5-7). The event sequences, listed in Table 5-7, can be classified as Category 1 or Category 2, however, DOE plans to rely on design features that will either prevent event sequences from occurring or prevent the release of radiological dose. The event sequences listed in Table 5-7 were excluded from Category 1 or Category 2 event sequences and were not considered in the safety assessment. Structures, systems, and components credited to prevent radiological release from the set of event sequences in Table 5-7 are disposal container/waste package, shipping cask, canisters, bridge crane and lifting fixtures, waste package lifting systems, and so on. In this regard, NRC stated that DOE should take into account the staff views and comments on this issue as quoted here:[4,5]

> DOE can screen [preclosure design basis events] based on a proposed design concept [that is] consistent with overall risk-informed performance-based philosophy in … [10 CFR] Part 63. Screening can be based on either: (i) probability, or (ii) consequences.
>
> DOE will need to demonstrate that the particular design feature can perform its intended mitigation function over the time period of regulatory interest.
>
> For supporting screening arguments, probability values for component failure or events potentially leading to the failure of the design feature, range, and distributions or relevant variables and/or boundary assumptions should be: technically defensible, and account for uncertainty and variability. [Similarly, screening by consequence should be technically defensible and account for uncertainty and variability in the parameters.]

The NRC position on events screened out by design was discussed at the DOE and NRC technical exchange.[6] DOE stated it would screen preclosure design basis events based on design features that reduce either probability or consequences consistent with the overall risk-informed, performance-based philosophy in 10 CFR Part 63. DOE further stated that the screening of design basis events will be defensible and the uncertainties will be addressed to the extent they may impact either categorization or consequences of the potential design basis events.

2.1.4.4 Status and Path Forward

The status on the closure of identification of event sequences is given in Table 2.1.4-1. There are two items pertaining to this preclosure topic. The staff review of DOE preclosure safety analysis, which is based on the preliminary design, is progressing. Limited concerns of a general nature on the first item, Justification for Methodology and Assumptions, were discussed

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[4]Lee, M. “FEP Screening Methodology: NRC Staff Views and Comments.” *Presentation (May 14) at Summary Highlights of U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration—Features, Events and Processes, May 15–17, 2001*. Attachment 5. Washington, DC: NRC. 2001.

[5]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration—Features, Events, and Processes (May 15–17, 2001).” Letter (May 30) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[6]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001).” Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

at the first DOE and NRC Technical Exchange on Preclosure Safety.[7] The second item was not discussed at the first DOE and NRC technical exchange.[8] The staff review on this preclosure topic will continue. Concerns with both items will be discussed in future technical exchanges.

Table 2.1.4-1. Summary of Resolution Status of Identification of Event Sequences Preclosure Topic

Preclosure Items	Status	Related Agreements	Comments
Justification for Methodology and Assumptions	Pending	None*	Staff Review Incomplete
Identification of Category 1 and 2 Event Sequences	Pending	†	Staff Review Incomplete

*Limited general concerns were discussed in the first DOE and NRC Technical Exchange and Management Meeting on Preclosure Safety, July 24–26, 2001, Las Vegas, Nevada. No agreements were reached.
†Not discussed at the first DOE and NRC Technical Exchange on Preclosure Safety.

2.1.4.5 References

CRWMS M&O. "DBE/Scenario Analysis for Preclosure Repository Subsurface Facilities." BCA000000–01717–00017. Revision 00. Las Vegas, Nevada: CRWMS M&O. 1997a.

———. "Waste Package Design Basis Events." BBA000000–01717–0200–00037. Revision 00. Las Vegas, Nevada: CRWMS M&O. 1997b.

———. "Preliminary Preclosure Design Basis Event Calculations for the Monitored Geologic Repository." BC000000–01717–0210–00001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 1998.

———. "Repository Surface Design Engineering Files Report." BCB000000–01717–5705–00009. Revision 03. Las Vegas, Nevada: CRWMS M&O. 1999a.

———. "Monitored Geologic Repository Internal Hazards Analysis." ANL–MGR–SE–000003. Revision 00. Las Vegas, Nevada: CRWMS M&O. 1999b.

[7]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[8]Ibid.

———. "Subsurface Transporter Safety Systems Analysis." ANL–WER–ME–000001. Revision 01. Las Vegas, Nevada: CRWMS M&O. 1999c.

———. "Design Basis Event Frequency and Dose Calculation for Site Recommendation." CAL–WHS–SE–000001. Revision 01. Las Vegas, Nevada: CRWMS M&O: 2000a.

———. "Preclosure Design Basis Events Related to Waste Packages." ANL–MGR–MD–000012. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000b.

DOE. "Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation." TDR–MGR–SE–000009. Revision 00 ICN 03. Las Vegas, Nevada: CRWMS M&O. 2001a.

———. "Yucca Mountain Science and Engineering Report Technical Information Site Recommendation Consideration." DOE/RW–0539. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 2001b.

———. "Yucca Mountain Preliminary Site Suitability Evaluation." DOE/RW–0540. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 2001c.

NRC. NUREG/CR–2300, "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants, Final Report—Vol. 1 and Vol. 2." Washington, DC: NRC. January 1983.

———. NUREG–1804, "Yucca Mountain Review Plan—Draft Report for Comment." Revision 2. Washington, DC: NRC. March 2002.

System Safety Society. *System Safety Analysis Handbook—A Source Book for Safety Practitioners.* 2nd Edition. Albuquerque, New Mexico: System Safety Society. 1997.

2.1.5 Consequence Analyses

2.1.5.1 Consequence Analysis Methodology and Demonstration That the Design Meets 10 CFR Parts 20 and 63 Numerical Radiation Protection Requirements for Normal Operations and Category 1 Event Sequences

2.1.5.1.1 Description of Issue

The consequence analyses assess the potential radiological doses to members of the public and on-site workers during the preclosure period from operations in the surface and subsurface facilities of the geologic repository operations area. The preclosure analyses consider potential radiological consequences resulting from normal operations, Category 1 event sequences, and Category 2 event sequences. Consequences are not required to be analyzed for those event sequences with frequencies less than the minimum frequency for categorization.

This section provides a review of the consequence analyses from normal operations and Category 1 event sequences contained within the DOE documentation for preclosure. The preclosure safety strategy is presented in CRWMS M&O (2000a). The DOE description of the preclosure consequence analyses, the dose calculation methodology, and the results are documented in DOE (2001a). CRWMS M&O (2000b) provides detailed documentation of the preclosure dose calculation. Portions of additional documentation were reviewed to the extent that they contain data or analyses that support the preclosure consequence analyses.

2.1.5.1.2 Importance to Safety

One aspect of a risk-informed NRC review was to determine how this issue is related to the DOE repository safety strategy during the preclosure period. The consequence analyses are critical for demonstrating compliance with the preclosure performance objectives during normal operations and Category 1 event sequences in 10 CFR 63.111(a).

2.1.5.1.3 Technical Basis

A review of the DOE consequence analyses for normal operations and Category 1 event sequences during the preclosure period is provided in the following subsections. The review is organized according to the three acceptance criteria consistent with the associated review methods and acceptance criteria in the Yucca Mountain Review Plan (NRC, 2002). The following acceptance criteria are based on meeting the requirements of 10 CFR 63.111(a)(1), (a)(2), (b)(1), (c)(1), and (c)(2), relating to consequence analysis methodology and demonstration that the design meets 10 CFR Parts 20 and 63 numerical radiation protection requirements for normal operations and Category 1 event sequences.

2.1.5.1.3.1 Hazard Consideration

DOE conducted consequence analyses for normal operations and Category 1 event sequences. The consequence analyses were performed for radiological releases corresponding to each identified Category 1 event sequence. Consequence analyses would be required for any additional event sequences identified in Sections 2.1.3, Identification of Hazard and Initiating Events, and 2.1.4, Identification of Event Sequences, of this report but not presently considered in the DOE preclosure safety analyses. The waste forms proposed for disposal in the repository are: commercial spent nuclear fuel, DOE spent nuclear fuel, Naval spent nuclear fuel, high-level waste, and DOE plutonium waste. The assemblies of commercial spent nuclear fuel will arrive at the proposed repository either as bare assemblies in a transportation cask or as canisters of assemblies within a transportation cask. DOE spent nuclear fuel, Naval spent nuclear fuel, high-level radioactive waste, and other non-commercial waste forms will arrive at the proposed repository in welded disposable canisters within a transportation cask.

Detailed consequence analyses were presented for commercial spent nuclear fuel assemblies-handling scenarios. The analysis of a breach of a disposable commercial spent fuel canister has not yet been performed (CRWMS M&O, 2000b). Additional consequence analyses were not performed for the other noncommercial waste forms because they are either bounded by the source term of commercial spent nuclear fuel or will not result in releases, because of preventive, mitigative, or both design features (DOE, 2001a). This assumption will continue to be evaluated as documentation on the noncommercial fuel waste forms and mitigative design features becomes available. Except for the Naval canisters and the disposable commercial spent nuclear fuel canisters, canister breach is not credible based on the canister certification for the handling equipment and operational design and is not considered a categorized event sequence. Because of the robust nature of the cladding of Naval spent nuclear fuel, credible impacts will not breach the cladding of Naval spent nuclear fuel. The validity of this assumption has not yet been assessed. Therefore, the Naval canisters are not certified to withstand credible impacts. To support this, off-site consequence analyses were performed for the release of activated corrosion products on Naval spent nuclear fuel (CRWMS M&O, 1999). Without taking credit for high-efficiency particulate air filters in the ventilation system, off-site doses from the breach of a disposable canister containing Naval spent fuel were determined to be below the regulatory limits in 10 CFR 63.111. For this reason, Naval canisters and disposable commercial spent fuel canisters are not certified to withstand all credible handling events (CRWMS M&O, 2000a).

The consequence analyses consider doses to the public offsite, but not to on-site workers. 10 CFR 63.111(a)(1) requires that the repository operations shall meet the requirements of 10 CFR Part 20. 10 CFR Part 20 stipulates dose limits for workers in Subpart C and for members of the public in Subpart D including the as low as is reasonably achievable requirements of 10 CFR 20.1101. The on-site consequences to workers should also be determined for a breach of Naval canisters and disposable commercial spent nuclear fuel canisters without high-efficiency particulate air filtration. This issue has not been previously raised with DOE. It is important to note that the consequence analyses for a breach of a disposable commercial spent nuclear fuel canister have not yet been performed (CRWMS M&O, 2000a) and credit should not be taken for these canisters to withstand all credible handling events unless the analysis results support this assertion.

DOE (2001a, Section 5.3.5.3) states, "… administrative controls will be in place to evacuate any members of the public who could potentially be located within the Yucca Mountain Project Withdrawal Area but outside of the Preclosure Controlled Area Boundary (Figure 5-4) following a Category 2 [Design Basis Event, also referred to as an event sequence]." Because evacuation after a Category 1 event sequence has not been addressed, there is a possibility that the public could be present within the 11-km [6.8-mi] withdrawal area boundary. If evacuation plans are not established for Category 1 event sequences, members of the public could be present within the 11-km [6.8-mi] withdrawal area boundary, which would require that the Category 1 consequence analyses consider these individuals {i.e., dose calculations for members of the public within 11 km [6.8 mi]}. DOE should justify whether an evacuation plan for members of the public is needed after a Category 1 event sequence. Considering that members of the public could be located within the withdrawal area boundary, DOE should provide additional justification for the selection of the 11-km [6.8-mi] distance to the withdrawal area boundary as the closest point that any member of the public could be located at the time of a postulated radiological release. This issue has not been previously raised with DOE.

2.1.5.1.3.2 Methods and Assumptions

The preclosure safety analysis is sensitive to what input parameters are used in the consequence calculations. In analyzing radiation doses from Category 1 event sequences, the repository safety strategy (CRWMS M&O, 2000a) proposes to use calculation input parameters, such as atmospheric dispersion factors, breathing rates, ingestion rates, and waste characteristics based on long-term average data. These long-term average data are appropriate for evaluating the chronic releases from normal operations of the surface and subsurface facilities. Releases from Category 1 event sequences will occur for a period of time that is short with respect to time for which the parameter data were averaged (i.e., not chronic). Because 10 CFR 63.111(a)(2) refers to a preclosure standard in 10 CFR 63.204 that is an annual dose to any real member of the public from Category 1 event sequences and normal operations that must not be exceeded in any year, parameters based on appropriate short-term data should be used to enable a demonstration with reasonable assurance that the parameters used in the calculations are appropriate for the scenario used. DOE should use short-term data for atmospheric dispersion and other parameters for which long-term data are inappropriate or provide a technical justification for the appropriateness of using long-term data for the dose calculations. This issue has not been previously raised with DOE.

CRWMS M&O (2000b, Attachment IV, Section 2.2) stated that the dose coefficients for external exposure are based on soil contaminated to a depth of 15 cm [5.9 in.], which may underestimate the external doses from increased self-attenuation by the contaminated soil, compared with a thinner contamination layer. Each airborne release would result in surface depositions of radionuclides, which slowly migrate deeper into the soil with time. Attachment IV (CRWMS M&O, 2000b) presents the dose calculation methodology for Category 1 event sequences, for which an exposure time of 1 year is assumed. Studies of the depth distribution of radionuclides in soil for depositions less than 1 year show that most of the radionuclide inventory is contained within the upper few centimeters of soil (International Commission on Radiation Units and Measurements, 1994). Although the deeper contaminated layer would seem appropriate for plowed fields, a thinner contaminated layer should be considered for the external dose calculations. It should be noted that selection of a normalized dose conversion (Sv yr^{-1} per Bq m^{-3}) based on a 15-cm [5.9-in.] contaminated layer in EPA (1993) is acceptable and thought to be conservative because a thicker contaminated layer adds to the source term and increases the normalized dose conversion (Sv yr^{-1} per Bq m^{-3}). The uniform distribution assumption, however, would inappropriately reduce the activity concentration (Bq m^{-3}) and result in an underestimation of the external dose. It is unclear if the expected activity of radionuclides deposited on the soil was distributed uniformly to a depth of 15 cm [5.9 in.]. This issue has not been previously raised with DOE.

The Yucca Mountain Review Plan (NRC, 2002) includes guidance on calculations of on-site and off-site direct exposures during normal operations and Category 1 event sequences. For completeness, direct exposure calculations are required for external radiation sources, whether related to the releases of radioactive material or not. DOE calculates direct exposure doses resulting from released radioactive material. The DOE consequence analyses, however, do not include direct exposure dose calculations from external sources not related to released radioactive material; however, this information should be included. This issue has not been previously raised with DOE. In addition, DOE should describe how direct radiation was considered in the facility design process.

The definition and use of the local deposition factor are conflicting. On page 11 (CRWMS M&O, 2000b), the local deposition factor is described as "… the fraction of the [airborne release fraction] that is deposited locally within the [Waste Handling Building]…." From this definition, a local deposition factor value of 1 would be equal to 100 percent of the material released being deposited in the Waste Handling Building and would imply no release from the Waste Handling Building. The local deposition factor was set at a value equal to1 to maximize releases from the Waste Handling Building as part of Assumption 3.20 (CRWMS M&O, 2000b), which is inconsistent with its definition. Furthermore, Eq. (11) (CRWMS M&O, 2000b) calculates the total release fraction to the environment and uses the local deposition factor directly to calculate the release fraction instead of one minus the local deposition factor. Staff suggest either (i) defining the local deposition factor as a release or leakage factor rather than a deposition factor or (ii) modifying Eq. (11) and Assumption 3.20 to be consistent with the actual definition of the local deposition factor. This issue has not been previously raised with DOE.

2.1.5.1.3.3 Compliance with Regulatory Requirements

Although the DOE approach for demonstrating compliance applies a frequency weighting to the doses for Category 1 event sequences, the approach does not consider multiple Category 1 event sequences occurring in a single year. 10 CFR 63.111(a)(2) refers to a preclosure standard, which is an annual dose to any real member of the public from Category 1 event sequences and normal operations, that shall not be exceeded in any year. Therefore, conditional or event doses for the Category 1 event sequences would be required to assess whether credible combinations of multiple Category 1 event sequences occurring in a single year could exceed the annual dose limit. DOE should present a table of the event doses for each of the Category 1 event sequences and ensure that each Category 1 event sequence does not exceed the limits specified in 10 CFR 63.111(a). The staff communicated these issues to DOE at the Technical Exchange and Management Meeting for Preclosure Safety,[1] and DOE agreed to demonstrate the dose from any single Category 1 event sequence will not exceed the regulatory limit.

Because 10 CFR 63.111(a) and 63.204 limit the annual dose to a real member of the public from Category 1 event sequences and normal operations, DOE should present analyses that demonstrate that combinations of multiple Category 1 event sequences occur within a single year. Only those combinations with a probability equal to or greater than 0.01 (the frequency limit specified by 10 CFR Part 63, which event sequences correspond to Category 1 event sequences) should be considered. This issue was discussed at the Technical Exchange and Management Meeting for Preclosure Safety.[2] DOE proposed a general path forward, but details were not made available at the meeting.

The DOE consequence analyses for workers from Category 1 event sequences are incomplete. Occupational doses were calculated only for a noninvolved worker at an outside distance of 100 m [328 ft] (CRWMS M&O, 2000b). Although DOE has only considered noninvolved workers outside, the Waste Handling Building floor plan (DOE, 2001b) clearly indicates worker involvement inside the building located in the operating galleries by the side of the canister transfer and assembly transfer areas. DOE (2001a, Section 5.3.6.2) asserts, “the potential radiological exposure during an accident for workers located less than 100 m [328 ft] from a radiological release (e.g., inside the Waste Handling Building) is expected to be minimal.” The higher radionuclide air concentrations and minimal dilution inside the building, as well as gravitational settling within the building and its ventilation system, however, have not been addressed and could result in higher worker doses. Analyses for involved workers inside the Waste Handling Building should also be provided for Category 1 event sequences (i) to ensure that the occupational limits of 10 CFR Part 20 can be met and (ii) for application of the QL–3 risk measure of a 0.05-Sv [5-rem] worker dose. Doses to workers inside the Waste Handling Building for gaseous releases from Category 1 event sequences in the pool have also not been addressed. These issues have not been previously raised with DOE.

CRWMS M&O (2000b) presents doses for a worker at a distance of 100 m [328 ft] from the routine releases (CRWMS M&O, 2000b, Attachment V). To demonstrate the performance requirements of 10 CFR Parts 20 and 63 have been met for workers inside the emplacement drifts, DOE should assess or, at a minimum, discuss how well the subsurface ventilation reduces the higher radionuclide concentrations expected within the drifts because of less radioactive decay and dilution. This issue has not been previously raised with DOE.

DOE (2001a, Section 5.3.5.3) report states that staff located on the Nevada Test Site and Nellis Air Force Range are government workers on government property, subject to evacuation if required, and, therefore, not considered part of the public. 10 CFR 20.1003 defines occupational dose as “… the dose received by an individual in the course of employment in

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[1]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001).” Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[2]Ibid.

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." 10 CFR 20.1003 defines member of the public as any individual except when that individual is receiving an occupational dose. It is acknowledged that administrative controls should be more effective for individuals on government property compared with those not on government property. Unless the assigned duties of all staff located on the Nevada Test Site and Nellis Air Force Range involve exposure to radiation or to radioactive material, however, those staff should be considered members of the public. If the duties of those workers are deemed to involve exposure to radiation, the survey and monitoring requirements of Subpart F to 10 CFR Part 20 and the reporting requirements of Subpart M to 10 CFR Part 20 must be complied with. Consequently, staff located on the Nevada Test Site and Nellis Air Force Range should be treated as members of the public unless trained, monitored, and protected by an established radiation protection program, or DOE should provide additional information about the classification of government workers as radiation workers in 10 CFR Part 20. This issue has not been previously raised with DOE.

2.1.5.1.4 Status and Path Forward

The consequence analyses for normal operations and Category 1 event sequences during the preclosure period are considered pending by the NRC staff. Further information will be required at the time of any license application.

At the first Technical Exchange and Management Meeting for Pre-Closure Safety,[3] the NRC staff agreed with the DOE general methodology for consequence analyses. Because the meeting focused on general methodologies, many specific comments were not raised at the meeting. The status of issue closure in the preclosure safety area was not discussed. Nor were specific agreements on the consequence analyses reached at that meeting. Table 2.1.5-1 provides the status of the preclosure consequence analyses for normal operations and Category 1 event sequences.

The preceding review also indicates that relevant acceptance criteria for the preclosure consequence analyses for normal operations and Category 1 event sequences from the Yucca Mountain Review Plan (NRC, 2002) have not been met by the proposed DOE approach.

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[3]Reamer, C.W. "U.S. Nuclear regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

Table 2.1.5-1. Summary of Resolution Status of Consequence Analyses for Normal Operations and Category 1 Event Sequences Preclosure Topic

Preclosure Items	Status	Related Agreements*	Comments
Hazard Consideration	Pending	None	Staff Review Incomplete
Methods and Assumptions	Pending	None	Staff Review Incomplete
Compliance with Regulatory Requirements	Pending	None	Staff Review Complete

*Limited general concerns were discussed in the first DOE and NRC Technical Exchange and Management Meeting on Pre-Closure Safety, July 24–26, 2001, Las Vegas, Nevada. No agreements were reached.

2.1.5.1.5 References

CRWMS M&O. "DOE SNF DBE Offsite Dose Calculations." CAL–WPS–SE–000004. Revision 00. Las Vegas, Nevada: CRWMS M&O. 1999.

———. "Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations." TDR–WIS–RL–000001. Revision 04 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000a.

———. "Design Basis Event Frequency and Dose Calculation for Site Recommendation." CAL–WHS–SE–000001. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000b.

DOE. "Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation." TDR–MGR–SE–000009. Revision 00 ICN 03. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 2001a.

———. "Yucca Mountain Science and Engineering Report Technical Information Site Recommendation Consideration." DOE/RW–0539. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 2001b.

EPA. "External Exposure of Radionuclides in Air, Water, and Soil." Federal Guidance Report No. 12. EPA402–R–93–081. Washington, DC: EPA. 1993.

International Commission on Radiation Units and Measurements. "Gamma-Ray Spectrometry in the Environment." Report 53. Bethesda, Maryland: International Commission on Radiation Units and Measurements. 1994.

NRC. NUREG–1804, "Yucca Mountain Review Plan—Draft Report for Comment." Revision 2. Washington, DC: NRC. March 2002.

2.1.5.2 Demonstration That the Design Meets 10 CFR Part 63 Numerical Radiation Protection Requirements for Category 2 Event Sequences

2.1.5.2.1 Description of Issue

This section provides a review of the consequence analyses for Category 2 event sequences contained within the DOE documentation for preclosure. The preclosure safety strategy is presented in CRWMS M&O (2000a). The DOE description of the preclosure consequence analyses and the dose calculation methodology and its results are documented in DOE (2001). CRWMS M&O (2000b) provides detailed documentation of the preclosure dose calculation. Portions of additional documentation were reviewed to the extent they contain data or analyses that support the preclosure consequence analyses.

2.1.5.2.2 Importance to Safety

One aspect of risk-informing the NRC review was to determine how this issue is related to the DOE preclosure repository safety strategy. The consequence analyses are critical for demonstrating compliance with the preclosure performance objectives resulting from Category 2 event sequences in 10 CFR 63.111(b).

2.1.5.2.3 Technical Basis

A review of the DOE consequence analyses for Category 2 event sequences during the preclosure period is provided in the following subsections. The review is organized according to the three acceptance criteria consistent with the associated review methods and acceptance criteria in the Yucca Mountain Review Plan (NRC, 2002). The following acceptance criteria are based on the requirements of 10 CFR 63.111(b)(2) and (c) related to the design complying with 10 CFR Part 63 numerical radiation protection requirements for Category 2 event sequences.

2.1.5.2.3.1 Hazard Consideration

The staff evaluation of the hazard event sequences for Category 2 event sequences is contained in Sections 2.1.4 and 2.1.3 of this report. Consequence analyses would be required for additional Category 2 event sequences identified in those sections. Based on the available documentation, staff have not identified other issues in this acceptance criterion.

2.1.5.2.3.2 Methods and Assumptions

An evacuation plan has not been described, but credit is taken for evacuating off-site members of the public, after a Category 2 event sequence by assuming a 2-hour occupancy time, in DOE (2001). Credit for evacuation is premature until a commitment has been made to develop an evacuation plan for off-site members of the public following a Category 2 event sequence. This issue has not been previously raised with DOE.

CRWMS M&O (2000b, Section 5.2.7) used incorrect bounding estimates for Co-60 crud. Based on a 33-GWd/MTU burnup and 3.2-percent enrichment, these Co-60 crud activities per fuel assembly surface area do not qualify as bounding estimates for the maximum pressurized water reactor and boiling water reactor fuel characteristics with a 75-GWd/MTU burnup and 5-percent enrichment. This issue has not been previously raised with DOE.

Failed fuel (e.g., with cladding damage, debris, or pieces of fuel present) is to be placed in disposable single element canisters. The source term from failed fuel was assumed to be bounded by the radiological consequences from commercial spent nuclear fuel. The release fraction calculations do not consider failed fuel (CRWMS M&O, 1999), which may have higher particulate release fraction and result in a larger released source term. The potentially higher particulate release fractions from failed fuel should be considered to adequately support the argument that failed fuel is bounded by commercial spent nuclear fuel. This issue has not been previously raised with DOE.

The Yucca Mountain Review Plan (NRC, 2002) includes guidance on calculations of off-site dose from direct exposure after Category 2 event sequences. For completeness, direct exposure calculations are required for external radiation sources, whether related to the releases of radioactive material or not. DOE calculates direct exposure doses resulting from released radioactive material. The DOE consequence analyses, however, do not include direct exposure dose calculations from external sources not related to released radioactive material. This issue has not been previously raised with DOE.

2.1.5.2.3.3 Compliance with Regulatory Requirements

Based on available documentation, the staff have not identified any issues in this acceptance criterion and find the DOE approach acceptable.

2.1.5.2.4 Status and Path Forward

The consequence analyses for Category 2 event sequences during the preclosure period are considered pending by the NRC staff. Further information will be required at the time of any license application.

At the first Technical Exchange and Management Meeting for Pre-Closure Safety,[4] NRC staff agreed with the DOE general methodology for consequence analyses. Because the meeting focused on general methodologies, many specific comments were not raised at the meeting. The status of issue closure in the preclosure safety area was not discussed. Nor were specific agreements on the consequence analyses reached at that meeting. Table 2.1.5-2 provides the status of the preclosure consequence analyses for Category 2 event sequences.

The preceding review also indicates that relevant acceptance criteria for the preclosure consequence analyses for Category 2 event sequences from the Yucca Mountain Review Plan (NRC, 2002) have not been met by the proposed DOE approach.

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[4]Reamer, C.W.. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

Table 2.1.5-2. Summary of Resolution Status of Consequence Analyses for Category 2 Event Sequences Preclosure Topic			
Preclosure Items	**Status**	**Related Agreements***	**Comments**
Hazard Consideration	Pending	None	Staff Review Incomplete
Methods and Assumptions	Pending	None	Staff Review Incomplete
Compliance with Regulatory Requirements	Pending	None	Staff Review Incomplete
*Limited general concerns were discussed in the first DOE and NRC Technical Exchange and Management Meeting on Preclosure Safety, July 24–26, 2001, Las Vegas, Nevada. No agreements were reached.			

2.1.5.2.5 References

CRWMS M&O. "Commercial SNF Accident Release Fractions." ANL–WHS–SE–000002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 1999.

———. "Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations." TDR–WIS–RL–000001. Revision 04 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000a.

———. "Design Basis Event Frequency and Dose Calculation for Site Recommendation." CAL–WHS–SE–000001. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000b.

DOE. "Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation." TDR–MGR–SE–000009. Revision 00 ICN 03. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 2001.

NRC. NUREG–1804, "Yucca Mountain Review Plan—Draft Report for Comment." Revision 2. Washington, DC: NRC. March 2002.

2.1.6 Identification of Structures, Systems, and Components Important to Safety; Safety Controls; and Measures to Ensure Availability of the Safety Systems

2.1.6.1 Description of Issue

Consistent with the requirements in 10 CFR 63.112, DOE is required to conduct a preclosure safety analysis of the proposed geologic repository operations area and identify the structures, systems, and components important to safety. Structures, systems, and components important to safety are defined in 10 CFR 63.2 as those engineered features whose functions are to (i) provide reasonable assurance that high-level waste can be received, handled, packaged, stored, emplaced, and retrieved without exceeding the requirements of 10 CFR 63.111(b)(1) for Category 1 event sequences or (ii) prevent or mitigate Category 2 event sequences that could result in radiological exposures exceeding the values specified in 10 CFR 63.111(b)(2) to any individual located on or beyond any point on the boundary of the site. As defined in 10 CFR 63.2, Category 1 event sequences are those expected to occur one or more times before permanent closure of the geologic repository operations area, and Category 2 event sequences are those sequences with at least 1 chance in 10,000 of occurring before permanent closure.

The preclosure safety analysis of the geologic repository operations area is defined in 10 CFR 63.2 as a systematic examination of the site; the design; and the potential hazards, initiating events, and event sequences and their consequences (e.g., radiological exposures to workers and the public). The preclosure safety analysis includes an analysis of the structures, systems, and components to identify those that are important to safety. The preclosure safety analysis also identifies and describes the controls relied on to prevent potential event sequences from occurring or to mitigate their consequences and identifies measures taken to ensure the availability of the safety systems. As a part of a potential license application, 10 CFR 63.142(c)(1) requires that DOE shall identify structures, systems, and components identified by the quality assurance program (e.g., structures, systems, and components important to safety and waste isolation). Additionally, 10 CFR 63.142(c)(1) states that a quality assurance program must control activities affecting the quality of the identified structures, systems, and components to an extent consistent with their importance to safety. Quality assurance can be accomplished by categorizing structures, systems, and components based on risk insight gained from the preclosure safety analysis.

Using Section 4.1.1.3, Identification of Hazards and Initiating Events; Section 4.1.1.4, Identification of Event Sequences; and Section 4.1.1.5, Consequence Analyses in NRC (2002), staff review will verify that analysis and identification of structures, systems, and components for the geologic repository operations area used the results of the iterative preclosure safety analysis and confirmed that structures, systems, and components are identified as important to safety according to the definition specified in 10 CFR 63.2. This section of this report provides the preliminary review of the identification of structures, systems, and components important to safety; safety controls; and measures to ensure availability of the safety systems based on review of DOE (2001a) and a selected number of classification reports (CRWMS M&O,

1999a,b, 2000a). The July 24–26, 2001, DOE and NRC Preclosure Technical Exchange[1] concentrated on the methodology for identifying structures, systems, and components important to safety and the risk-significance categorization process; two agreements were reached. Staff will continue to review additional DOE reports and develop a comprehensive list of concerns relating to the identification of structures, systems, and components important to safety.

2.1.6.2 Importance to Safety

The identification and classification of structures, systems, and components important to safety are necessary to protect the health and safety of the public and facility workers. As required in 10 CFR Part 63, the preclosure safety analysis must be used to identify structures, systems, and components important to safety and demonstrate compliance with the performance objectives contained in 10 CFR 63.111. Structures, systems, and components important to safety must be identified based on their capabilities to prevent or mitigate potential event sequences that have the potential to exceed the performance objectives for normal operations and Category 1 event sequences and to prevent or mitigate the dose consequence of Category 2 event sequences. DOE presented a preliminary list of structures, systems, and components determined to be important to safety (DOE, 2000, 2001a). This preliminary listing of structures, systems, and components was categorized according to their importance to safety. DOE intends to use the classification of structures, systems, and components to focus on the level of design details to be provided in the license application and the application of quality assurance controls through a graded quality assurance program, as required by 10 CFR 63.142(c)(1). Inaccurate identification or misclassification of structures, systems, and components important to safety has the potential to affect adversely preclosure repository safety.

2.1.6.3 Technical Basis

In compliance with 10 CFR 63.112(e), an analysis of the performance of structures, systems, and components is required to identify those structures, systems, and components important to safety. This analysis identifies and describes the controls relied on to limit or prevent potential event sequences or to mitigate their consequences. This analysis also identifies measures taken to ensure the availability of safety systems. The quality assurance program specified in 10 CFR 63.142(c)(1) controls activities affecting the quality of the identified structures, systems, and components to an extent consistent with their importance to safety. DOE proposes using the preclosure safety analysis to identify those structures, systems, and components important to safety and to categorize them using a risk-informed categorization process. The DOE approach to the risk-significance categorization, which is still evolving, has been described in several documents (DOE, 2001a–c; CRWMS M&O, 1999c, 2000b). The classification analysis evaluates the structures, systems, and components using a quality assurance procedure QAP–2–3 (CRWMS M&O, 1999c) to categorize a particular item based on the criteria shown in

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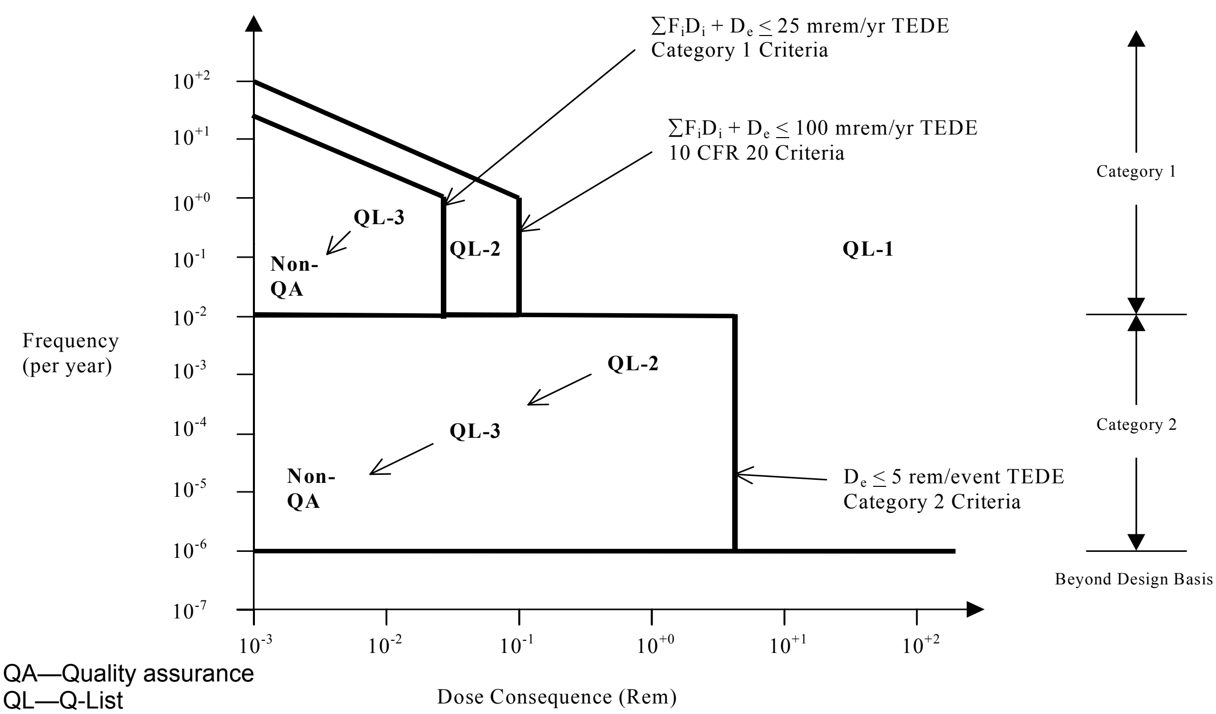
[1]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

Figure 2.1.6-1 (CRWMS M&O, 2000b). The Categories 1 and 2 frequency limits shown in Figure 2.1.6-1 are based on the assumption that the preclosure period is 100 years. The DOE categorization process screens the structures, systems, and components important to safety or waste isolation into three quality levels (DOE, 2001a): Quality Level 1 items, considered to be of high safety significance, have direct impact on worker and public health and safety; Quality Level 2 items, considered to be of low safety significance, have limited or indirect impact on worker and public health and safety; and Quality Level 3 items, to have minor impact on public or worker safety, include defense-in-depth design features intended to keep doses as low as reasonably achievable. The structures, systems, and components that do not meet any of the definitions for Quality Levels 1, 2, or 3 have been classified as conventional quality. Staff review of the DOE proposed classification process is discussed in Section 2.1.6.3.3.

Based on the preliminary design of the geologic repository operations area, DOE (2000) compiled a Q-List consisting of 185 structures, systems, and components. The selection of structures, systems, and components in the Q-List is based on the system design and functions established in system description documents cited in DOE (2000). The structures, systems, and components were further categorized as 17 Quality Level 1 items, 45 Quality Level 2 items, 19 Quality Level 3 items, and 104 conventional quality items. The categorization of each item is based on classification analyses documented in reports cited in DOE (2000). DOE also provided a list of structures, systems, and components for each category in Tables 4-1, 4-2, and 4-3 in DOE (2001a). DOE intends to update the Q-List as the design of the geologic operations area develops and evolves.



QA—Quality assurance
QL—Q-List
TEDE—Total effective dose equivalent

Figure 2.1.6-1. DOE Preclosure Classification Criteria (CRWMS M&O, 2000b)

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The NRC staff developed a position paper[2] on an acceptable approach to risk-significance categorization of structures, systems, and components important to safety for the proposed geologic operations area. The paper discusses the governing regulation and applicable policy and guidance and develops general acceptance criteria based on this information. Further, it discusses the DOE-proposed approach to risk-significance categorization and evaluates it against the general acceptance criteria, governing regulation, and applicable policy and guidance. This paper also summarizes the staff position regarding the DOE-proposed approach to risk-significance categorization and identifies potential concerns resulting from this review.

This section is organized according to the three acceptance criteria consistent with the associated review methods and acceptance criteria in Section 4.1.1.6 of NRC (2002). The following acceptance criteria are based on meeting the requirements of 10 CFR 63.112(e) related to the identifying structures, systems, and components important to safety and 10 CFR 63.142(c)(1) related to categorizing the structures, systems, and components.

2.1.6.3.1 List of Structures, Systems, and Components Identified as Important to Safety Based on Preclosure Safety Analysis

This section verifies that the iterative preclosure safety analysis (identification of hazards and initiating events, event sequences, and consequence analysis) forms the basis for DOE identification of structures, systems, and components important to safety. This section also confirms that analyses used to identify structures, systems, and components important to safety; safety controls; and measures to ensure the availability of the safety systems include adequate consideration of all structures, systems, and components and controls that function to meet the performance objectives and that structures, systems, and components are classified as important to safety according to the definition specified in 10 CFR 63.2.

The following discussion identifies concerns associated with the DOE list of structures, systems, and components important to safety. Each of the following concerns was discussed in the DOE and NRC Technical Exchange and Management Meeting on Preclosure Safety and, agreements were reached for the resolution of each concern.[3]

The DOE schematic representation of preclosure safety analysis methodology is not consistent with the requirements of preclosure safety analysis designated in 10 CFR 63.112. The

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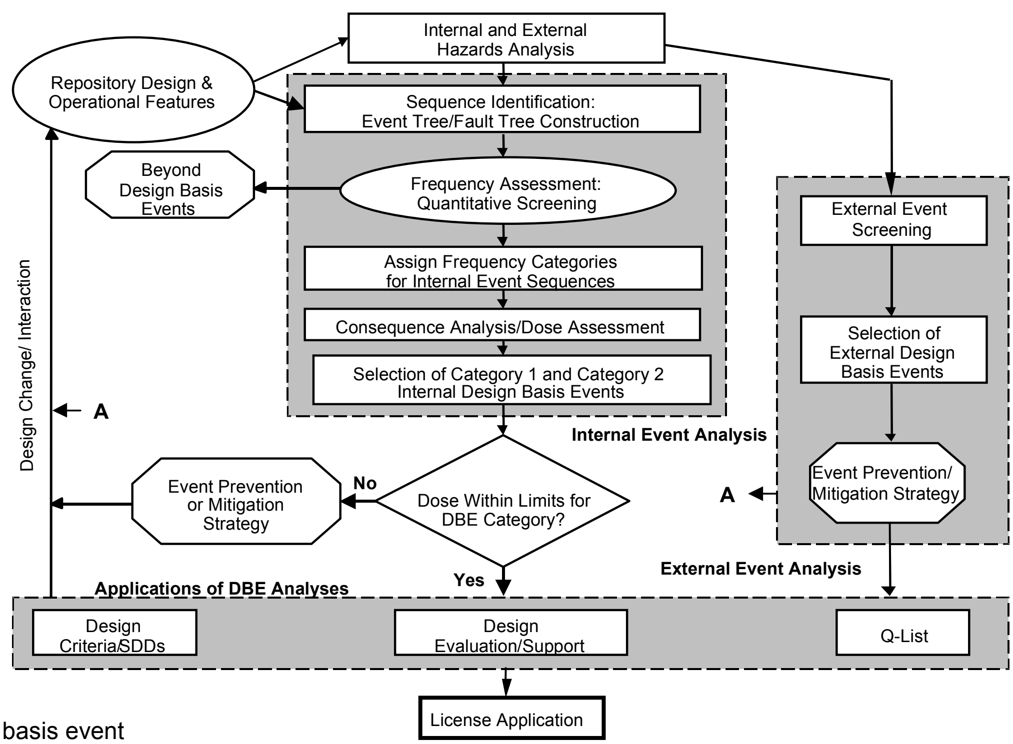
[2]Reamer, C.W. "U.S. Nuclear Regulatory Commission Staff Review of the U.S. Department of Energy's Proposed Approach to Risk Significance Categorization of Structures, Systems, and Components Important-to-Safety." Letter (September 28) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[3]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

preclosure safety analysis process, as shown in Figure 2.1.6-2, was described at the DOE and NRC technical exchange[4] and presented in several reports (DOE 2001a–c). The block diagram in Figure 2.1.6-2 explains the process of implementation of DOE preclosure safety analysis. NRC expressed concern that the naturally occurring and human-induced (external) hazard analysis and operational (internal) hazard analyses are treated separately in the preclosure safety analysis process. NRC indicated that DOE should consider integrating the hazard analyses to identify events and event sequences during facility operations that may be initiated by naturally occurring and human-induced events. DOE stated that the naturally occurring and human-induced and operational hazard analyses were coupled and were not treated



DBE—Design basis event
SDD—System description document

Figure 2.1.6-2. Overview of DOE Preclosure Safety Analysis Process[5]

[4]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001.)" Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[5]Richardson, D. "Development of the Integrated Safety Analysis for License Application." *Presentation to DOE and NRC Preclosure Issues Technical Exchange July 24–26, 2001.* Slide 4. Las Vegas, Nevada: DOE. 2001.

separately. DOE will revise the block diagram to show that the naturally occurring and human-induced hazard analysis is an integral process in the preclosure safety analysis.[6]

In its identification and classification of the structures, systems, and components important to safety for the proposed geologic repository operations area, DOE does not use the results of the preclosure safety analysis. The preclosure safety analysis required by 10 CFR 63.112 is the basis for identification of the structures, systems, and components important to safety. The DOE classification analyses consider the system design and functions of structures, systems, and components and analyze their effects on the facility safety using the screening criteria developed in a checklist in procedure QAP–2–3 (CRWMS M&O, 1999c). The DOE classification analyses, which are based on qualitative screening criteria, do not evaluate quantitative risk measures to classify the structures, systems, and components important to safety (CRWMS M&O, 1999a,b, 2000a). For example, DOE identified Categories 1 and 2 event sequences based on their frequencies of occurrence and evaluated radiological dose consequence to the members of the public from potential operational hazards in the assembly transfer system (CRWMS M&O, 1998, 2000c). DOE should use the results from the preclosure safety analysis and the classification criteria shown in Figure 2.1.6-1 in its assembly transfer system classification analysis (CRWMS M&O, 1999a). In the DOE and NRC exchange,[7] DOE stated that its current classification analysis is based on engineering judgment, project strategies, and preliminary calculations. DOE acknowledged the categorizations of structures, systems, and components that support license application need to be based on the preclosure safety analysis results. DOE stated that it is revising its risk-significance determination and categorization process to be consistent with the risk-informed requirements and will be closely linked to the preclosure safety analysis. The DOE categorization process will individually consider each event sequence frequency and consequences from the preclosure safety analysis to determine risk measures (dose after categorization). These risk measures for each of the event sequences will be compared with the revised proceduralized screening criteria (CRWMS M&O, 1999c), which will be based on the performance objectives identified in 10 CFR 63.111. In addition, a take-away analysis will be performed on each of the structures, systems, and components to establish a measure of risk associated with not taking credit for the safety function associated with individual structures systems and components Each of these structures, systems, and components will be categorized consistent with the dose mitigation importance. Finally, this iteration of the categorization process will be completed by adding the appropriate structures, systems, and components to the Q-List. DOE proposes to use a modified classification criteria diagram,[8] given in Figure 2.1.6-3 (assuming a 100-year preclosure period), that includes dose from the surface and subsurface normal operational

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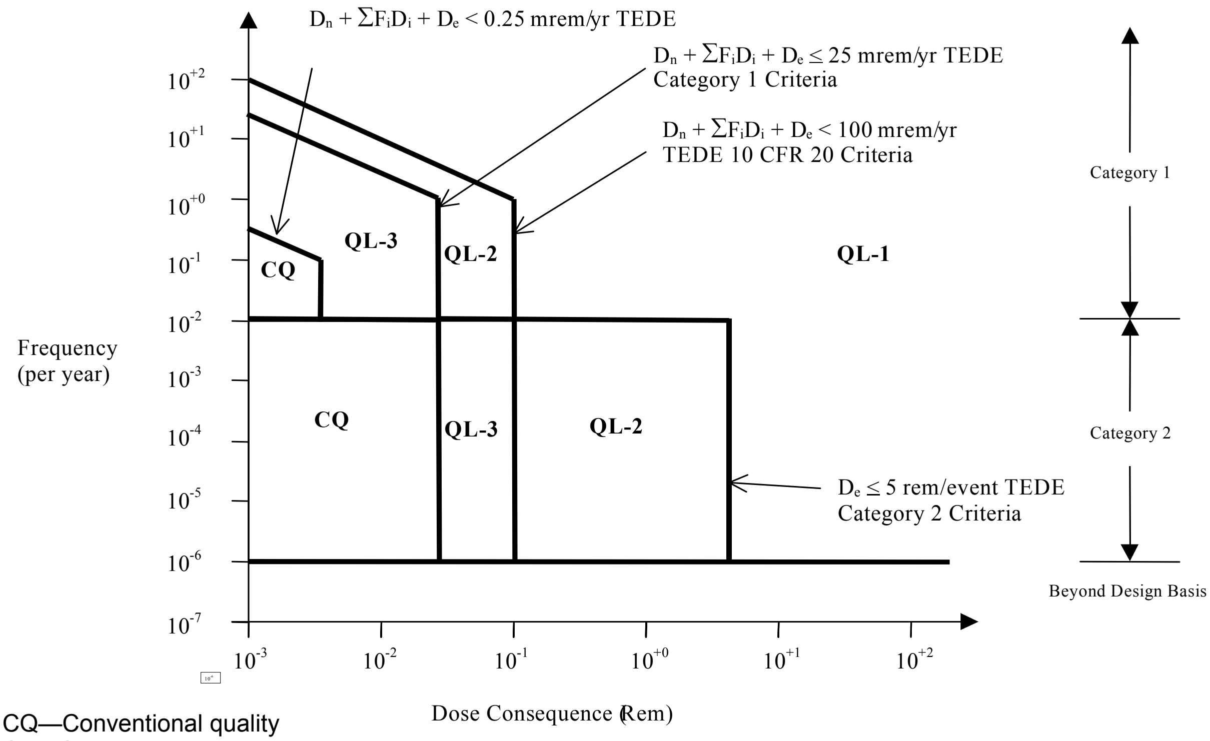
[6]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Tehnical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[7]Ibid.

[8]Gwyn, D. "Identification of SSCs Important to Safety—NRC Items 6(a) and 6(b)." *Presentation to DOE and NRC Preclosure Issues Technical Exchange July 24–26*, 2001 Vegas, Nevada: DOE. 2001.



CQ—Conventional quality
QL—Q-List
TEDE—Total effective dose equivalent

Figure 2.1.6-3. Modified DOE Preclosure Classification Criteria[9]

release in the annualized dose expression and also shows the risk measures for Quality Levels 2 and 3 and conventional quality for Categories 1 and 2 event sequences. DOE stated it is revising the procedure QAP–2–3 (CRWMS M&O, 1999c) and developing a desktop reference that will provide a clear description of the categorization process, screening criteria, and take-away analysis. Staff agreed with the overall DOE approach to categorize structures, systems, and components important to safety. Staff will review the revised procedure QAP–2–3 and the desktop reference document when it becomes available.

Although significant progress was made in the area of the quality level classification at the Technical Exchange and Management Meeting on Precosure Safety,[10] questions asked about the consequence analysis used in the proposed take-away analysis were not answered. The DOE consequence analyses used best-estimate parameter values for normal operations and Category 1 event sequences and bounding parameter values for Category 2 event sequences

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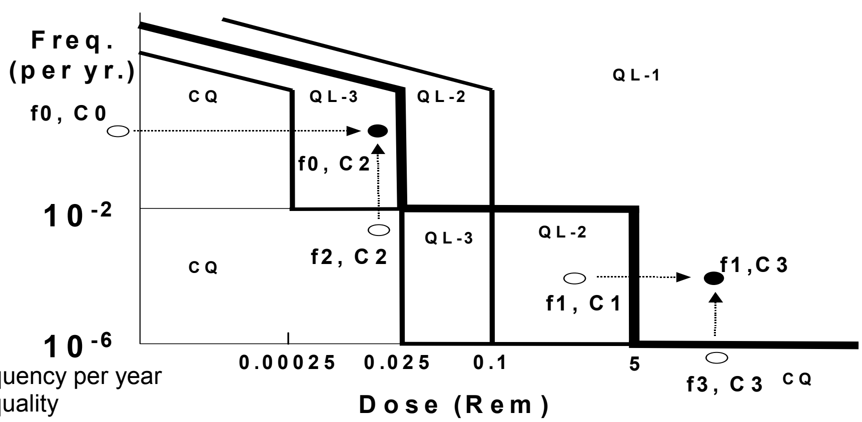
[9]Gwyn, D. "Identification of SSCs Important to Safety–NRC Items 6(a) and 6(b)." *Presentation to DOE and NRC Preclosure Issues Technical Exchange July 24–26, 2001*. Las Vegas, Nevada: DOE. 2001.

[10]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

(CRWMS M&O, 2000c). It is unclear what consequence analysis assumptions are used in those take-away analyses that result in crossing frequency thresholds for event sequence categorization. For example, the end state (f0,C0) should not map to the end state (f0,C2) when structure, system, or component A fails, as indicated on Slide 12[11] and shown in Figure 2.1.6-4, because C0 would be calculated with best-estimate parameter values, and C2 would be calculated with bounding parameter values. In addition, f3 represented a frequency below the lowest frequency for event sequence categorization for which consequences have not been calculated (CRWMS M&O, 2000c). It is, therefore, unknown what parameter value assumptions would be used for calculating the consequence denoted by C3. These issues will be discussed with DOE in a future technical exchange.

The DOE Q-List (2000) does not include all structures, systems, and components used in the geologic repository operations area. The DOE Q-List of structures, systems, and components and quality level characterization are based on the current system design described in several system description documents. 10 CFR 63.112 requires that the preclosure safety analysis of the geologic repository operations area identify those structures, systems, and components important to safety and also identify controls relied on to prevent potential event sequences or mitigate their consequences. DOE should consider all structures, systems, and components used in the geologic repository operations area to identify those important to safety. For example, shield doors and isolation doors, described in assembly transfer, canister transfer, disposal container handling and subsurface facility system description documents (CRWMS M&O 2000d–g), are not included in the Q-List. DOE should provide acceptable justification for



Freq. (per yr.)—Frequency per year
CQ—Conventional quality
QL—Q-List

Figure 2.1.6-4. Overview of the DOE Proposed Classification Process[12]

[11]Orvis, D.D. “Identification of SSCs Important to Safety—NRC Items 6(a) and 6(b): Examples.” *Presentation to DOE and NRC Preclosure Issues Technical Exchange July 24–26, 2001.* Las Vegas, Nevada: DOE. 2001.

[12]Ibid.

not identifying and classifying these structures, systems, and components that perform radiation-protection functions during surface and subsurface operations. In the preclosure safety analysis, DOE should analyze the performance of all structures, systems, and components. DOE agreed[13] with the NRC concern and stated DOE will provide adequate justification for the classification of all structures, systems, and components. DOE also stated that, at this stage, the geologic repository operations area design does not reflect all major components, and classification of the items will evolve consistent with the maturity of the design and the preclosure safety analysis. At the time of license application, the DOE Q-List will include the classifications of all major components. Staff believe this information will be adequate to review the DOE license application.

The proposed DOE approach for classification of structures, systems, or components does not account for multiple Category 1 event sequences occurring in a single year. Based on the frequencies for the Category 1 event sequences (DOE, 2001a), it can be expected that, for the entire preclosure operational period, more than one Category 1 event sequence will occur within a single year. 10 CFR Part 63 specifies an annual dose limit of 0.15 mSv [15 mrem] for members of the public. DOE proposed to classify individual structures, systems, or components for Category 1 event sequences with a take-away analysis that includes the summation of three terms:[14] (i) annual dose from normal operations of the surface and subsurface facilities; (ii) the frequency-weighted dose from all Category 1 event sequences; and (iii) the worst-case event dose from a Category 1 event sequence involving the failure of that particular structure, system, or component. In this analysis, only the value of the worst-case event dose changes for different structures, systems, and components. When determining a quality-level classification for Category 1 event sequences, DOE should consider only those combinations of multiple Category 1 event sequences expected to occur one or more times before permanent closure. For such combinations, the event doses from those particular event sequences could be summed to yield a total annual dose from Category 1 event sequences. Adequate consideration of multiple Category 1 event sequences occurring within a single year could be achieved with a take-away analysis that includes multiple terms of the worst-case event dose corresponding to the event doses for the multiple Category 1 event sequences. DOE stated it will consider combinations of Category 1 event sequences occurring in a single year when performing structure, system, and component classifications, and additional dose terms for those multiple Category 1 event sequences would be included in the quality-level classification equation. Staff agreed with the general DOE-proposed path forward.

DOE defined a structure, system, or component with a Quality Level 3 classification as one “whose failure would not significantly impact public or worker safety, including those defense-in-depth design features intended to keep radiation doses ALARA [as low as is reasonably achievable]” (CRWMS M&O, 2000b). A Quality Level 3 classification was assigned to those structures, systems, or components required to limit worker doses from normal operations and

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[13]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001).” Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[14]Gwyn, D. “Identification of SSCs Important to Safety—NRC Items 6(a) and 6(b).” *Presentation to DOE and NRC Preclosure Issues Technical Exchange July 24–26, 2001*. Las Vegas, Nevada: DOE. 2001.

Category 1 event sequences from exceeding the occupational dose limit of 10 CFR Part 20 (CRWMS M&O, 1999c). DOE provided rationale for this assignment by stating that Quality Level 3 controls are consistent with nuclear power precedent. Reliance on activity controls (e.g., worker training, radiation protection programs, and procedures) has been demonstrated to be successful in the nuclear industry. DOE takes the position that these activity controls, in combination with the Quality Level 3 controls, are more than adequate to address worker safety. Although current analyses calculate worker doses for an uninvolved worker located outside the waste-handling building at a distance of 100 m [328 ft] (CRWMS M&O, 2000c), DOE stated it plans to incorporate radiation-worker safety practices that would eventually include worker dose analyses inside the waste-handling building. With regard to nuclear power plant licensees, NRC staff stated certain quality levels are typically placed on particular structures, systems, or components (e.g., radiation monitors and reading of dosimetry badges), and DOE anticipated no problem in adhering to such NRC precedents. Staff agreed with the DOE-proposed path forward.

2.1.6.3.2 Administrative or Procedural Safety Controls Are Adequate

In compliance with 10 CFR Part 63, DOE is required to include in the list of structures, systems, and components important to safety those administrative or procedural safety controls needed to prevent event sequences or mitigate their effects. DOE (2001a) does not, however, include in the list of structures, systems, and components important to safety those administrative or procedural safety controls required for structures, systems, and components to be functional and to meet dose requirements. Further, management systems and procedures that are sufficient to ensure administrative or procedural controls function properly have not been provided. This preclosure item was not discussed at the July 24–26, 2001, DOE and NRC technical exchange.[15]

2.1.6.3.3 Risk Significance Categorization of Structures, Systems, and Components Important to Safety

The NRC staff developed a position paper[16] on risk-significance categorization of structures, systems, and components important to safety, as identified in Section 2.1.6.3 of CRWMS M&O (1999c). 10 CFR Parts 63, 20, 50, and 70 do not identify or require any specific process or methodology for the risk-significance categorization of structures, systems, and components important to safety. Further, there is no regulatory guidance or policy specifically addressing risk categorization of structures, systems, and components important to safety for a potential geologic repository operations area. NRC, however, has developed extensive direction (in the form of regulatory policy and guidance) on risk-informed decisionmaking directly related to risk-significance categorization. To review the DOE-proposed risk-significance categorization

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[15]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001).” Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[16]Reamer, C.W. “U.S. Nuclear Regulatory Commission Staff Review of the U.S. Department of Energy’s Proposed Approach to Risk Significance Categorization of Structures, Systems, and Components Important-to-Safety.” Letter (September 28) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

methodology adequately, it is necessary to consider the applicable policy and guidance governing the design, construction, and operation of a potential geologic repository operations area at the Yucca Mountain site and other similar NRC-regulated facilities. In the position paper,[17] the NRC staff performed an exhaustive review of the governing regulations and applicable regulatory policy and guidance. Additionally, the staff outlined the attributes of an acceptable risk-significance categorization process for structures, systems, and components identified as important to safety. These attributes include

- The risk-significance categorization of structures, systems, and components important to safety shall be consistent with existing regulatory framework.

- The risk-significance categorization of structures, systems, and components important to safety shall be consistent with their relative importance to safety.

- The risk-significance categorization of structures, systems, and components important to safety shall demonstrate flexibility.

- The documentation and analysis for the risk-significance categorization of structures, systems, and components identified as important to safety shall be transparent and traceable.

These attributes and the subsequent discussion form the basis for the acceptance criteria contained in Section 4.1.1.6.3 of NRC (2002). The paper also describes the DOE-proposed approach to risk-significance categorization of structures, systems, and components (CRWMS M&O, 1999c) and the NRC staff position on the DOE-proposed approach to categorization.

The proposed DOE risk-categorization methodology is based on the quality levels defined in procedure QAP–2–3 (CRWMS M&O, 1999c) and its associated screening criteria.[18] DOE stated the quality level or important-to-safety classification is consistent[19] with the three-tier approach and classification categories described in NRC (1996). The staff have several concerns regarding DOE use of the classification categories described in NUREG/CR–6407 (McConnel, et al., 1996) for the risk-significance categorization of structures, systems, and components important to safety for a potential geologic repository operations area. The approach identified in NUREG/CR–6407 [and its predecessor Regulatory Guide 7.10 (NRC, 1986)], however, predates all the risk-informed policy and guidance developed by NRC since the NRC document was issued in NRC (1995). In particular, the approach to classification identified in NUREG/CR–6407 does not require the consideration of risk insights or

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[17]Reamer, C.W. "U.S. Nuclear Regulatory Commission Staff Review of the U.S. Department of Energy's Proposed Approach to Risk Significance Categorization of Structures, Systems, and Components Important-to-Safety." Letter (September 28) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[18]Ibid.

[19]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

significance, nor does it consider probability of event sequence. The approach only assesses consequences as the maximum activity of radioactive material permitted in the transportation package. And, it assigns classification categories using a strictly deterministic approach. These concerns were discussed, and DOE agreed to clarify the approach to risk-significance categorization.[20]

DOE will need to show compliance with all requirements contained in 10 CFR Part 63. Although NRC requires compliance with all its requirements, NRC does not expect the same level of quality assurance is necessary to demonstrate compliance for each requirement. The NRC regulations provide flexibility to DOE for developing its quality assurance program, subject to review and approval by the NRC staff. The objective of a graded quality assurance program is to provide a level of quality assurance consistent with its importance to safety to ensure that each structure, system, or component will perform its safety function. As indicated in the staff position paper[21] and 10 CFR 63.142(c)(1), the DOE demonstration of compliance with the NRC requirements may include a graded quality assurance program that must control activity affecting the quality of identified structures, systems, and components to an extent consistent with its importance to safety. NRC, however, has the authority to make certain exceptions and specify additional requirements for certain attributes of the DOE quality assurance plan.

DOE is allowed by 10 CFR Part 63 to categorize or assign different levels of quality assurance to structures, systems, and components whose failure to function would result in different risk or dose implications. In approving such an approach, the NRC staff will take into account such items as the regulatory basis for the specific requirements, regulatory precedence, and risk significance.[22] For example, DOE suggested Quality Level 1 for structures, systems, and components related to meeting the overall public dose limit of 1.0 mSv/yr [100 mrem/yr] and Quality Level 2 for structures, systems, and components necessary for meeting the preclosure dose limit of 0.15 mSv/yr [15 mrem/yr]. Subject to further staff review of the quality provisions associated with Quality Levels 1 and 2, this approach appears appropriate.[23]

The following discussion identifies issues and concerns associated with the DOE-proposed approach to the risk-significance categorization of structures, systems, and components

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[20]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[21]Ibid.

[22]Reamer, C.W. "U.S. Nuclear Regulatory Commission Staff Review of the U.S. Department of Energy's Proposed Approach to Risk Significance Categorization of Structures, Systems, and Components Important-to-Safety." Letter (September 28) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[23]Ibid.

important to safety. Each of the following issues and concerns was discussed in the DOE and NRC Technical Exchange and Management Meeting.[24]

NRC was concerned that two of the DOE Quality Level 2 screening criteria [QAP–2–3, Appendix II, Checklist Items 8.2.5 and 8.2.6 (CRWMS M&O, 1999c)] are not consistent with the definition of event sequences provided in 10 CFR 63.2. These screening criteria consider the failure of only one item in conjunction with *an additional item* or *administrative control* (i.e., indirect impact). Whereas, the definition of event sequences (10 CFR 63.2) does not limit the number of component failures and states, "An event sequence includes one or more initiating events and *associated combinations of repository system component failures* …". DOE agreed the classification procedure should be clarified and linked to the preclosure safety assessment approach and processes to be used in the license application. DOE stated the preclosure safety assessment approach will make extensive use of event sequences that will clearly reveal any combination of events that leads to a release of, or exposure to, radioactivity. Events considered in potential event sequences will include potential failures or unavailability of structures, systems, and components in addition to potential human errors, including potential common-cause or dependent failures. Quality-level classifications will be assigned to structures, systems, and components important to safety consistent with their significance in preventing or mitigating event sequences. Consideration of multiple failures in credible scenarios will be included when determining items important to safety. DOE is updating the classification procedure (CRWMS M&O, 1999c) to clarify the process and tie it to the preclosure safety assessment. Also, the DOE preclosure safety assessment desktop reference should clarify how multiple failures will be considered when determining items important to safety. The response provided by DOE to comments in Section 2.1.6.3.1 (and the revised risk matrix in Figure 2.1.6-3) helps to address this concern.

NRC was concerned with the potential for the misclassification of structures, systems, and components identified as important to safety using QAP–2–3, Appendix II, Checklist Item 8.2.2, to identify Quality Level 2 items (CRWMS M&O, 1999c). This criterion asks, "Does the item provide fire protection, fire suppression, or otherwise protect important to radiological safety or waste isolation functions of Quality Level 1 structures, systems, and components identified as important to safety from the hazards of a fire?" According to the definition of Q-List 1 provided in procedure QAP–2–3, it would appear that structures, systems, and components meeting the requirements identified in QAP–2–3, Appendix II, Checklist Item 8.2.2, would more appropriately be categorized as Q-List 1 structures, systems, and components. DOE stated this screening criteria will be implemented consistent with the guidance provided in Regulatory Guide 1.189 (NRC, 2001). DOE agreed the classification procedure can be clarified to highlight consistency with Regulatory Guide 1.189 and the role of the item in the preclosure safety assessment process. Additionally, the preclosure safety assessment desktop reference will include guidance to the analyst for approaches to adequately address the criteria.

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[24]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

Repository Safety Before Permanent Closure

NRC was concerned with the potential for the misclassification of structures, systems, and components identified as important to safety using QAP–2–3, Appendix II, Quality Level 2, Checklist Item 8.2.3 (CRWMS M&O, 1999c). This criterion asks, “As a result of DBE [design basis event], could consequential failure of the item, which is not intended to perform a Quality Level 1 radiological safety function, prevent Quality Level 1 structures, systems, and components as important to safety from performing their intended radiological safety function?” The purpose and justification for this screening criterion are unclear. According to the DOE definition of Quality Level 1, this screening criterion appears to identify structures, systems, and components as important to safety “whose failure could *directly* result in a condition adversely affecting public safety” or risk, and should not be categorized as Quality Level 2 but Quality Level 1 structures, systems, and components identified as important to safety. DOE stated that structures, systems, and components classified as a result of interaction (i.e., seismic) issues have been traditionally classified as nonnuclear safety related in the commercial nuclear power industry and placed in augmented quality assurance programs. Criterion 8.2.3 recognizes that the structure, system, and component itself does not have to function to meet regulatory requirements, but its failure might potentially impact a Quality Level 1 structure, system, and component function. These criteria are included in Quality Level 2 to identify the potential safety significance of the item; however, following the NRC licensing precedent, full application of the quality assurance program is not required. Inclusion of these criteria in Quality Level 2 will require that the item be appropriately restrained to prevent interaction; however, quality assurance controls are not required to be related to the safety function of the item. DOE stated these screening criteria are indicated for the seismic interaction item and will be implemented consistent with the guidance provided in Regulatory Guide 1.29 (NRC, 1978). DOE agreed the classification procedure can be clarified to highlight consistency with Regulatory Guide 1.29 (NRC, 1978) and the role of the item in the preclosure safety assessment process. Additionally, the preclosure safety assessment desktop reference will include guidance to the analyst for approaches to address the criteria adequately.

NRC was concerned with the use of the terms in conjunction with and indirect impact as described in QAP–2–3, Appendix II, Checklist Items 8.2.5 and 8.2.6 (CRWMS M&O, 1999c). These screening criteria are not well defined. As described in QAP–2–3 (CRWMS M&O, 1999c), it appears that DOE could have a situation in which the failure of two Quality Level 2 structures, systems, and components identified as important to safety could potentially have the same risk as the failure of a single Quality Level 1 structure, system, or component identified as important to safety. The purpose and justification for this screening criterion are unclear. This screening criterion is more consistent with the DOE definition of Quality Level 1. Further, it would appear that either one or both these structures, systems, and components identified as important to safety would be categorized as Quality Level 1. DOE agreed to provide a definition of the term indirect impact that is based on, and consistent with, Regulatory Guides 1.29 (NRC, 1978) and 1.189 (NRC, 2001).

NRC was concerned that DOE was not planning to perform any uncertainty or sensitivity analyses of the quantification of event sequence frequencies. Uncertainty analyses are important because they can be used to identify and quantify sources of uncertainty and variability associated with the quantification of event sequence frequencies. It is important to understand the uncertainty and variability associated with the quantification of event sequence frequencies because the DOE risk thresholds are the same as the performance objective in

10 CFR 63.111. It is also necessary to have a clear understanding of the uncertainty and variability associated with the DOE frequency calculations because these frequency calculations are used to determine the frequency category of each of the respective event sequences and which performance objective applies to that particular event sequence. Uncertainty and sensitivity analyses will also be important in addressing some of the potential complexities associated with the DOE risk calculations for the event sequences. DOE needs to consider the use of uncertainty and sensitivity analyses where applicable or provide justification that explains why these analyses are not necessary. DOE concurs that uncertainty and sensitivity issues must be handled appropriately to support a license application. DOE agrees that the screening of design basis events must be defensible. One of the factors to consider is how well the screening basis is understood (e.g., failure probabilities, event sequence probabilities, or consequences). Uncertainties must be addressed to the extent they may impact either the categorization or the consequences of a potential design basis event. DOE also agreed that all design basis event categorizations, component failure probabilities, and consequence analyses must be technically defensible to support their use. DOE also agreed to justify the correctness and appropriateness of failure rates used in preclosure safety analyses. This justification would include discussions of the uncertainties and sensitivities associated with any failure rates (or other inputs used in the analyses).

The DOE classification analyses and subsequent risk categorization may benefit from the use of a multidisciplinary review group similar to the expert panel described in NRC (1998). The DOE-proposed approach to risk categorization relies on the screening criteria identified in QAP–2–3 (CRWMS M&O, 1999c) and the associated classification analyses. Specifically, DOE is relying heavily on those individuals performing these classification analyses. The NRC guidance recommends use of a multidisciplinary review group of technical and professional individuals, referred to as the expert panel, to support the risk-informed decisionmaking process. This expert panel performs an integrated assessment of quantitative risk insights to determine the safety significance ranking of structures, systems, and components identified as important to safety. DOE notes that the preclosure safety assessment preparation; structures, systems, and components classification; and the specification of quality assurance controls will involve a multidisciplinary team from safety analysis, licensing, design, criticality, fire safety, quality assurance, and others. Further, all documents will be subjected to multidisciplinary review. As such, DOE agreed to use a multidisciplinary review group similar to the expert panel described in NRC (1998).

2.1.6.4 Status and Path Forward

The status of identification of structures, systems, and components important to safety; safety controls; and measures to ensure availability of safety systems is given in Table 2.1.6-1. Limited general concerns on the methodology and assumptions pertaining to this preclosure topic were discussed at the first DOE and NRC Technical Exchange on Preclosure Safety.[25]

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[25]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001).” Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

Repository Safety Before Permanent Closure

The staff review of this preclosure topic is in progress. Additional concerns identified will be discussed in future technical exchanges.

Table 2.1.6-1. Summary of Resolution Status of Identification of Event Sequences Preclosure Topic

Preclosure Items	Status	Related Agreements	Comments
List of Structures, Systems, and Components Identified as Important to Safety	Pending	None*	Staff Review Incomplete
Administrative or Procedural Safety Controls	Pending	†	Staff Review Incomplete
Risk Significance Categorization of Structures, Systems, and Components Important to Safety	Pending	PRE.06.01 PRE.06.02	Staff Review Incomplete

*Limited general concerns were discussed in the first DOE and NRC Technical Exchange and Management Meeting on Preclosure Safety, July 24–26, 2001, Las Vegas, Nevada. No agreements were reached.
†Not discussed at the first DOE and NRC Technical Exchange on Preclosure Safety.

2.1.6.5 References

CRWMS M&O. "Preliminary Preclosure Design Basis Event Calculations for the Monitored Geologic Repository." BC0000000–01717–0210–00001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 1998.

———. "Classification of the MGR Assembly Transfer System." ANL–ATS–SE–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 1999a.

———. "Classification of the MGR Disposal Container Handling System." ANL–DCH–SE–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 1999b.

———. "Classification of Permanent Items." QAP–2–3. Revision 10. Las Vegas, Nevada: CRWMS M&O. 1999c.

———. "Classification of the MGR Waste Emplacement/Retrieval System." ANL–WES–SE–000001. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000a.

———. "Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations." TDR–WIS–RL–000001. Revision 04 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000b.

———. "Design Basis Event Frequency and Dose Calculation for Site Recommendation." CAL–WHS–SE–000001. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000c.

———. “Assembly Transfer System Description Document.” SDD–ATS–SE–000001. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000d.

———. “Canister Transfer System Description Document.” SDD–CTS–SE–000001. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000e.

———. “Disposal Container Handling System Description Document.” SDD–DCH–SE–000001. Revision 01 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000f.

———. “Subsurface Facility System Description Document.” SDD–SFS–SE–000001. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000g.

DOE. “Yucca Mountain Site Characterization Project Q-List.” YMP/90–55Q. Revision 6. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 2000.

———. “Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation.” TDR–MGR–SE–000009. Revision 00 ICN 03. Las Vegas, Nevada: DOE. 2001a.

———. “Yucca Mountain Science and Engineering Report Technical Information Site Recommendation Consideration.” DOE/RW–0539. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 2001b.

———. “Yucca Mountain Preliminary Site Suitability Evaluation.” DOE/RW–0540. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 2001c.

McConnell, Jr., J.W., A.L. Ayers, Jr., and M.J. Tyacke. NUREG/CR–6407, INEL–95/0551, “Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety.” Washington, DC: NRC. 1996.

NRC. Regulatory Guide 1.29, “Seismic Design Classification.” Revision 3. Washington, DC: NRC, Office of Standards Development. 1978.

———. Regulatory Guide 7.10, “Establishing Quality Assurance Programs for Packaging Used in the Transport of Radioactive Material.” Revision 1. Washington, DC: NRC, Office of Standards Development. 1986.

———. “Use of Probabilistic Risk Assessment in Nuclear Regulatory Activities: Final Policy Statement.” *Federal Register*. Vol. 60, No. 158. pp. 42622–42629. 1995.

———. Regulatory Guide 1.176, “An Approach for Plant-Specific, Risk-Informed Decisionmaking: Graded Quality Assurance.” Washington, DC: NRC, Office of Standards Development. 1998.

———. Regulatory Guide 1.189, “Fire Protection for Operating Nuclear Power Plants.” Washington, DC: NRC, Office of Standards Development. 2001.

———. NUREG–1804, "Yucca Mountain Review Plan—Draft Report for Comment." Revision 2. Washington, DC: NRC. March 2002.

2.1.7 Design of Structures, Systems, and Components Important to Safety and Safety Controls

2.1.7.1 Description of Issue

This section of the Integrated Issue Resolution Status Report addresses the design, specifications, component assessment, and fabrication methods (as applicable) for the important to safety surface and subsurface facilities and the waste package and engineered barrier subsystem. A license application for construction authorization of a geologic repository is required to include a preclosure safety analysis, 10 CFR 63.111(c). The preclosure safety analysis is to be used to demonstrate the safety of the proposed design and operations in the geologic repository operations area with regard to the overall preclosure performance objectives through a systematic examination of the site; the design; the potential hazards, the initiating events, and their resulting event sequences; and the potential radiological exposures to workers and the public (see 10 CFR 63.112). The geologic repository operations area must meet the requirements of 10 CFR Part 20. Category 1 design basis events are those natural and human-induced event sequences expected to occur one or more times before permanent closure. The annual dose limit for Category 1 events is 150 μSv [15 mrem] to the public and no greater than 50 mSv [5 rem] to the workers. Category 2 design basis events are those natural and human-induced event sequences that have at least one chance in 10,000 of occurring before permanent closure. The dose limit for Category 2 events is 50 mSv [5 rem] to the public per event sequence [see 10 CFR 63.111(b)(2) for additional information pertaining to individual organ or tissue dose limits]. Beyond design basis events are those events that have less than one chance in 10,000 of occurring within the preclosure period. The preclosure safety analysis is specifically required to include a general description and discussion of the design, both surface and subsurface, of the geologic repository area [10 CFR 63.112(f)]. In addition, 10 CFR 63.112(e) requires that preclosure safety analysis be used to assess the performance of the structures, systems, and components to identify those that are important to safety. These analyses should include consideration of suitable shielding [10 CFR 63.112(e)(3)]; means to prevent and control criticality [10 CFR 63.112(e)(6)]; ability of structures, systems, and components to perform their intended safety functions, assuming the occurrence of event sequences [10 CFR 63.112(e)(8)]; and means to inspect, test, and maintain structures, systems, and components important to safety [10 CFR 63.112(e)(13)]. Moreover, 10 CFR 63.21(c)(3) requires the safety analyses report, filed with the license application, to include a description and discussion of the design of the various components of the geologic repository operations area and the engineered barrier subsystem. This description and discussion must include (i) dimensions, material properties, specifications, and analytical and design methods used, along with any applicable codes and standards; (ii) the design criteria used and their relationships to the preclosure performance objectives specified in 10 CFR 63.111(b), 63.113(b), and 63.113(c); and (iii) the design bases and their relation to the design criteria.

Surface Facility

An assessment of the proposed surface facility will be provided at a later date.

Subsurface Facility

The subsurface facility consists of CRWMS M&O (2000a) (i) portals and access ramps, (ii) access mains, (iii) emplacement drifts, (iv) openings to support the subsurface ventilation, and (v) openings to support monitoring and performance confirmation testing.

The portals and access ramps (North Portal, South Portal, North Ramp, and South Ramp) of the existing Exploratory Studies Facility would be integrated into the proposed repository and would connect the surface and subsurface facilities through the access mains (CRWMS M&O, 2000a). The North Ramp provides access to the emplacement side of the subsurface facility, and the South Ramp provides access to the development side (CRWMS M&O, 2000a).

The access mains are a network of tunnels that define the perimeter of and provide access to the proposed emplacement area. The access mains are comprised of the north-south trending east main and west main, which are interconnected through other shorter tunnels, such as the north main and south main, and to the surface facility through the access ramps (CRWMS M&O, 2000b, Figure 2). The access mains have a nominal diameter of 7.62 m [25 ft] and are provided with rail lines to support the transport of waste packages to and from the emplacement area. The east and west mains will also serve to conduct intake ventilation air to the emplacement area (CRWMS M&O, 2000c).

The emplacement drifts are an array of horizontal tunnels trending approximately east-northeast–west-southwest (252° azimuth) between the east and west mains. Each drift will have a diameter of 5.5 m [18.5 ft] and will be separated from the adjacent drifts by a center-to-center distance of 81 m [265.7 ft]. The transition from the east and west mains to the emplacement drifts (which are nearly perpendicular to the mains) is provided through the emplacement-drift turnouts (CRWMS M&O, 2000a, Figure 1). A pair of isolation doors located near the emplacement-drift and access-main ends of each turnout will help control airflow into the emplacement drifts and protect the access mains from radiation that emanates from waste packages in the emplacement drifts. The ground-support system for the emplacement drifts will consist of steel sets and wire mesh, with occasional rock bolts installed in the roof area if considered necessary during construction. The ground support will be of carbon-steel material and will be designed for an operational life up to 175 years with possible extension to 300 years (CRWMS M&O, 2000d).

The other openings of the underground facility include the north-south-trending exhaust main located below the emplacement drifts, ventilation raises (i.e., shafts excavated from the floor of the emplacement drifts to the roof of the exhaust main), the intake and exhaust shafts, and other drifts within the emplacement block that will be used for various purposes other than waste emplacement. The ground-support system for the nonemplacement openings (including the access mains) will initially consist of pattern rock bolts and welded wire fabric and, where necessary, shotcrete or steel sets. A final ground support consisting of a cast-in-place concrete lining will be installed to provide long-term support for such openings during the preclosure period.

The design of the subsurface facility incorporates subject matter previously reviewed within the framework of two subissues of the Repository Design and Thermal-Mechanical Effects Key Technical Issue (NRC, 2000a): Subissue 2, Seismic Design Methodology; and Subissue 3, Component (i), Thermal-Mechanical Effects on Underground Facility Design. In the subsequent sections, applicable portions of these subissues are considered but no effort is made to explicitly identify them.

Engineered Barrier Subsystem

In addition to the waste package, other components of the engineered barrier subsystem that may be used during preclosure operations at the proposed geologic repository include a drip shield, drift invert, waste package pallet, and backfill. The DOE site recommendation reference design (CRWMS M&O, 1999a) indicates that several variations of the basic waste package design will have to be implemented to accommodate the different types of spent nuclear fuel and high-level waste glass. The basic waste package design concept uses two concentric cylinders of different metallic materials. The outer container or barrier will be made from a highly corrosion-resistant Alloy 22, surrounding an inner container made of Type 316 nuclear grade stainless steel (CRWMS M&O, 2000e). Fabrication processes used in the construction of the waste packages (e.g., forming, welding, and stress-relieving operations) may alter the performance of the container materials. The waste packages will be supported by pallets and emplaced in a horizontal orientation within the repository drifts. In addition to the spent nuclear fuel and high-level waste, the waste packages will also contain a number of engineered components designed to provide criticality control, provide structural support, and transfer heat from the waste package interior to the waste package surface (CRWMS M&O, 2000f). Each waste package will rest on an emplacement pallet made of two V-shaped Alloy 22 supports connected by hollow stainless steel tubes with square-shaped cross sections. The waste package pallets will, in turn, rest on the drift invert. A mailbox-shaped drip shield, fabricated with a titanium-palladium alloy (Titanium Grades 7 and 24), will be placed over the waste packages and, by interlocking the individual drip shield units, will extend continuously over the entire length of the emplacement drifts. The drip shields will rest on the drift invert and provide shielding for both the top and sides of the waste packages (CRWMS M&O, 2000g). The current repository reference design does not include backfill.

The design of the waste package and engineered barrier subsystem components incorporates subject matter previously reviewed within the framework of four subissues of the Container Life and Source Term Key Technical Issue (NRC, 2001) and Subissue 1, System Description and Demonstration of Multiple Barriers, of the Total System Performance Assessment and Integration Key Technical Issue (NRC, 2000b). The specific applicable Container Life and Source Term Key Technical Issue subissues are Subissue 1, Effects of Corrosion Processes on the Life of the Containers; Subissue 2, Effects of Phase Instability of Materials and Initial Defects on the Mechanical Failure and Life of the Containers; and Subissue 6, Effects of Alternate Engineered Barrier Subsystem Design Features on Container Life and Radionuclide Release from the Engineered Barrier Subsystem.

Repository Safety Before Permanent Closure

The Design of Structures, Systems, and Components and Safety Controls that are safety related for the waste package and engineered barrier subsystem is also related to Container Life and Source Term Key Technical Issue Subissue 5, Effect of In-package Criticality on Waste Package and Engineered Barrier Subsystem Performance. The relationship exists, in the case of phase instability of materials, because microstructural changes (e.g., ordering transformation, intermetallic precipitation, and metalloid segregation) that may affect the mechanical properties of the containers could result from welding operations, weld repairs, and postweld treatments. Mechanical failure of the container and subsequent penetration of water are necessary conditions for a criticality event. At present, criticality has been screened out on the basis of low probability. The technical basis for this screening argument is the anticipated long life of the waste packages. In the subsequent sections, applicable portions of these subissues are considered, and the current resolution status is provided.

Design descriptions as well as details of the fabrication, inspection, repair, and emplacement of the waste package and engineered barrier subsystem components are necessary to evaluate the DOE preclosure safety strategy. DOE provided information for the current designs of the waste packages and engineered barrier subsystem components (CRWMS M&O, 2000 e–g). Fabrication methods that may be used to construct the waste packages and engineered barrier subsystem components are also provided in DOE documents (CRWMS M&O, 2001a,b). This section of the Integrated Issue Resolution Status Report has been prepared based on a review of these reports, other DOE documents, and discussions at the first preclosure technical exchange.[1] Agreements were reached on specific issues concerning waste package design, inspection methods, variations in the mechanical properties of the waste packages, and the effects of fabrication and repair on waste package performance.

2.1.7.2 Importance to Safety

The DOE repository safety strategy (CRWMS M&O, 2000h) for preclosure focuses on the regulatory performance objectives for the repository system through permanent closure. Elements of the repository preclosure safety case include Preclosure Safety Analyses (referred to as Integrated Safety Analyses by DOE), margin and defense-in-depth evaluations, consequence analyses of various event sequences, commercial nuclear industry precedent and experience, and license specifications and surveillances. Compliance with the repository preclosure performance objectives will be demonstrated through the Preclosure Safety Analyses. The purpose of the Preclosure Safety Analyses is to ensure relevant hazards that could result in unacceptable consequences have been evaluated, and preventive or mitigative features are included in the repository design to limit radiation exposures to those specified in 10 CFR 63.111.

Surface Facility

An assessment of the surface facility will be provided at a later date.

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[1]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Repository Design and Thermal-Mechanical Effects (February 6–8, 2001)." Letter (February 28) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

Subsurface Facility

Among the subsurface facility openings, only the emplacement drifts are classified as important to safety (the drifts are assigned Quality Level 1, and the supporting ground-control system is assigned Quality Level 2) in the DOE safety categorization of structures, systems, and components (DOE, 2000). The emplacement drifts provide the space and physical support for the structures, systems, and components used for emplacement and retrieval operations, as well as shielding the rest of the underground facilities from radiation that will emanate from the waste packages. The emplacement-drift invert provides physical support for the gantry rail and cranes critical to the movement of waste packages into and out of the emplacement drifts (CRWMS M&O, 2000b). The radiation-shielding function of the emplacement drifts requires proper functioning of the isolation doors (between the emplacement drifts and the access mains). Although the isolation doors are not identified explicitly in the DOE safety categorization of structures, systems, and components, their design should receive the same level of scrutiny as the emplacement-drift design to ensure the radiation-shielding function of the drifts would be performed satisfactorily.

The rock mass surrounding the emplacement drifts will be subjected to loadings from *in-situ* stress, thermal stress resulting from waste-generated heat, and seismically induced stress. In addition, there may be other loadings arising from the repository operations. These loadings may cause drift collapse, dynamic rockfall impact on the waste packages, or buckling of the gantry rail or isolation doors, which can interfere with the safety functions of the emplacement-drift system. DOE will be required (10 CFR 63.112) to demonstrate that the emplacement-drift system would perform its safety functions adequately (i.e., provide adequate space and physical support for the emplacement and retrieval structures, systems, and components; operations; and adequate radiation shielding) through the preclosure period. This section presents a review of the DOE information on subsurface facility design. The object of the review is to determine if DOE has assembled enough information for inclusion in the initial license application for NRC review and regulatory decisionmaking.

Engineered Barrier Subsystem

DOE states that the disposal containers (i.e., waste packages) will prevent releases during various event sequences, including falling objects striking the disposal containers or the waste package, waste package drops, waste package slapdown, waste package collisions during transport and emplacement, missiles and explosive overpressures, fires and thermal hazards, waste package overpressure, and waste package criticality (CRWMS M&O, 2000h). In addition, the waste package is cited as a design mitigation feature that limits dose for several different event sequences, including criticality caused by internal geometry failure, rockfall on the waste package or the transporter, and transporter runaway. As a result, the waste package has been designated as a Quality Level 1 important to safety structure (CRWMS M&O, 2000h).

The potential for mechanical failure of the waste package during preclosure operations needs to be evaluated because of DOE reliance on its ability to maintain confinement of the spent nuclear fuel and high-level waste during normal handling or when subjected to Categories 1 or 2 events. Normal handling operations that will subject the waste package to mechanical loading include lifting, transport, and emplacement. Operational events, such as waste package drops, have the potential to cause mechanical damage by loading the waste package beyond the yield strength of the material. The design and construction of the waste package will be important in the assessment of mechanical loading events resulting in plastic deformation (i.e., loads that exceed the yield strength of the waste package materials). The mechanical properties of the welded regions may be different from the original rolled plate. In addition, the effects of stress mitigation methods may also alter the mechanical properties of the waste package materials.

2.1.7.3 Technical Basis

The review uses the acceptance criteria provided in NRC (2002).

2.1.7.3.1 Relationship Between the Design Criteria and Design Bases and the Regulatory Requirements

Text for this section will be provided at a later date.

2.1.7.3.2 Geologic Repository Operations Area Design Methodologies

Text for this section will be provided at a later date.

2.1.7.3.3 Geologic Repository Operations Area Design and Design Analyses

2.1.7.3.3.1 Surface Facilities

Assumptions, Codes, and Standards for Surface Facilities Design

Text for this section will be provided at a later date.

Materials for Surface Facilities Design

Text for this section will be provided at a later date.

Load Combinations for Surface Facilities Design

Text for this section will be provided at a later date.

Design Analyses and Documentation

Text for this section will be provided at a later date.

2.1.7.3.3.2 Subsurface Facility

Assumptions, Codes, and Standards for Subsurface Facility Design

Text for this section will be provided at a later date.

Subsurface Operating Systems

Text for this section will be provided at a later date.

Materials and Material Properties for Subsurface Facility Design

The scope of this acceptance criterion includes the materials used for the ground support and drift invert but does not include the material properties of the surrounding rock. The proposed material for the ground support (steel sets, wire mesh, and rock bolt) and structural components of the invert is carbon steel (CRWMS M&O, 2000d,i). The ground support will be designed for an operational life up to 175 years, with a possible extension to 300 years. An analysis of the invert has not been presented, but DOE indicated that the invert will be designed to maintain the waste packages in their horizontal emplacement positions through the period of regulatory concern (CRWMS M&O, 2000i).

DOE concluded that the lifetime of carbon steel is sufficient to provide the required service life for the ground support (CRWMS M&O, 2000j). This lifetime prediction is based on (i) no aqueous corrosion will occur during the preclosure period because of an assumption that ventilation will remove any water that percolates into the emplacement drifts; (ii) no pitting or crevice corrosion is expected because the relative humidity will be low, the chloride concentration of the groundwater is low, and the pH of the groundwater is near neutral; and (iii) humid-air corrosion may occur but will not be sufficient to affect the mechanical properties of carbon steel for at least 300 years. The analysis was made using the humid-air corrosion rate at a relative humidity of 40 percent, which was assumed to be 0.001 to 0.2 times the humid-air corrosion rate for carbon steel at a relative humidity above the critical relative humidity for humid-air corrosion. The corrosion-rate data were taken from results of experiments conducted to assess the performance of the waste package design for viability assessment (McCright, 1998), which used a carbon steel outer barrier.

Dry-air oxidation of the ground-support material was also evaluated (CRWMS M&O, 2000j) but was predicted to be insignificant. The penetration of the carbon steel ground support by dry oxidation was calculated to be 1×10^{-5} mm [3.9×10^{-7} in] at 100 °C [212 °F] or 1×10^{-4} mm [3.9×10^{-6} in] at 150 °C [302 °F] over a period of 300 years. The potentially detrimental effects of microbial activity were not considered because the environmental conditions (i.e., lack of water, low relative humidity, and high temperatures) are not expected to support microbial populations.

There are two concerns with the DOE prediction of ground-support service life. First, the service-life estimate was based entirely on an estimation of the humid-air corrosion rate for carbon steel at a relative humidity in the range of 1–40 percent. The effect of higher relative humidity on the service life was not determined, and a technical basis was not presented for the assumption that the relative humidity of the emplacement drifts will be at 40 percent or less. Second, the basis for not considering the possibility of aqueous corrosion of the ground-support materials during preclosure is that ventilation will remove any water that percolates into the drift. However, the corrosion effects of water trapped in crevices between the ground support and the drift wall were not evaluated. Water trapped in such crevices may evaporate slowly because ventilation in such locations may be substantially reduced compared with the overall ventilation rate in the drift. In addition, dryout and rewetting of the crevice regions may result in variations in the pH and chloride concentrations that will increase the corrosion rate of the carbon steel materials. For example, localized corrosion of carbon steel is known to result in significant acidification of pit and crevice solutions (pH ~2–4.5) from hydrolysis of the Fe^{2+} cations (Szklarska-Smialowska, 1986), and the acidic pH in the crevice region increases the corrosion rate of the carbon steel. Dryout and rewetting cycles may also increase the chloride concentration and promote localized corrosion.

To address these concerns, DOE agreed at the DOE and NRC Technical Exchange on Repository Design and Thermal-Mechanical Effects[2] to provide additional documentation. The information will be provided as part of the issue resolution process and, if provided by DOE by the time of any license application, should afford sufficient information for NRC to conduct its licensing review. As agreed, DOE will provide the technical basis for the ranges of relative humidity and temperature used for the preclosure assessment of ground-support performance, and an assessment of, and the technical basis for, the potential effects of localized liquid phase water on ground-support systems during the preclosure period.

Also, DOE should present a technical basis for the service life of the drift invert to support the assertion (CRWMS M&O, 2000i, Section 1.2.1) that the drift invert will maintain its horizontal position through the preclosure period. This technical basis will be discussed during future preclosure meetings. There are also concerns about the postclosure service life of the drift invert, but these concerns are discussed in Section 3.3.4, Radionuclide Release Rates and Solubility Limits.

Load Combinations for Subsurface Facility Design

This acceptance criterion would be satisfied if the appropriate load combinations for normal and Categories 1 and 2 event sequence conditions are used in the design analyses of subsurface structures, systems, and components important to safety.

DOE has set performance criteria for several structures, systems, and components that call for a design against the worst-case load combinations (e.g., CRWMS M&O, 2000d, Section 1.2.1.6). In the stability analyses of emplacement drifts for site recommendation (CRWMS M&O, 2000k), the worst-case load combination was assumed to be achieved by superimposing seismic loading on thermal loading at about 10 years after waste emplacement (i.e., when the drift-wall temperature was close to its peak value).

The potential failure modes of structures, systems, and components, however, should be considered in determining the appropriate load combinations for design. For example, because buckling of structural members is an important failure mode for the drift invert, loading conditions that may cause axial compression of the structural members would be considered

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[2]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Repository Design and Thermal-Mechanical Effects (February 6–8, 2001)." Letter (February 28) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

critical for their design. Hence, the performance of the structural members under peak temperature conditions may govern their design. On the other hand, the critical combination of thermal and seismic loading for the stability of the emplacement drifts may not necessarily correspond to the peak drift-wall temperature. The effect of combined thermal and seismically induced stresses on the stability of underground openings depends to a large extent on the timing of the seismic-loading episode. In general, a seismic-loading episode that occurs when rock temperatures (and, therefore, the interlocking effects of thermal stress) are relatively high may cause less damage than a seismic episode that either occurs when the rock temperature is lower or is superimposed on preexisting thermally induced shear failure. Therefore, several different loading combinations need to be considered to determine the loading combination that should govern the ground-support design.

The repository thermal loading is dependent on the subsurface-facility design (CRWMS M&O, 2000a) and the heat-output history of the waste packages (CRWMS M&O, 2000l). Also, the amount of the waste-generated heat transmitted into the host rock and subsurface-facility structures, systems, and components may be affected by ventilation (CRWMS M&O, 2000c). DOE expects to develop a numerical modeling approach to calculate the amount of heat removed by ventilation and verify the model using laboratory test data. This information will be submitted to NRC in 2002, based on a DOE and NRC agreement.[3] Also, the DOE characterization of the seismic-loading and fault-displacement histories for Yucca Mountain will be provided in Seismic Topical Report 3, which will be submitted to NRC in 2002.[4]

To address the NRC concerns regarding the load combinations used for the design and analysis of structures, systems, and components important to safety, DOE agreed at the DOE and NRC Technical Exchange on Repository Design and Thermal-Mechanical Effects[5] to provide additional documentation. The information will be provided as part of the issue resolution process and, if provided by DOE by the time of any license application, should afford sufficient information for NRC to conduct its licensing review. As agreed, DOE will provide the critical combinations of *in-situ*, thermal, and seismic loadings; the technical basis for the critical combinations; and their effects on preclosure ground-support performance. Although this agreement specifically addresses only the ground support, it is assumed that the same information (the description, technical basis, and performance impact of the critical load combinations) will be provided for all structures, systems, and components important to safety including, for example, the drift invert and isolation doors.

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[3]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Repository Design and Thermal-Mechanical Effects (February 6–8, 2001).” Letter (February 28) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[4]Ibid.

[5]Ibid.

Models and Rock Properties for Subsurface Facility Design

This acceptance criterion would be satisfied if appropriate models and site-specific rock properties are used for the design analyses of subsurface structures, systems, and components, and the spatial and temporal variations and uncertainties in the rock properties are adequately considered in the analyses. The DOE design analyses for the subsurface structures, systems, and components to support the site recommendation are documented in CRWMS M&O (2000k), which presents analyses for the emplacement drifts and for nonemplacement openings, such as the exhaust main. The drift invert and isolation doors were not discussed in the report. Analyses of the emplacement and nonemplacement drifts were conducted using numerical modeling to examine the performance of the openings when subjected to loadings from *in-situ* stress, waste-generated heat, and seismic ground motion. The performance of the openings with and without ground support was examined using continuum rock-mass modeling. Analyses were also conducted using discontinuum models of the rock mass, but only for openings without ground support. The performance of the openings was based on ground-support loading (from continuum analyses only), deformation of the perimeter walls of the openings, and the occurrence of inelastic deformation in the surrounding rock.

Because of several insufficiencies, the analyses of the subsurface structures, systems, and components used to support the DOE site recommendation (CRWMS M&O, 2000k) would not satisfy the acceptance criterion that design analyses use appropriate models and site-specific properties of the host rock and consider the spatial and temporal variations and uncertainties in such properties (NRC, 2000a). To address these insufficiencies by license application, DOE agreed at the DOE and NRC Technical Exchange on Repository Design and Thermal-Mechanical Effects[6] to provide additional documentation. This information will be provided as part of the issue resolution process and, if provided by DOE by the time of any license application, should afford sufficient information for NRC to conduct its licensing review. The specific concerns raised by the NRC staff are discussed in the following paragraphs.

<u>Model Boundary Conditions</u>

Thermal-mechanical analyses of the emplacement drifts were conducted using a drift-scale model truncated at a distance of 50 m [164 ft] above and below the emplacement-drift axis. The base of the model {i.e., at 50 m [164 ft] below the axis} was held at zero vertical displacement, whereas the model top {i.e., at 50 m [164 ft] above the axis} was held at constant normal traction equivalent to the preemplacement *in-situ* stress, through a simulation time of 200 years after waste emplacement (CRWMS M&O, 2000k, Figures 6-4 and 6-5). Such a model is inappropriate because it allows excessive free upward thermal expansion, thereby interfering with the development of thermally induced stress consistent with the geometry of the emplacement area.

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[6]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Repository Design and Thermal-Mechanical Effects (February 6–8, 2001)." Letter (February 28) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

As shown in Figure 2.1.7-1 (Ofoegbu, 2001), the emplacement geometry will have a strong influence on the nature and magnitude of thermally induced stress and the associated mechanism and distribution of potential rock failure. Two features of the emplacement geometry that influence the anticipated thermal-mechanical behavior are the large lateral extent of the emplacement-drift array relative to the vertical extent and the closeness of the drift array to the ground surface relative to the distance to other boundaries of the host rock mass (CRWMS M&O, 2000m). For a typical drift within the emplacement-drift array, thermal expansion of the surrounding rock would be fully suppressed laterally, but a limited amount of upward expansion can occur because of free movement at the ground surface

(Figure 2.1.7-1). Consequently, the anticipated horizontal component of thermal stress is much higher than the vertical component. The only exception is in areas close to the sidewall of the drift openings where the vertical component of thermal stress would be higher than the drift-normal horizontal component because of the closeness of a traction-free boundary. The upward expansion of the heated zones around a drift would impose an upward pull on cooler areas in the pillars, resulting in thermally induced tension in the vertical direction (Figure 2.1.7-1). The vertical component of rock stress near the pillar centers would, thus, be expected to decrease and may occasionally be tensile. These stress conditions, which depend only on the emplacement geometry, favor the development of potential zones of rock failure (by fracture slip) through the mechanisms illustrated in Figure 2.1.7-1 (i.e., reverse-faulting style in the roof and floor areas of the drifts and in the pillars, and strike-slip or normal-faulting styles near the drift sidewalls). The magnitudes of the induced stresses and whether such stresses

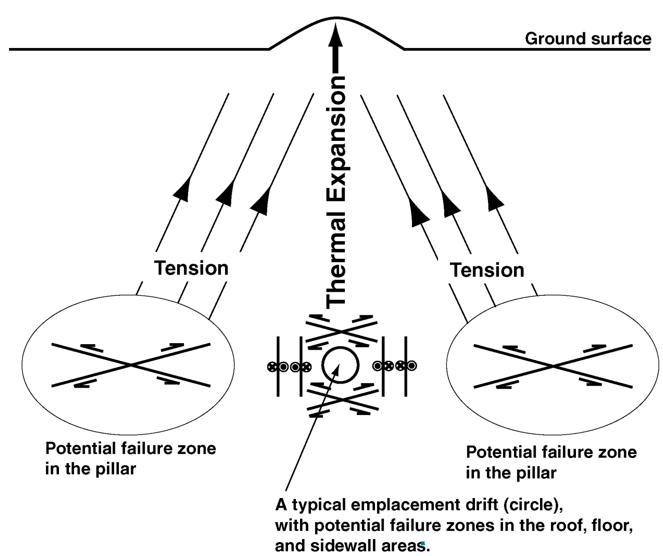

Figure 2.1.7-1. Schematic Illustration of the Anticipated Mechanisms of Thermal-Mechanical Response, Showing the Effects of the Emplacement Geometry on the Distributions of Zones of Potential Rock Failure in a Horizontal Array of Drifts. (Actual Development of the Failure Zones Would Be Determined by the Rock-Mass Mechanical Properties and the Induced Temperature and Temperature Gradients.)

are sufficient to cause rock failure will, of course, depend on the induced temperature and the rock-mass mechanical properties. For example, results from numerical modeling (Ofoegbu, 1999, 2000, 2001; Ofoegbu, et al., 2001) indicate that the development of failure in the pillars would be more likely in higher-stiffness rock, in which the magnitude of induced thermal stress may be sufficient to satisfy the failure criteria. The occurrence of thermally induced stress change sufficient to cause failure and an appreciable reorientation of principal stresses in the pillar adjacent to a heated underground opening have previously been predicted through numerical modeling of steam-injection processes in a petroleum reservoir (Ofoegbu and Curran, 1987).

As illustrated in Figure 2.1.7-1 (Ofoegbu, 2001) and discussed in the foregoing paragraph, the effect of geometry on thermally induced stress depends to a large extent on the location of a mechanically free boundary, such as the ground surface. The topography of Yucca Mountain (e.g., Section 2.1.1) is such that the distance to the closest free surface and the orientation of the direct line from an emplacement drift to the free surface vary over the proposed emplacement area. For example, a typical east-west vertical section through Yucca Mountain (e.g., DOE, 2001a, Figure 1-10) indicates that the direct line from the emplacement area to the closest free surface would be inclined approximately 45 degrees to the vertical in the west (where the closest free surface is the Solitario Canyon) but would be nearly vertical in the east. Therefore, the orientation of the thermally induced tension in Figure 2.1.7-1 would vary over the emplacement area. For this reason, the topography of Yucca Mountain may have an important effect on the distributions of thermally induced stress and potential failure zones within the proposed emplacement area.

The DOE drift-scale model (CRWMS M&O, 2000k, Figures 6-4 and 6-5) would not permit the development of thermal stresses consistent with the proposed emplacement geometry because the boundary conditions applied at 50 m [164 ft] above and below the drift axis in the model allow excessive upward freedom. Therefore, the model does not represent the anticipated thermal-mechanical environment within and around the emplacement area and, consequently, is inappropriate for predicting the performance of the emplacement drifts. DOE agreed[7] to address this concern.

Model Dimensionality

The thermal-mechanical analyses for site recommendation (CRWMS M&O, 2000k) were conducted using two-dimensional models based on a vertical section normal to the proposed emplacement-drift alignment. DOE stated, without technical basis, that the two-dimensional models give satisfactory estimates of the performance of the subsurface openings.

The NRC staff concern about the appropriateness of two-dimensional thermal-mechanical modeling of the emplacement drifts arises because the *in-situ* horizontal principal stresses (Stock, et al., 1985) and several of the fracture sets (CRWMS M&O, 2000n) are oblique to the

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proposed drift alignment (252° azimuth, that is S72 °W). The ambient minimum principal stress is horizontal and oriented N60 °W–N65 °W (Stock, et al., 1985), which is 40–45 degrees from the drift-normal plane (the assumed orientation of the minimum principal stress for the two-dimensional modeling). Also, the dip direction of the subhorizontal fractures, which are likely to dominate the rock-failure mechanism as illustrated in Figure 2.1.7-1 (Ofoegbu, 2001), lies in the 40–60-degree range (i.e., 10–30 ° from the drift orientation). Therefore, the two-dimensional models are not favorably oriented to detect slip on the subhorizontal fractures. Three-dimensional modeling may be necessary to determine the effects of these structural features that are oblique to the drift alignment.

Other areas for which three-dimensional modeling may also be necessary include (i) stability of the turnout area (between the emplacement drifts and the access mains), which may be subjected to a combination of vertical tension and high-horizontal compression similar to the phenomenon illustrated in Figure 2.1.7-1 (Ofoegbu, 2001); (ii) effects of greater heat conduction rates through the drift floor because steel members in the floor (invert and pallet) that are in direct or indirect contact with the waste package provide a faster heat-flow path into the rock; (iii) stability of the structural components of the invert (transverse and longitudinal beams) and the interaction of the transverse beams with the drift wall under heated conditions; and (iv) effects of ground-surface topography drift-parallel thermal gradients on thermal stress and, consequently, drift stability. DOE has agreed[8] to address this concern.

Model Representation of Fracture Network

Discontinuum models used in the thermal-mechanical analyses for site recommendation (CRWMS M&O, 2000k) were based on a regular fracture pattern composed from the mean fracture-set attitudes (dip and dip direction) and spacing, but the uncertainties in the fracture-set properties and their effects on the calculated results were not discussed. The DOE fracture data (CRWMS M&O, 2000n,o) indicate a considerable variation of the fracture-attitude parameters and spacing around the mean values for fracture sets, which means that the *in-situ* fracture pattern is irregular and variable. The simplified pattern used in the DOE analyses may be adequate for conducting numerical experiments, but the differences between the model and *in-situ* fracture patterns should be understood and factored into the interpretation of the analyses results and the facility design. DOE has agreed[9] to address this concern.

Model Representation of Seismic Loading

Seismic loading was represented in the models as a sinusoidal velocity history with a frequency of 10 Hz, an amplitude equal to the estimated peak ground velocity for the site, and a duration of 3 seconds (CRWMS M&O, 2000k). This approach for representing seismic loading was based on three assumptions (CRWMS M&O, 2000k, Sections 5.3.1–5.3.3). DOE assumed that (i) the use of a sinusoidal wave of constant amplitude is conservative because it results in

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[8]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Repository Design and Thermal-Mechanical Effects (February 6–8, 2001)." Letter (February 28) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[9]Ibid.

applying more cycles of the peak ground velocity at a point than would occur in an actual seismic event; (ii) a frequency of 10 Hz results in a seismic wavelength of a few hundred meters {considering the estimated shear wave velocity of approximately 3,000 m/s [9,843 ft/s]}, and this wave length is appropriate because seismic waves generally have large wave lengths; and (iii) the 30 cycles of motion that result from applying a 10-Hz sinusoidal motion for 3 seconds is conservative because the host rock does not show significant nonlinear behavior during seismic loading.

The justifications given for the three assumptions do not include an explanation of how it was determined that the applied velocity history constitutes an adequate representation of the ground-motion time history for Yucca Mountain. The site-specific ground-motion time history would differ from the model velocity history in terms of frequency content, amplitude variation, and duration of loading, so a comparison of the two might examine the total energy delivered to the rock in either case and the amount of that energy available to cause rock failure (e.g., by fracture slip). Such a comparison may be accomplished through a combination of theoretical analysis, scaled-model testing, and numerical experimentation. Numerical modeling results indicate that the dynamic response of the rock mass surrounding the emplacement drifts could be underestimated if a sinusoidal motion with a frequency of 10 Hz and a duration of 3 seconds is used in the analysis instead of the site-specific ground motions (Hsiung, et al., 2001). This overestimation could potentially result in a design of a ground-support system that is insufficient. DOE has agreed[10] to address this concern.

<u>Rock-Mass Mechanical Properties: Effects of Lithophysae</u>

The values of rock-mass mechanical properties for lithophysal and nonlithophysal rock units were determined using empirical correlations between such properties and the rock-mass quality indices, such as the *Q* index of Barton, et al. (1974) or the RMR (Rock Mass Rating) index of Bieniawski (1979). These quality indices were developed to account for the effects of fractures on the mechanical characteristics of a rock mass. The use of the *Q* and RMR indexes to account for the effects of lithophysae (CRWMS M&O, 2000k) is unprecedented and not supported by any data on or model investigation of the effects of lithophysae on the mechanical characteristics of rock.

The values of the *Q* and RMR indexes are determined through an accumulation of a set of categorical variables that are assigned values to represent aspects of the mechanical attributes of fractures. For example,

$$Q = (\mathrm{RQD}/J_n) \times (J_r/J_a) \times (J_w/\mathrm{SRF}) \qquad (2.1.7\text{-}1)$$

where RQD is the rock quality designation, J_n is the joint-set number, J_r and J_a represent joint roughness and alteration, and J_w and SRF are factors used to represent water pressure and rock stress (Barton, et al., 1974). The ratio (J_w/SRF) is set to one if *Q* is used to determine

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parameter values for stress analyses (instead of being used directly to design ground support) because the effects of water pressure and rock stress can be accounted for directly in such analyses. Each of the parameters used to calculate *Q* is assigned a value from tables compiled by the original developers of the technique (Barton, et al., 1974). Generally, the ratio (RQD/J_n) represents the unfractured-rock block size, (J_r/J_a) represents the strength of the joint (or fracture) surfaces, and (J_w/SRF) represents the stress state. It is conceivable that the lithophysal content of a rock may be correlated somewhat with the RQD value, but none of the other parameters can be readily correlated to the mechanical attributes of lithophysae.

Therefore, using the *Q* index to characterize the effects of lithophysae on the mechanical characteristics of a rock mass is tantamount to assuming the RQD alone is sufficient as a mechanical-behavior index. This assumption was rejected several decades ago (e.g., consider the histories of the *Q* and RMR indexes). Therefore, there is currently inadequate technical basis to support the use of either *Q* or RMR to characterize the mechanical behavior of the lithophysal tuff. Although these indices may be appropriate for accounting for the effects of fractures, some modification of their values would be necessary if DOE uses the indexes to account for the effects of lithophysae. The technical basis for such modification is all the more important because about 75 percent of the proposed emplacement area may lie within the lithophysal rock units. To address these insufficiencies by license application, DOE agreed at the DOE and NRC Technical Exchange on Repository Design and Thermal-Mechanical Effects[11] to provide additional documentation.

<u>Rock-Mass Mechanical Properties: Effects of Fractures</u>

The DOE approach to mechanical characterization of Yucca Mountain is to determine the values of mechanical properties using empirical correlations between the properties and the rock-mass quality indexes, such as *Q* and RMR. Two sets of *Q* and RMR values were determined along the Exploratory Studies Facility main drift and North and South Ramps based on a scan-line survey and a full-periphery map of the tunnel (CRWMS M&O, 1997a, Figures 39 and 40). The rock mass was classified into five quality categories: RMQ1, RMQ2, RMQ3, RMQ4, and RMQ5 (with RMQ1 associated with the smallest *Q* value and RMQ5 the greatest), based on the frequency distribution of *Q* and RMR values determined from the Exploratory Studies Facility and augmented with data from borehole logs (CRWMS M&O, 1997b). The range of *Q* and RMR values associated with each quality category is different for each of the stratigraphic units that comprise the repository host rock [i.e., the middle nonlithophysal, lower lithophysal, and lower nonlithophysal units of the Topopah Spring Welded Tuff (CRWMS M&O, 2000m, Figure 5)]. It is expected that approximately 75 percent of the repository block would lie within the lower lithophysal unit, but the part of the Exploratory Studies Facility that intersects the repository host rock lies mainly within the middle nonlithophysal unit. A second exploratory drift, the cross-block drift, was excavated to obtain more data for the lower lithophysal unit. Although the fracture data from the cross-block drift have been reported (CRWMS M&O, 2000n), the resulting *Q* and RMR data have not been

[11]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Repository Design and Thermal-Mechanical Effects (February 6–8, 2001)." Letter (February 28) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

compiled in any DOE report known to the NRC staff. The available *Q* and RMR data have been combined with intact rock data from laboratory testing (CRWMS M&O, 1997b) to determine the values of rock-mass mechanical properties using empirical relationships from the literature (CRWMS M&O, 1997a).

This DOE approach to mechanical characterization is generally consistent with the current methods of accounting for the effects of fractures on the mechanical characteristics of rock masses (e.g., Barton, et al., 1974; Bieniawski, 1979; Hoek and Brown, 1997). There are, however, two concerns about the DOE implementation of the approach: (i) DOE uses empirical relationships (between rock-mass quality indices and mechanical properties) from the literature without sufficient site-specific data to verify the applicability of the relationships to the site and, hence, to determine the uncertainties associated with using such relationships; and (ii) DOE has not presented sufficient information to permit an independent assessment of the appropriateness of the intact rock data used in conjunction with the rock-mass quality indices to evaluate the rock-mass mechanical properties. To address these concerns by license application, DOE agreed at the DOE and NRC Technical Exchange on Repository Design and Thermal-Mechanical Effects[12] to provide additional documentation. These concerns are best illustrated through a discussion of the specific rock-mass mechanical properties, as in the following.

<u>Rock-Mass Young's Modulus, E_m:</u> DOE determined values of E_m using two empirical relationships from the literature (Serafim and Pereira, 1983; Palmstrom, 1996) and examined the sensitivity of the calculated E_m to the scan-line or full-periphery data and to different methods of interpreting the *Q* and RMR values for the empirical relationships (CRWMS M&O, 1997a). The results show E_m values for Topopah Spring Welded Tuff rocks in the range 8.98–14.62 GPa [1,302.5–2,120.5 ksi] for the RMQ1 and 24.46–45.08 GPa [3,547.7–6,538.4 ksi] for RMQ5. DOE concluded (CRWMS M&O, 1997a, p.74), based on the variability of these results, that "*In-situ* field testing from several spatially correlated intervals within each thermomechanical unit in the Exploratory Studies Facility Main Loop is recommended to validate the range of empirically based rock mass modulus estimates." In March 1997, DOE expressed a similar conclusion (CRWMS M&O, 1997b, Table 2-16) that the information available on rock-mass stiffness would not satisfy the DOE standard for either the viability assessment or license application. The site-specific E_m data collected by DOE to date (based on information known by NRC staff) consist of six data points from Exploratory Studies Facility convergence analyses and one data point each from plate-loading and Goodman-Jack tests. As argued earlier (NRC, 2000a), these data are too sparse [in its coverage within the E_m-versus-*Q* (or RMR) space] to provide a reliable estimate of the uncertainties associated with using the empirical relationships from the literature. E_m is important because the induced thermal stress is directly proportional to the rock-mass stiffness. Consequently, the induced thermal stress can be known no better than the uncertainty in the rock-mass stiffness. Therefore, the predicted performance of underground openings under thermal-loading conditions is at best as uncertain as the knowledge of the rock-mass stiffness.

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[12]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Repository Design and Thermal-Mechanical Effects (February 6–8, 2001)." Letter (February 28) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

<u>Rock-Mass Strength:</u> DOE determined the values of rock-mass strength parameters for implementing the Mohr-Coulomb strength criterion (friction angle, Φ_m, and cohesion intercept, c_m) using an empirical approach developed by Hoek and Brown (1997). The Hoek-Brown approach consists of using the Hoek-Brown failure criterion (Hoek and Brown, 1980, 1997) to calculate sets of σ_1-versus-σ_3 values (where σ_1 and σ_3 are the maximum and minimum principal compressive stresses) to define the failure envelope for a rock mass and fitting a straight line to the results to determine Φ_m and c_m. Hoek and Brown (1997) indicated that the values of Φ_m and c_m determined using this approach are sensitive to the range of σ_3 values and the values of the intact-rock parameters—unconfined compressive strength, σ_{ci}, and Hoek-Brown parameter, m_i—used to generate the failure envelope. The intact rock parameters σ_{ci} and m_i should be evaluated using statistical analyses of laboratory triaxial-test results obtained with values of σ_3 in the range $0 < \sigma_3 < 0.5\sigma_{ci}$ (Hoek and Brown, 1997).

The DOE implementation of the Hoek-Brown approach using Topopah Spring Welded Tuff data from the Exploratory Studies Facility gave Φ_m = 56–57° and c_m = 1.9–2.6 MPa [0.276–0.377 ksi] for the RMQ1 rock-mass category and Φ_m = 58° and c_m = 3.9–6.6 MPa [0.566–0.957 ksi] for RMQ5, based on straight-line fits to the strength envelope for σ_3 values in the range $0 \leq \sigma_3 \leq 3$ MPa [0–0.44 ksi] (CRWMS M&O, 1997a). A revision of the calculation using strength envelopes in the range $0 \leq \sigma_3 \leq 42$ MPa [0–6.1 ksi] (CRWMS M&O, 2000k) gave Φ_m = 37° and c_m = 8 MPa [1.2 ksi] for RMQ1, and Φ_m = 42–43° and c_m = 12–13 MPa [1.7–1.9 ksi] for RMQ5. The two sets of strength parameters [i.e., the original set from CRWMS M&O (1997a) and the revised set from CRWMS M&O (2000k)] are given in CRWMS M&O (2000k, Tables 4-5a and 4-5b), but the original set was used for continuum analyses of the stability of the emplacement drifts. The five sets of continuum thermal-mechanical analyses presented in CRWMS M&O (2000k, Figures 6-22, 6-23, and 6-27) were based on Φ_m = 56° and c_m = 2 MPa [0.3 ksi] for RMQ1 and Φ_m = 58° and c_m = 4.1 MPa [0.6 ksi] for RMQ5. One analysis was presented based on Φ_m = 37° and c_m = 2 MPa [0.3 ksi] for RMQ1 (CRWMS M&O, 2000k, Figure 6-29), and the failure zone predicted from this analysis (for an unsupported opening) extended into the rock mass from the drift wall approximately 2.5 times as much as the failure zone predicted using Φ_m = 56° and c_m = 2 MPa [0.3 ksi].

The friction angle values suggested in the original strength-parameter set are significantly larger than the values commonly encountered in the literature. For example, an implementation of the Hoek-Brown approach in Hoek and Brown (1997) using Q = 0.53 for RMQ1 and Q = 12 for RMQ5 [based on CRWMS M&O (2000k)] would give Φ_m = 23–40°for RMQ1 and Φ_m = 27–47° for RMQ5, for m_i values in the 5–35 range. The Hoek-Brown implementation of the approach (Hoek and Brown, 1997, Figure 8) suggests a maximum Φ_m value of approximately 52 degrees for a rock mass with Q = 166, which is at least one order of magnitude greater than the Q values of approximately 0.5–15 for the repository host rock mass. The Hoek-Brown implementation would, therefore, imply much smaller values of Φ_m for the repository rock mass than the values of Φ_m suggested in CRWMS M&O (1997a).

DOE addressed this concern by providing the revised strength-parameter set (CRWMS M&O, 2000k) based on an application of the Hoek-Brown approach (Hoek and Brown, 1997) using a broader range of confining pressure than the range used to obtain the original strength-parameter set (CRWMS M&O, 1997a). The use of a broader range of confining pressure, however, addresses only one of the staff concerns regarding the

CRWMS M&O (1997a) strength-parameter set. There are still unresolved concerns that can potentially affect the values of the rock-mass strength parameters. First, the value of m_i used for the calculations was specified as 20 [based on CRWMS M&O (1997b)], but the laboratory data used to evaluate m_i or the range of the m_i values were not provided. Second, the value of σ_{ci} was based on conventional unconfined compression test data without any adjustments to account for the effects of sustained loading (infinitely slow loading rates) at the site. The relationship between the unconfined compressive strength of intact rock under fast loading (conventional loading rates used for laboratory testing) and sustained loading (slow loading rates that occur *in situ*) is well documented in the literature (e.g., Lajtai and Schmidtke, 1986; Martin and Chandler, 1994). The effect of the relationship is that only approximately 50 percent of the laboratory intact-rock strength is applicable to site conditions, considering the loading-rate effects only. DOE uses 100 percent of the laboratory σ_{ci} value and has not presented the technical basis for doing so.

<u>Rock-Mass Thermal Expansivity, α_m:</u> The thermal-mechanical analyses for site recommendation (CRWMS M&O, 2000k) were conducted using average intact-rock thermal-expansivity for the repository-level stratigraphic units, based on laboratory data from CRWMS M&O (1997b). DOE argued that the use of intact-rock thermal expansivity, instead of rock-mass expansivity, would be adequate for assessing the stability of underground openings because the intact-rock expansivity would result in greater-than-anticipated stresses. The NRC staff agree that the intact-rock thermal expansivity would give upper-bound estimates of the anticipated thermal expansion of the rock mass at a given location, but using an average thermal expansivity for the different stratigraphic units may result in a misleading assessment of the stability of the emplacement drifts. Because the stratigraphic interfaces are approximately horizontal, the differences in thermal expansivity between the stratigraphic units will likely increase the thermally induced shear stress on the subhorizontal fractures. Because slip on the subhorizontal fractures is potentially the dominant rock-failure mechanism in the emplacement area, the features of the environment that may affect the magnitudes of shear stress on the subhorizontal fractures deserve specific attention. DOE stated (CRWMS M&O, 2000k) that the differences between the intact-rock expansivity for the different stratigraphic units (CRWMS M&O, 1997b, Table 5-15) are not significant. The differences may be significant, however, because of their potential effect on slip on the subhorizontal fractures; therefore, DOE should develop sufficient technical information to evaluate the significance.

<u>Rock-Mass Thermal Properties</u>

DOE uses intact-rock thermal properties (thermal conductivity, specific heat capacity, and density) to characterize the rock-mass thermal behavior (CRWMS M&O, 2000k). As discussed (NRC, 2000a), the NRC staff agree that the thermal response of a rock mass (evolution of temperature distributions around a buried heat source in the rock mass) can be assessed satisfactorily using the intact-rock thermal properties.

<u>Fracture-Surface Mechanical Properties</u>

The fracture-surface mechanical properties, which are used for discontinuum modeling, are the stiffness parameters (shear and normal stiffness), the strength parameters (friction angle and cohesion), and the postfailure dilation parameter. DOE reported fracture-surface mechanical properties from two sources. First, CRWMS M&O (1997b) gives data from laboratory testing of core specimens. The data consist of normal stiffness of approximately 74 MPa/mm [271 ksi/in] from 11 Topopah Spring Welded Tuff core specimens tested with a normal stress of 2.5 MPa, [0.36 ksi] and a friction angle of approximately 41° from 12 Topopah Spring Welded Tuff core specimens (5 lower nonlithophysal, 5 lower lithophysal, and 2 middle nonlithophysal). Second, friction angles in the range 60–64 degrees were determined for Topopah Spring Welded Tuff fracture surfaces based on an interpretation of Exploratory Studies Facility fracture data. The interpretation, however, included an incorrect assumption that the residual friction angle of fractures is equal to the rock-mass friction angle (CRWMS M&O, 1997a, Section 7.3), which provides a possible explanation for the unusually high values of fracture friction angle from the Exploratory Studies Facility data.

The laboratory fracture data (CRWMS M&O, 1997b, Tables 5-39 and 5-40) are potentially useful, but DOE needs to determine if the data are representative of the site and provide the associated technical bases. Furthermore, no information has been provided about the fracture shear stiffness, dilation, or variation of shear or normal stiffness with normal stress.

Spatial and Temporal Variations of Mechanical Properties

Rock-mass mechanical properties vary both vertically and laterally at Yucca Mountain because of the site stratigraphy and variations in the mechanical properties of intact rock and fractures, other fracture properties (such as frequency, spacing, and continuity), and lithophysae content. The mechanical properties may also vary with time because of potential changes resulting from coupled thermal-hydrological-chemical-mechanical processes.

Spatial Variation of Mechanical Properties: DOE (CRWMS M&O, 2000k) stated that using the mechanical properties for the RMQ1 and RMQ5 rock-mass categories in thermal-mechanical analyses adequately represented the spatial variation of mechanical properties at Yucca Mountain because these two rock-mass categories envelop the worst and best expected rock conditions at the site. To support this argument, DOE needs to demonstrate the validity of two premises: (i) that the range of rock-mass quality determined from the Exploratory Studies Facility and, possibly, the cross-block drift, envelops the qualities within the repository block; and (ii) that the quality classification based on the *Q* and RMR indices, which were developed to account for the effects of fractures, is applicable to the lower lithophysal rock unit, in which lithophysae are expected to contribute significantly to the mechanical behavior.

Time-Dependent Degradation of Mechanical Properties: Time-dependent degradation of the repository host rock was not discussed in the DOE thermal-mechanical analyses for site recommendation (CRWMS M&O, 2000k), but is potentially important because an operational life up to 175 years with possible extension to 300 years may be expected for the ground-support system (CRWMS M&O, 2000d). A DOE expert panel on drift stability (Brekke, et al., 1999) indicated that degradation of the rock mass can be expected because of coupled thermal-hydrological-mechanical processes operating over a long period of time. Thermal, water-pressure, and rock-stress gradients that occur in the rock mass after the emplacement of nuclear waste would drive processes such as thermally induced fracture propagation, rock loosening, and cyclical evaporation and condensation of water. Such processes can be expected to cause degradation of the rock mass.

Rock-mass degradation related to the geochemical response of the system to elevated temperature can also be expected. Heat generated from nuclear waste is expected to cause a geochemical response because mineral stabilities and equilibria depend on temperature; geochemical reaction rates in the presence of water would accelerate at elevated temperature; and the thermal gradients would cause redistribution of moisture, solutes, and carbon dioxide, which are essential to the chemical reactions (Murphy, 1993). Reaction-path modeling of the natural gas-water-rock geochemical system at Yucca Mountain (Murphy, 1993) indicates that the anticipated geochemical reactions include dissolution of feldspars; precipitation of secondary minerals, such as clinoptilolite, smectite, and calcite; and increase in pH and aqueous sodium bicarbonate concentrations. Although the repository-induced mineralogical changes are likely to affect only a small rock volume, the changes are expected to be localized at fluid-rock interfaces such as fracture walls and lithophysal cavities. Consequently, the alteration minerals would be expected to develop as lithophysal-cavity deposits or fracture coatings.

Mineral-alteration products currently occur at Yucca Mountain mostly as fracture coating and as lithophysal-cavity deposits (Carlos, et al., 1995). The mineralogy, thickness, and amount and uniformity of coverage of fracture coatings are highly variable and uncertain (Thoma, et al., 1992). The coatings consist mainly of zeolites, manganese oxide minerals, silica phases, carbonates (mostly calcite), and clay minerals (mostly smectite but occasionally illite). Smectite is fairly ubiquitous in fractures throughout the volcanic sequence (Carlos, et al., 1995). The genesis of the fracture coatings at Yucca Mountain is not well understood, but the coatings are generally secondary minerals formed as alteration products of primary minerals such as glass, feldspar, and silica phases (Murphy, 1993; Carlos, et al., 1995; Levy, et al., 1996).

If the fracture coatings that develop after waste emplacement consist dominantly of quartz and other silica phases (e.g., Lin and Daily, 1984; Daily, et al., 1987; Matyskiela, 1997), the shear strength of fractures and, therefore, the rock-mass strength can be expected to increase. If fracture coatings consist mainly of secondary minerals, such as smectite and calcite that are mechanically weaker than the primary minerals (Kenney, 1967; Mitchell, 1976), a weakening of the fractures and, therefore, the rock mass can be expected. The secondary minerals would develop either as fracture-wall precipitates from aqueous solutions or in-place alteration products of fracture-wall rock. The result would be a change in the mechanical characteristics of fractures within the affected zone from their current classification as generally “rough, irregular, and tightly healed” to a mechanically weaker category of generally “wide and filled with clay minerals (or other alteration products) thick enough to prevent wall-rock contact” (Barton, et al., 1974).

The magnitudes, rates, and spatial distributions of the anticipated degradation of the repository host rock will be difficult, if at all possible, to evaluate. However, degradation of the host rock can reasonably be expected (Brekke, et al., 1999), and it can produce a significant impact on the stability of the emplacement drifts (NRC, 2000a). Therefore, degradation of the host rock should be accounted for in assessing the performance of the subsurface structures, systems, and components and adequate technical basis provided to support the approach used to account for it.

Uncertainties in Mechanical Properties

Mechanical-property uncertainties were not discussed in the DOE analyses of ground-support performance for site recommendation (CRWMS M&O, 2000k). CRWMS M&O (1997b), Table 2-9, for example, indicates a mean value of 104 MPa [15.1 ksi] with a standard deviation of 61 MPa [8.8 ksi] for the unconfined compressive strength of the lower lithophysal intact rock, but this uncertainty in the intact-rock strength is not reflected in the ground-support design analyses (CRWMS M&O, 2000k). As discussed earlier, there are considerable uncertainties in all the mechanical properties needed for design analyses. The influence of such uncertainties on the assessment of the performance of the subsurface structures, systems, and components should be clearly identified, and the identification should be supported with adequate technical basis.

As previously discussed, DOE agreed to address these NRC concerns regarding specific rock-mass mechanical properties during the DOE and NRC Technical Exchange on Repository Design and Thermal-Mechanical Effects.[13]

Subsurface Ground-Support Systems Design

There is currently no outstanding NRC staff concern about design methodology. NRC has accepted the DOE proposed design methodology in DOE (1997). There are, however, several concerns with the DOE implementation of the design methodology as discussed previously in this Subsection.

Subsurface Ventilation System Design

Text for this section will be provided at a later date.

Subsurface Power and Power Distribution Systems Design

Text for this section will be provided at a later date.

Maintenance Plan for Subsurface Facility Design

Text for this section will be provided at a later date.

2.1.7.3.3.3 Waste Package and Engineered Barrier Subsystem Design

Engineered Barrier Subsystem and Controls Are Adequately Designed

The acceptance criterion for waste package and engineered barrier subsystem structures, systems, and components and their controls addresses the need to prevent waste form

[13]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Repository Design and Thermal-Mechanical Effects (February 6–8, 2001)." Letter (February 28) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

degradation and provide containment, criticality control, shielding, and thermal control of the high-level waste during the preclosure period. In addition to the waste package, other engineered barrier subsystem structures, systems, and components that may be used to achieve these requirements include, but are not limited to, drip shields, waste package pallet supports and invert, backfill, and sorption barriers. To demonstrate that this acceptance criterion has been satisfied, DOE must provide a description and assessment of the components for the various types of waste packages including containers and internal structures. This information must also be provided for other relevant important to safety engineered barrier subsystem components (e.g., drip shield, waste package supports and invert, and such).

Specific information expected from DOE includes the following: (i) identification of the materials, methods, and processes used in the fabrication of containers, internal waste package components, and engineered barrier subsystem components (must be consistent with accepted design criteria, codes, standards, and specifications); (ii) specifications for container and internal waste package materials that are in agreement with those established in the final design (including consideration of the specifications for the closure welding, preparation for welding, materials to be used in the welds, and inspection of the welds that comply with applicable American Society of Mechanical Engineers codes); (iii) basis for nondestructive examination methods used to detect and evaluate defects that may lead to premature failure of the fabricated containers and other structural components of the waste packages; (iv) criticality design criteria consistent with those used in model calculations that support the design; (v) analyses demonstrating that the shielding provided by the containers is sufficient (including estimates of dose rates, a description of the source of data for the evaluation and the methods for estimating dose rate, and identification of the computational codes used); (vi) analyses demonstrating that the components of the waste package and internals are designed to sustain loads from normal operation and Categories 1 and 2 event sequences; (vii) analyses demonstrating that thermal control is such that the fuel cladding temperature will be sufficiently low to prevent cladding failure; (viii) evidence the materials used in construction of the internal components of the waste package are compatible with the waste form; (ix) analyses demonstrating the design of any drip shield, including materials of construction, configuration, and method of emplacement, is sufficient to prevent water from contacting the waste packages and does not impair safe handling of the waste package during subsurface maintenance operations; (x) analyses demonstrating that the design of any backfill, including materials and physical characteristics, configuration, and methods of emplacement and compaction, is adequate to reduce the relative humidity near the waste packages; and (xi) analyses demonstrating that the design of any sorption barrier is adequate to control the migration of radionuclides and materials. The postclosure performance of the engineered barrier subsystem is addressed in Sections 3.3.1 and 3.3.2.

Overall, the current information, along with the information to be provided according to the agreements reached between DOE and NRC in the Container Life and Source Term,[14]

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[14]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000)." Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

Repository Design and Thermal-Mechanical Effects,[15] Preclosure Safety,[16] and Range of Operating Temperatures[17] Technical Exchanges, is sufficient to conclude that the necessary information needed to assess the design of the waste package and engineered barrier subsystem structures, systems, and components and safety controls will be available at the time of a potential license application. The designs of the waste package, drip shields, and the waste package pallet have yet to be finalized. In addition, the fabrication, remediation, and waste package and drip shield emplacement methods are currently being developed.

Waste Package Design Description

The current waste package design consists of two concentric cylinders (i.e., disposal containers, fabricated from plate material). The inner disposal container will be fabricated using Type 316 nuclear grade stainless steel that is a minimum of 50 mm [1.97 in]-thick (CRWMS M&O, 2001a). The inner disposal container will fit inside the outer disposal container that is constructed from 20-mm [0.79-in]-thick Alloy 22. A radial gap of 0 to 4 mm [0 to 0.16 in] will be used between the inner and outer disposal containers to allow for differential thermal expansion to occur without introducing thermally induced stresses. The axial gap between the inner and outer disposal containers, which may be more important as far as differential thermal expansion stresses are concerned, is 10 mm [0.39 in] (CRWMS M&O, 2000e). Type 316 nuclear grade stainless steel was selected for the inner disposal container to provide mechanical integrity to the waste package during both the preclosure and postclosure periods of the proposed repository. The selection of Alloy 22 as the outer disposal container material was based on the resistance of this nickel-chromium-molybdenum-tungsten alloy to both localized corrosion and stress corrosion cracking in chloride-containing environments. Placement of the corrosion-resistant Alloy 22 container on the outside of the Type 316 nuclear grade stainless steel is designed to provide long-term protection of the inner container material (CRWMS M&O, 2000f).

There are several waste package configurations for the site recommendation waste package design needed to encapsulate the various commercial spent nuclear fuel waste forms (CRWMS M&O, 2000f). These configurations include designs for pressurized water reactor fuel containing either 12 or 21 pressurized water reactor assemblies with absorber plates and 21 pressurized water reactor assemblies with control rods. Two waste package configurations are required for boiling water reactor fuel that contains either 44 boiling water fuel assemblies with absorber plates or 24 boiling water reactor fuel assemblies with thick absorber plates.

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[15]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Repository Design and Thermal-Mechanical Effects (February 6–8, 2001).” Letter (February 28) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[16]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001).” Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[17]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Range of Operating Temperatures.” Letter (October 2) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

Moreover, there are additional waste package configurations for the disposal of defense high-level waste and DOE-owned spent nuclear fuel.

The waste package will be constructed by rolling the plate materials into cylinders. A longitudinal weld will be used to complete the cylinder. Welding will also be used to connect two cylinders together to provide sufficient length for the spent nuclear fuel and high-level waste. The bottom lids of the disposal containers are also welded in place. Although the Type 316 nuclear grade stainless steel inner disposal container provides mechanical integrity to the waste package, the Alloy 22 outer disposal container will be required to sustain loads during lifting and transport. Lifting trunnions will be attached to the outer surface of the Alloy 22 disposal container to facilitate the necessary lifting and transport operations. The design of the inner disposal container will be specific to the waste package contents. Unique internal support structures are required for pressurized water reactor fuel, boiling water reactor fuel, and high-level waste glass (CRWMS M&O, 2000f). After the internal support structure is constructed inside the inner disposal container, the inner Type 316 nuclear grade stainless steel container will be inserted into the Alloy 22 outer disposal container. After the loading of the disposal containers, the containers will be sealed with lids that are welded in place. One lid is used for the Type 316 nuclear grade stainless steel, and a dual-closure lid design is used for the Alloy 22 outer disposal container (CRWMS M&O, 2000e).

In summary, the waste package design description appears to incorporate design features for containment. The design of the waste package is still under development, so DOE will provide additional design information in future documents. These documents will be reviewed as they become available.

Waste Package Internal Components Design Description

Internal components of the waste packages include basket guides, corner guides, fuel tubes, and defense high-level waste canister guides (CRWMS M&O, 2000f). The internal components are designed to facilitate heat transfer from the interior of the waste package to the exterior surface of the outer disposal container, by way of thermal conduction, to keep fuel cladding temperatures within specified limits, control criticality, and provide structural support to the waste package. In addition, the materials used in the waste packages must be compatible with the waste form, spent nuclear fuel cladding, and the waste package disposal container materials. The materials should not be reactive or pyrophoric.

The design of the waste packages for commercial spent nuclear fuel also contains stainless steel boron alloy plates (absorber plates) to provide criticality control. When criticality control is provided by the spent nuclear fuel control rods, the absorber plates are replaced with carbon steel plates to provide structural support and maintain the desired geometric configuration. The internal structure must maintain the desired geometric configuration when subjected to mechanical loads to provide criticality protection during handling, emplacement, and retrieval (CRWMS M&O, 2000f). In addition, the material used to provide criticality control must be compatible with the other materials and components inside the waste package and must not degrade the waste form. DOE identified Neutronit A978, which is similar in composition to Type 316L stainless steel with 1.6 percent boron added, as the material that will be used for the absorber plates.

The DOE description of the internal components of the waste package includes the necessary components for configuring the waste, providing criticality control, and transferring heat necessary to keep the internal temperature of the waste packages below design limits (see the appropriate topical discussions provided in this section for additional details pertaining to criticality design criteria and fuel cladding temperature control). The design of the waste package is still being developed, so DOE will provide additional design information in future documents. These documents will be reviewed as they become available.

Drip Shield Design Description

The description of the drip shield, its fabrication sequence, and the emplacement methods are not complete. The design of the drip shield is still under development (CRWMS M&O, 2001a). The current drip shield design calls for a Titanium Grade 24 support structure covered with 15-mm [0.59-in]-thick Titanium Grade 7 plate. Individual segments of the drip shield are connected together using a vertically sliding interlock configuration. The drip shield will be installed at the end of the preclosure period. The intended function of the drip shield is to divert any dripping water from contacting the waste packages and protect the waste package against rockfall and drift collapse in the postclosure period (CRWMS M&O, 2001b). Emplacement of the drip shields at earlier times would prevent the inspection of the waste packages when using remotely controlled inspection gantries (CRWMS M&O, 2000p).

DOE has provided a conceptual design description for the drip shield, including the materials of construction, configuration, and method of emplacement. Details of the fabrication methods have yet to be provided, however. An assessment of the ability of the proposed drip shield to withstand mechanically disruptive events for the postclosure period is provided in Section 3.3.2.4.4.1. Even though all potential postclosure design basis events are not applicable to the preclosure period, the comments pertaining to the general analysis methodology used by DOE to demonstrate the structural integrity of the drip shield are relevant to the preclosure safety case. DOE will provide additional design information in future documents. These documents will be reviewed as they become available.

Waste Package Pallet

The waste package pallet is designed using Alloy 22 plate material (CRWMS M&O, 2000g). Each waste package pallet has two V-shaped supports that are connected together using stainless steel rails. Two sizes of emplacement pallets will be required to accommodate the different waste package lengths.

DOE performed structural evaluations of the emplacement pallet corresponding to static loading by the waste package and lifting during handling operations (CRWMS M&O, 2000q,r). The results of analyses used to support these structural evaluations are reported using stress intensity values. Because no clear definition of stress intensity was provided, however, it has been assumed that the reported values of stress intensity are consistent with the definition provided in American Society of Mechanical Engineers (2001, Subparagraph NB–3213.1). In addition, it is not clear if the normal stress components generated at the contact interface between the waste package and pallet were taken into consideration when calculating the stress intensity results presented in the reports. Seismic loads were not addressed in the lifting of a loaded pallet structural evaluation. DOE must either assess the effects of seismic loads on a loaded pallet for all relevant handling operations or justify their exclusion. Similarly, DOE must assess the potential consequences of dropping a loaded emplacement pallet or provide the basis for excluding this particular event from consideration.

Disposal Container Fabrication and Closure

The disposal container will be fabricated according to American Society of Mechanical Engineers (1995a, Section III, Division 1, Subsection NB, Class 1 Rules for Construction of Nuclear Power Plant Components) to the maximum extent practicable (CRWMS M&O, 2001a). Deviations from the code will be documented and submitted for approval, but the disposal containers will not be nuclear or “N”-stamped pressure vessels (CRWMS M&O, 2001a). American Society of Mechanical Engineers (1995a) provides a standard for the fabrication of the disposal containers and requirements for inspection.

American Society of Mechanical Engineers (2001) provides rules for construction with the objective of protecting life and property, and a margin for deterioration in service, to assure a safe period of usefulness for boilers, pressure vessels, and nuclear components. The official American Society of Mechanical Engineers Boiler and Pressure Vessel Code symbol stamp may only be used to identify components constructed in accordance with the applicable rules of the code, which include requirements for materials, design, fabrication, examination, and inspection. Items not constructed in accordance with rules of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code may not be stamped, and such items may not meet the objectives of the code. DOE stated that the materials used in the fabrication of the disposal containers and the drip shield will meet the requirements in American Society of Mechanical Engineers (1995a, Section III, Division 1, Article NB–2000).

Filler materials used in welding processes must conform to the requirements specified in American Society of Mechanical Engineers (1995b, Section II, Part C). For the Type 316 nuclear grade stainless steel inner container, the filler material will be selected to control the delta ferrite content of the as-deposited weld metal. A ferrite number between 5 and 15, determined by Magna-gage measurements, is required in the inner disposal container fabrication welds (CRWMS M&O, 2001a). The weld filler material for the Alloy 22 outer container will be ENiCrMo-10 or a filler material used for welding alloys with the UNS (Unified Numbering System) number N06022 designation (CRWMS M&O, 2001a).

The preparation of the disposal containers and the procedures for welding will be in accordance with American Society of Mechanical Engineers requirements (1995c, Section IX). Welding will not be performed if the temperature of the base metal is lower than 0 °C [32 °F]. The maximum interpass temperature for austenitic stainless steels (including Type 316 nuclear grade stainless steel) and nickel alloys (including Alloy 22) is 175 °C [347 °F]. Each weld layer is required to be free of slag, inclusions, cracks, unacceptable porosity, and lack of fusion. Welding processes for the fabrication of the disposal containers may include shielded metal arc, gas tungsten arc, submerged arc, and gas metal arc, provided the processes are qualified (CRWMS M&O, 2001a).

Defects in the disposal container can be repaired by welding provided that the requirements in American Society of Mechanical Engineers (1995a, Section III) are met. All material defects and repairs must be appropriately documented (CRWMS M&O, 2001a). Weld repairs will be performed in accordance with American Society of Mechanical Engineers requirements (1995a, Section III, Division I, Article NB–4000). Only three repair cycles will be permitted without special approval (CRWMS M&O, 2001a). DOE did not provide any rationale or basis for this specification.

Fabrication of both the inner and outer disposal containers involves cutting, rolling, and welding operations. Fabrication of the cylinders that form the sides of the disposal containers is similar for both the inner and outer containers. After the plates are inspected, they are cut to form the cylinders and lids. The plates are then rolled into cylinders. The dimensions of the cylinders are adjusted to assure the final design dimensions can be achieved and to minimize distortion from welding. The longitudinal seam is then welded, and the completed weld is inspected. After the ends of the cylinders have been satisfactorily prepared, the two cylinders are welded together. A dimensional inspection is then performed, and if needed, the cylinder is machined to tolerance.

The remaining fabrication steps for the disposal containers are specific to the inner and outer containers. For the Type 316 nuclear grade stainless steel inner container, the bottom lid and the internal parts, such as baskets, corner guides, and separator plates, are installed. For the Alloy 22 outer container, an assembly support ring used to support the Type 316 nuclear grade stainless steel inner containers is welded into place, and the welds are machined to allow the inner cylinder to be properly installed into the outer container. The bottom lid is then fit and welded in place. The trunnion collar sleeve is then installed on the outside of the Alloy 22 outer container and welded in place. Solution annealing is performed at approximately 1,125 °C [2,057 °F] to eliminate residual stresses created during the fabrication processes. The solution annealing should also dissolve any secondary phase precipitates such as topologically close packed phases formed as a consequence of the welding processes. The Alloy 22 outer container is annealed in a furnace on a furnace car. The furnace car is used to transport the disposal container out of the oven where it is sprayed with water on both the inside and outside surfaces. The water quench is designed to reduce the temperature of the Alloy 22 outer container from 1,150 °C [2,102 °F] to below 800 °C [1,472 °F] in approximately 4 minutes. The cooling rate is then decreased to allow for the formation of compressive stresses.

For the inner Type 316 nuclear grade stainless steel container, the closure lid and shear rings are installed, and a seal weld is used to hold the shear rings in place using the gas metal arc weld method, which allows faster deposition rates (Stephenson, 1990). The evaluation of an Alloy 22 closure lid welding method has recently been reported (CRWMS M&O, 2001b). Welding methods considered were narrow groove gas tungsten arc welding, optimized gas tungsten arc welding, and plasma arc welding. The selection criteria considered, in decreasing weight of importance, were process recovery, residual stresses, equipment reliability, production rate, fit-up tolerances, remote operation capability, radiation hardening, and industrial experience. Plasma arc welding was rated the best for residual stresses and production rates. Optimized gas tungsten arc welding was rated the best for radiation hardening considerations. For all other selection criteria, the narrow groove gas tungsten arc welding method was determined to be the best method for the Alloy 22 closure lids.

To reduce residual stresses in the Alloy 22 final closure welds, laser peening is used on the inner Alloy 22 closure lid weld. Details of the process have not been reported. For the outer closure lid, local induction annealing of the extended outer shell is proposed as a method to eliminate residual tensile stresses in the Alloy 22 outer closure weld. Although the process is under development, the proposed induction annealing process would be used to heat the end of the Alloy 22 disposal container with the completed closure weld to a temperature of 1,150 °C [2,012 °F]. Forced air or water will be used to rapidly reduce the temperature of the closure weld region (CRWMS M&O, 2001b). Because the process is still under development, specifications for cooling times and temperature distributions have not been established.

The combination of cold work used in forming and machining operations and elevated temperature exposures as a result of welding and annealing processes may result in the precipitation of topologically close packed phases. During the solidification of the weld metal, molybdenum and tungsten segregate to the interdendritic regions leaving the dendrite core rich in nickel (Cieslak, et al., 1986a,b). The depletion of nickel and enrichment of molybdenum and tungsten in the interdendritic regions promote the precipitation of topologically close packed phases. The composition of all the topologically close packed phases, including σ, μ, and P phases, can contain more than 30-percent molybdenum (Raghavan, et al., 1984). The high concentration of molybdenum in these phases results in a depletion of molybdenum adjacent to the precipitates that reduces the resistance of the alloy to localized corrosion. Because the formation of the precipitates preferentially occurs in the weld regions and in the intergranular regions of the heat-affected zone adjacent to the welds, localized corrosion in the form of interdendritic and intergranular corrosion may be a consequence of the precipitation of topologically close packed phases (Heubner, et al., 1989). The ductility of σ, μ, and P phases is typically low compared with the austenitic matrix of the nickel-base alloy (Matthews, 1976; Tawancy, 1996). As a result, the precipitation of topologically close packed phases may reduce the ductility and impact strength of the alloy, particularly in welds or in the heat-affected zones of the welds.

The thermal stability of nickel-chromium-molybdenum alloys was evaluated using several criteria: (i) microstructural examination for the presence of secondary phase precipitates at the grain boundaries or in the interdendritic regions of welds; (ii) intergranular corrosion susceptibility; and (iii) mechanical properties such as ductility, yield strength, or impact toughness. Heubner, et al. (1989) provided a phase stability diagram for Alloy 22, based on microstructural examinations conducted after isothermal exposures at temperatures ranging from 550 to 900 °C [1,022 to 1,652 °F]. Heubner, et al. (1989) reported the precipitation of topologically close packed phases in times as short as 15 minutes at temperatures in the range 800–900 °C [1,022–1,652 °F]. A significant increase in the intergranular corrosion rate was observed after 1 hour at 800 °C [1,472 °F] based on the results of standardized tests (American Society for Testing and Materials International, 1999). Bulk precipitation of topologically close packed phases was reported to occur after 10 hours at 800 °C [1,472 °F] and after 3 hours at 900 °C [1,652 °F]. In contrast, the results reported by Rebak, et al. (2000) indicate complete grain boundary precipitation after 10 hours at 800 °C [1,472 °F] and bulk precipitation within the grains after 100 hours at 800 °C [1,472 °F].

The effect of topologically close packed phase precipitation on the mechanical properties of Alloy 22 has been reported at temperatures in the range 593–760 °C [1,099–1,400 °F]

(CRWMS M&O, 2000s; Rebak, et al., 2000). Table 2.1.7-1 combines the mechanical properties and corrosion rates reported by Rebak, et al. (2000) with the microstructural observations of the material after isothermal exposures. It is apparent that the corrosion rate increases in response to partial grain boundary precipitation. In contrast, the Charpy impact energy for Alloy 22, after thermal aging that results in partial coverage of the grain boundaries with topologically close packed phase precipitates, is quite high and similar to the impact energy for material in the solution-annealed condition. The reduction in area measured on tensile test specimens decreased slightly from 75 to 80 percent in the solution annealed condition to 70 to 75 percent. Complete grain boundary precipitation was required for significant decreases in ductility or impact toughness. The activation energy necessary to decrease the impact energy to 203 J [150 ft·lb] was determined to be 247 kJ/mol [59 kcal/mol].

At 760 °C [1,400 °F], the highest temperature for which Charpy data were reported by Rebak, et al. (2000), an exposure of 10 hours is required to decrease the Charpy impact energy to 203 J [150 ft·lb]. Assuming the extrapolation of activation energy is valid at temperatures greater than 760 °C [1,400 °F], an isothermal exposure after 1 hour at 870 °C [1,598 °F] would decrease the Charpy impact energy from 360 to 203 J [266 to 150 ft·lb].

Table 2.1.7-1. Relationship Between Alloy 22 Condition, Ductility, Impact Resistance, and Corrosion Rate Using American Society of Mechanical Engineers Standard Corrosion Test Methods

Alloy 22 Condition	Tensile Specimen Reduction in Area	Charpy Specimens Impact Energy, J [ft·lb]	Corrosion Rate in ASTM* G28A Test, mm/yr [in/yr]
No precipitates	75 to 80 percent	360 [266]	1 [0.04]
Precipitates partially cover grain boundary	70 to 75 percent	360 [266]	2 to 4 [0.08 to 0.16]
Complete coverage of grain boundaries	55 to 65 percent	140 to 240 [103 to 177]	4 to 20 [0.16 to 0.79]
Complete coverage of grain boundaries plus precipitation within grains	20 to 50 percent	< 100 [< 74]	> 20 [> 0.79]

*American Society for Testing and Materials. "Standard Test Methods of Detecting Susceptibility to Intergranular Corrosion in Wrought, Nickel-Rich, Chromium-Bearing Alloys." ASTM G 28-97. 2001 Annual Book of ASTM Standards. Volume 3.02. West Conshohocken, Pennsylvania: American Society for Testing and Materials. 2001.

Systematic studies on the effect of compositional variations of Alloy 22 on thermal stability have shown that molybdenum, tungsten, and iron decrease the phase stability of the alloy and increase the precipitation kinetics of topologically close packed phases (Heubner, et al., 1989). The compositional specifications for Alloy 22 include 12.5 to 14.5-percent molybdenum, 2.5 to 3.5-percent tungsten and 2 to 6-percent iron. These specifications are external specifications, and the internal specifications used at production mills are more stringent for alloying concentration variations. The ENiCrMo-10 welding filler metal compositional specifications include 2.5 to 4.5-percent tungsten, which is a broader specification range compared with Alloy 22. Variations in the composition of the Alloy 22 plate and the filler metal used in the welding process may alter the kinetics of topologically close packed phase precipitation.

Additional evaluation is needed to determine the effects of microstructural and compositional variations of the plate and filler materials on the thermal stability and mechanical properties of the Alloy 22 waste package outer container. This evaluation may result in unanticipated variations in waste package corrosion resistance and mechanical properties. To address these concerns, DOE agreed[18] to provide justification that the American Society for Mechanical Engineers Boiler and Pressure Vessel Code case for the use of Alloy 22 results in acceptable waste package mechanical properties considering allowed microstructural and compositional variations of Alloy 22 base metal and the allowed compositional variations in the weld filler metals used in the fabrication of the waste packages. In addition, DOE agreed[19] to provide justification that the mechanical properties of the disposal container fabrication and waste package closure welds are adequately represented considering the (i) range of welding methods used to construct the disposal containers, (ii) postweld annealing and stress mitigation processes, and (iii) postweld repairs. DOE indicated that future work will include development and testing of welding, heat treating, and inspection equipment and processes.

In summary, microstructural and compositional variations of the plate material and filler metals may alter the kinetics of topologically close packed phase precipitation because of welding and thermal exposures. As a result, the waste package mechanical properties may be affected by the fabrication processes used to construct and close the disposal containers. Additional information is needed to assess the effects of fabrication processes and compositional and microstructural variations on the mechanical properties of the waste package. With the DOE agreement to provide the additional information, sufficient information should be available at the time of a potential license application for NRC to make a regulatory decision.

<u>Nondestructive Evaluation of the Disposal Container</u>

Before fabrication, DOE plans to examine the plate material to be used in the fabrication of the disposal containers, according to American Society of Mechanical Engineers requirements (1995d, Section V). This examination will include an ultrasonic inspection of the plates to be

[18]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[19]Ibid.

used for fabrication of the inner and outer cylinders of the disposal container (CRWMS M&O, 2001a).

As described in previous sections, fabrication methods used for the outer and inner cylinders involve longitudinal and circumferential seam welds. DOE plans to perform nondestructive examination of both types of welds. Fabrication welds for the Alloy 22 outer cylinder will be examined using liquid-penetrant, radiographic, and ultrasonic testing techniques. In the case of the Type 316 nuclear grade stainless steel inner cylinder, however, the nondestructive examinations will be limited to liquid-penetrant testing (CRWMS M&O, 2001a).

The fabrication of the top outer lid of the disposal container is detailed in the waste package design sketch (CRWMS M&O, 2000f, design sketch SK-0175). There will be two circumferential partial penetration welds and two circumferential fillet welds involved in the fabrication of this lid. DOE does not intend to perform nondestructive examination of any of these lid fabrication welds (CRWMS M&O, 2001a).

Fabrication of the Alloy 22 outer container will include a support ring designed to hold the weight of the inner container after assembly of the two containers in a nested arrangement (CRWMS M&O, 2000f). The welds of the ring will be machined to allow the bottom lid of the outer disposal container to be installed flush to the bottom of the ring and the inner disposal container to sit on the top of the ring. The machined surfaces will be inspected using liquid-penetrant testing (CRWMS M&O, 2001a).

After the inner and outer cylinders of the disposal container are fabricated, the bottom lid for each cylinder will be welded in place. The welds will be subjected to nondestructive examinations using liquid-penetrant, radiographic, and ultrasonic testing techniques (CRWMS M&O, 2001a). The DOE does not plan to perform nondestructive examinations of any other welds in the disposal container.

DOE originally intended to perform liquid-penetrant, radiographic, and ultrasonic testing of all disposal container inner cylinder fabrication welds (CRWMS M&O, 2000t). As delineated in a revision of this report (CRWMS M&O, 2001a), DOE now plans to limit the nondestructive evaluation to liquid penetrant testing for these welds. Since liquid penetrant testing can only uncover surface flaws, this new approach will fail to detect subsurface flaws. The integrity of these welds is particularly important because the inner container is relied on to maintain the structural strength and integrity of the waste package after emplacement. American Society of Mechanical Engineers (1995a, Subarticle NB–5210), which deals with vessel welded joints, requires volumetric and surface nondestructive evaluation of the welds. DOE should justify why it intends to rely solely on liquid-penetrant testing for inspection of inner cylinder fabrication welds.

In the case of the Alloy 22 outer closure lid of the waste package, DOE plans to do volumetric nondestructive evaluation of the closure weld (CRWMS M&O, 2001b) but does not plan to carry out any nondestructive evaluation of the other welds used in fabrication of the lid. Further, DOE will also carry out liquid-penetrant, radiographic, and ultrasonic testing of the Alloy 22 bottom lid weld for the waste package (CRWMS M&O, 2001a). Because the failure of any of these component welds can lead to a failure of the waste package, it is not clear why a graded approach is being adopted for nondestructive evaluation of the various welds. DOE should clearly state the reasons for conducting varying degrees of nondestructive evaluation on the welds involved in the fabrication of the waste package.

DOE agreed[20] to provide justification that the nondestructive evaluation methods used to inspect the Alloy 22 and Type 316 nuclear grade stainless steel plate material and welds are sufficient and capable of detecting defects that may adversely affect waste package preclosure structural performance. An assessment of the nondestructive examination methods used in the fabrication of the disposal containers has not been provided. Although the applicability of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code for the design and construction of the disposal containers has not been established, the fabrication and nondestructive evaluation sequence that DOE proposed is not consistent with recent versions of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. American Society of Mechanical Engineers (1995a, Subarticle NB–5130) requires the examination of the weld edge before welding when the material is greater than 51 mm [2 in] thick. In addition, American Society of Mechanical Engineers (1995a, Subarticle NB–5210 and Paragraph NB–5221) requires the volumetric inspection of circumferential and longitudinal welds. Because the minimum thickness of the Type 316 nuclear grade stainless steel inner disposal container is 50 mm [1.97 in], some of the disposal container designs may require additional inspection before welding according to the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. The proposed use of liquid-penetrant testing as the only method to inspect the inner disposal container fabrication welds does not meet the requirements of volumetric inspection.

Nondestructive Evaluation of the Closure Welds

The waste package design involves three closure lids (CRWMS M&O, 2000f, design sketch SK–0175). Because of the high radiation fields that will be present after the containers are loaded, remote welding processes are required to close the disposal containers. Before installation of the closure lid, the prepared surfaces will be visually inspected using a remote camera, followed by a tactile coordinate measurement using a coordinate measuring machine. The coordinate measuring machine will locate the center of the disposal container, relative to the closure gantry manipulator coordinate system, and determine disposal container cylindricity. It will provide a redundant check of the visual inspection for the weld preparations. The lids will be tack welded first and then circumferentially welded using remote gas metal arc or gas tungsten arc welding methods. Three remote cameras (lead, trail, and inspection) on the robotic arm welder will provide real time weld inspection with digital image processing and machine vision techniques. In case of any alarm, the welding process will be stopped and the operator notified of the problem. It may be possible to immediately perform the repair at the weld station, and then resume the welding process. If the repair requires extensive machining, the disposal container will be moved to a repair station (CRWMS M&O, 2001b).

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[20]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

The inner disposal container lid, made of Type 316 nuclear grade stainless steel, will be 95 mm [3.74 in]-thick (CRWMS M&O, 2000e). A shear ring will be used with the inner lid. It will be assembled from three or four segments and welded in place. Gas metal arc welding will be used to perform this operation. The gas metal arc welding robotic arm will have the ability to perform a full circumferential weld with a rotational range greater than 360 degrees. All critical parameters will be recorded in process, and alarm or fault set points in the closure cell control system will notify the operator immediately of any parameter anomalies, and place a flag in the data stream. After welding the inner lid, the inner container will be evacuated and filled with inert helium gas via a purge port. The inner container will then be leak tested to confirm the integrity of the welds. The process sequence flowchart for disposal container closure (CRWMS M&O, 2001b) indicates DOE does not plan to conduct a nondestructive examination of the inner container lid weld.

The middle lid, made of Alloy 22, will be 10 mm [0.39 in]-thick and will be welded to the outer barrier using a partial penetration weld. The original square root partial penetration weld design may be modified to include a chamfer at the root of the weld. The gas tungsten arc welding method is presently being considered for remote welding of this lid (CRWMS M&O, 2001b). The welding sequence will be similar to that described in the previous two paragraphs. There will be a remote visual inspection of the weld preparation surfaces followed by a dimensional inspection using a tactile coordinate measuring system, tack welding, and then circumferential welding of the lid. Nondestructive evaluation of the weld will be performed to ensure acceptability. Laser peening will be used for stress relief of the weld, followed by a second nondestructive evaluation of the weld. There is no identifiable method for performing a volumetric inspection of the middle closure lid weld at present. It is expected, however, that a suitable process and tooling for this nondestructive evaluation will be developed later (CRWMS M&O, 2001b).

The extended outer shell lid is also made of Alloy 22. It will be tack welded and then circumferentially welded to the outer container using the narrow groove gas tungsten arc welding method. For the most part, the welding sequence will be similar to that described in the preceding paragraphs. Remote visual inspection of the weld preparation surfaces will be used to ensure that the surfaces are free of deposits and scale. The weld joint will be back purged using Argon, followed by tack welding, and then circumferential welding of the lid. Nondestructive evaluation of the weld will be performed to ensure acceptability. The inspection will require two passes (rotations). A surface examination will be performed using an alternating current field measurement probe, followed by a volumetric inspection using ultrasonic testing and a couplant. The weld will then be induction annealed, and the nondestructive evaluation will be repeated one final time. This strategy allows repairs to be made before the postweld heat treatment, ensuring the postweld heat treatment does not have to be repeated because it is thought that additional postweld heat treatments would be detrimental to the long-term performance of the waste package (CRWMS M&O, 2001b).

To experimentally determine the minimum detectable flaw size using ultrasonic testing, DOE fabricated two Alloy 22 mockups fabricated using 25-mm [1-in]-thick material. The plates were welded using gas tungsten arc welding with joint dimensions similar to that proposed for the outer closure weld. The Alloy 22 mockup was then machined so the dimensions were representative of the cross-sectional geometry of the extended outer shell lid of the waste package in the area of the closure weld. The lid configuration limits the available scan surfaces for ultrasonic testing to the top surface of the waste package on each side of the weld and to the side of the weld from the outside diameter of the waste package at the elevation of the weld. Geometric features in the weld area do not allow ultrasonic testing from other surfaces. The two mockups were constructed so they reflected these constraints to available ultrasonic testing locations. Each mockup contained five flaws of known dimension and location. There were two types of planar flaws, lack of fusion and lack of penetration. The third type of implanted flaw was porosity. Examinations were performed by scanning from the top of the mockup plate with 45 and 70°-angle beams directed toward the weld from each side of the weld. A straight beam scan was performed on the closure weld mockup specimen by placing the transducer on the crown of the weld. An additional straight beam scan was performed by placing the transducer on the side of the weld mockup specimen, which was machined so that the ultrasonic beam path was equivalent to the distance between the waste package outside diameter and the closure weld. The last scan orientation resulted in a sound beam traveling normal to the weld axis and was optimum for detecting fabrication flaws that follow the weld fusion line, such as lack of fusion and lack of penetration (CRWMS M&O, 2001b).

Results obtained from the scans indicated that the last scan orientation described in the preceding paragraph provided the greatest response from planar flaws. Also, planar type flaws (i.e., fusion and penetration flaws) with a minimum area of 16 mm^2 [0.025 in^2] can be detected in this weld joint geometry. Small volumetric porosity reflectors, however, were not detected, primarily because of the scattering of the sound wave from the round-shaped individual gas pores. The inability to detect small volumetric porosity reflectors may be acceptable (American Society for Mechanical Engineers, 1995e) because the geometric discontinuities associated with the individual gas pores do not cause localized increases in stress that appreciably affect the initiation of stress corrosion cracking or mechanical failure.

In summary, DOE agreed[21] to provide justification that the nondestructive evaluation methods used to inspect the Alloy 22 and Type 316 nuclear grade stainless steel plate materials and welds are sufficient and capable of detecting defects that may adversely affect waste package preclosure structural performance. Subsequent to the technical exchange agreement, DOE demonstrated, through an assessment of the ultrasonic inspection of the closure weld mockup, that flaws, such as lack of penetration and lack of fusion, can be detected (CRWMS M&O, 2001b). Further, information DOE provided subsequent to the technical exchange agreement suggests that, because of waste package weld geometry, a full volumetric inspection may not be suitable for the middle Alloy 22 closure lid. A demonstration of a suitable nondestructive evaluation of this closure lid weld will require some development and may require an adjustment to the joint geometry. Finally, for the inner Type 316 nuclear grade stainless steel closure weld, there does not appear to be a method or process to perform a remote nondestructive examination of this weld. The potential consequence of not performing a nondestructive examination of the inner disposal container closure lid has not been assessed.

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[21]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

Criticality Design Criteria

The general preclosure criticality control requirement is specified under 10 CFR 63.112(e)(6), which indicates that the structures, systems, and components must be designed in such a way that would " … prevent and control criticality … ". In its review of the preliminary preclosure safety assessment (DOE, 2001b), NRC identified the following concerns. The first was the DOE reliance on the level of the burnup in the commercial spent nuclear fuel assemblies for designing the criticality control systems of the waste packages. Another concern included consideration of events (e.g., internal and external flooding; spent nuclear fuel assembly misload events; events in the pools and storage racks; and, in general, Categories 1 and 2 events with respect to criticality), when designing the surface and subsurface facilities. Furthermore, the issues NRC identified when reviewing the DOE report (2001b) are briefly discussed.

According to NRC Regulatory Guide 3.71 (NRC, 1998), burnup of the spent nuclear fuel assemblies must be verified through measurements before they can be loaded into waste packages if the licensee chooses to take credit for the burnup when designing the criticality control system of the waste package. During the preclosure technical exchange,[22] DOE agreed to provide an approach for verification of fuel assembly burnup. DOE stated that burnup credit is only being sought for commercial spent nuclear fuel, and that burnup information for the majority of the fuel developed and available through reactor records maintained according to NRC-accepted quality assurance requirements is the best source of assembly burnup information. NRC agreed that reactor records are a more accurate source of fuel assembly burnup data than physical measurements. NRC stated that its current position, however, is that measurements are needed to verify the burnup indicated by reactor records.

Several waste package internal component configurations are considered in the determination of the effective neutron multiplication factor (i.e., k_{eff}): (i) an intact basket with a neutron absorber inside the waste package, (ii) a degraded basket with the neutron absorber flushed from the waste package and iron-oxide corrosion product uniformly distributed throughout the waste package, and (iii) a degraded basket with iron oxide settled to the lowest 3.5 rows of assemblies (CRWMS M&O, 2000f). Although the configurations with degraded baskets are more significant for postclosure performance than for preclosure performance, the analyses of the degraded configurations suggest that up to 11.2 percent of the pressurized water reactor fuel waste packages will need some additional criticality control measures. Several criticality control options have been considered including new reactor control rod assemblies, spent reactor control rod assemblies, and disposable control rod assemblies specifically manufactured for the waste packages. The zirconium clad B_4C disposable control rods are the preferred option for the site recommendation waste package design.

With respect to the consideration of events such as flooding, misload, and the like, DOE stated " … established design requirements that preclude preclosure criticality unless two unlikely

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[22]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

independent events occur [e.g., CRWMS M&O (2000u)]. The probability of two unlikely independent events occurring will be less than 10^{-6}/yr." Staff believe the double-contingency principle (i.e., two unlikely events), which has been used historically in designing criticality control systems for facilities, storage, and transportation packages, does not require the licensee to quantify the probability of the unlikely events. According to 10 CFR Part 63, however, events must be identified, their probabilities quantified, and assigned designation as Categories 1 or 2 events. On the other hand, 10 CFR 63.112(e)(6) indicates that the structures, systems, and components must be designed in such a way that nuclear criticality is prevented. Therefore, as DOE has indicated, the repository preclosure structures, systems, and components will be designed to prevent criticality under normal operation and Categories 1 and 2 events.[23]

Waste Package Shielding

The current site recommendation waste package design does not provide additional shielding for personnel protection (CRWMS M&O, 1999b). It is intended that the waste package containment barriers provide sufficient shielding to protect the waste package materials from radiation-enhanced corrosion (CRWMS M&O, 2000f). The maximum dose rate on the external surfaces of the waste package with 21 pressurized water reactor fuel assemblies is 13.30 ± 0.60 Sv/hr [1,330 ± 60 rem/hr], whereas the maximum dose rate for a waste package with 44 boiling water reactor fuel assemblies is 14.09 ± 0.32 Sv/hr [1,409 ± 32 rem/hr] (CRWMS M&O, 2000e). Shielding for staff protection is to be achieved by operational procedures, in conjunction with other structures, systems, and components, during waste package handling and transport.

The current DOE waste package design description appears to adequately provide shielding to prevent radiolysis-induced corrosion. Additional protection for workers is provided by other structures, systems, and components. The design of the waste package is still being developed, so DOE will provide additional design information in future documents. These documents will be reviewed as they become available.

Designing for Normal Operation and Categories 1 and 2 Event Sequences

DOE identified event sequences presently being considered in establishing the design criteria and specifications for important to safety structures, systems, and components (DOE, 2001b). A detailed discussion of the DOE identification and categorization of event sequences that pertain to the preclosure period of the proposed repository can be found in Subsections 2.1.4 and 2.1.5. The discussion presented in this section is limited to the postulated waste package drop event. As more information becomes available the scope of this discussion will be expanded to include other relevant important to safety structures, systems, and components event sequence and consequence analyses.

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[23]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

The waste package drop event has been characterized as an internal event sequence that is not expected to result in a radiological release because it is prevented by the design of the waste package (CRWMS M&O, 2000h). Analyses intended to support this characterization have been performed (CRWMS M&O, 2000v). The scope of these analyses was limited to a single waste package drop orientation. It is not clear that a single drop orientation scenario is sufficient to bound the potential for waste package failure, considering the number of different waste package handling operations and the present lack of design detail for the various cranes and other devices that will be used to transfer the waste package from the waste handling building to its emplacement within the drift. DOE stated during the preclosure technical exchange[24] that, as part of the normal design process, design basis dynamic events will be reevaluated as the designs for both the surface and subsurface facilities mature. It should be noted that DOE does not consider the waste package to be breached if the inner disposal container remains intact.

No specific requirements are provided in 10 CFR Part 63 that mandate waste package drop tests or any other empirical evaluations that will demonstrate the structural integrity of the waste package subjected to other design basis events, such as those required by 10 CFR Part 71. As a result, the means used to demonstrate the ability of the waste package to withstand the postulated event sequences is at the discretion of DOE. In the case of demonstrating the ability of the waste package to withstand handling drops without breaching, DOE has chosen to use numerical simulations based on the finite element method as the sole basis for its safety case. Although DOE has not precluded the use of actual waste package drop tests in the future to demonstrate the structural integrity of the waste package, there are no specific plans to do so at this time.

Because of the reliance on computer simulations to demonstrate the performance capabilities of the waste package, the assumptions, boundary conditions, material characterization, numerical formulations (along with their inherent limitations), level of mesh discretization, and failure criteria will have to be scrutinized more rigorously. As a result, DOE agreed[25] to (i) demonstrate that the mesh discretizations of the finite element models used to simulate the effects of waste package drop events are sufficient to provide reasonably convergent results that can be used to assess potential failure, (ii) justify the constitutive models used to represent the response of the waste package materials to impact loads (e.g., the inclusion or exclusion of temperature and strain rate effects), (iii) provide documentation of all boundary conditions used for the numerical models and the technical basis or rationale for them, and (iv) provide evidence that the criterion used to establish failure adequately bounds the uncertainties associated with effects not explicitly considered in the simulation. Specific uncertainties not presently considered in the waste package drop analyses are (i) residual stresses arising from the closure weld fabrication process, (ii) dimensional and material variability, (iii) ground motion effects caused by a seismic event (waste package drops are more likely to occur during seismic events), (iv) sliding and inertial effects of the spent nuclear fuel, and related matters.

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[24]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001).” Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[25]Ibid.

The waste package drop analyses DOE performed (CRWMS M&O, 2000v) does not indicate whether the structural integrity of the spent nuclear fuel was considered when establishing allowable drop heights. At the preclosure technical exchange,[26] DOE stated that in case of a drop, an assessment would be made as to whether the waste form must be repackaged, but the primary consideration when establishing drop heights is the integrity of the waste package. DOE also noted that the repackaging requirements have not yet been established, but they will be based on long-term performance needs.

Fuel Cladding Thermal Control

Temperature control for commercial spent nuclear fuel waste packages after emplacement within the repository will be provided using a combination of drift spacing, waste package spacing, ventilation during the preclosure period, waste package configuration, and thermal blending of the spent nuclear fuel. The maximum allowed thermal output of any waste package is 11.8 kW [40,263 BTU/hr] (CRWMS M&O, 2000f). With the exception of waste packages with 24 boiling water reactor fuel assemblies, the waste packages containing commercial spent nuclear fuel have aluminum thermal shunts added to conduct heat from the interior of the waste package to the waste package inner container. The axial and radial gaps between the inner and outer containers after differential thermal expansion will affect the steady-state waste package temperatures. Larger gaps will tend to cause higher interior and lower exterior (i.e., outer container) temperatures. Aluminum Alloys 6061 and 6063 were chosen instead of copper because of concerns that copper may react with chloride introduced by water entering the waste package and cause accelerated degradation of the zirconium alloy cladding. For the commercial spent nuclear fuel waste package configurations, the 21 pressurized water reactor fuel waste packages with absorber plates have the highest heat output with an average of 11.33 kW [38,650 BTU/hr] (CRWMS M&O, 2000f). Peak cladding temperatures are calculated to be less than 300 °C [572 °F], even with close waste package spacing (CRWMS M&O, 2000e). The heating, ventilation, and air conditioning system within the waste handling building will maintain fuel cladding temperatures within acceptable limits before packaging and emplacement.

The current DOE waste package design description appears to include components to provide thermal control so the fuel cladding temperature will be maintained within acceptable limits. The design of the waste package is still under development, so DOE will provide additional design information in future documents. These documents will be reviewed as they become available.

Backfill Design

Backfill is not used in the present conceptual design of the proposed repository. As a result, no assessment is required.

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[26]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001).” Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

<u>Sorption Barrier Design</u>

A sorption barrier is not used in the present conceptual design of the proposed repository. As a result, no assessment is required.

2.1.7.4 Status and Path Forward

Table 2.1.7-2 provides the status of the Design of Structures, Systems, and Components Important to Safety and Safety Controls. The table also enumerates the related DOE and NRC agreements pertaining to the Repository Design and Thermal-Mechanical Effects and Container Life and Source Term Key Technical Issues. The agreements listed in the table are associated with acceptance criteria discussed in Sections 2.1.7.3.3.2 and 2.1.7.3.3.3. Note that the status and the detailed agreements (or path forward) pertaining to all the key technical issue subissues are provided in Table 1.1-3 and Appendix A.

Table 2.1.7-2. Summary of Resolution Status for Design for Structures, Systems, and Components Important to Safety and Safety Controls Preclosure Topic

Preclosure Items	Status	Related Agreements*	Comments
Relationship between the Design Criteria and Design Basis and the Regulatory Requirements	Pending	†	Staff Review Incomplete
Geologic Repository Operations Area Design Methodologies	Pending	†	Staff Review Incomplete
Assumptions, Codes, and Standards for Surface Facilities Design	Pending	†	Staff Review Incomplete
Materials for Surface Facilities Design	Pending	†	Staff Review Incomplete
Load Combinations for Surface Facilities Design	Pending	†	Staff Review Incomplete
Surface Facilities Design Analyses and Documentation	Pending	†	Staff Review Incomplete
Assumptions, Codes, and Standards for Subsurface Facility Design	Pending	†	Staff Review Incomplete
Subsurface Operating Systems Design	Pending	†	Staff Review Incomplete
Material and Material Properties for Subsurface Facility Design	Pending	RDTME.3.01	Impact of corrosion on the effectiveness of ground-support system

Table 2.1.7-2. Summary of Resolution Status for Design for Structures, Systems, and Components Important to Safety and Safety Controls Preclosure Topic (continued)

Preclosure Items	Status	Related Agreements*	Comments
Load Combinations for Subsurface Facility Design	Pending	RDTME.2.01 RDTME.2.02 RDTME.3.02 RDTME.3.03	Seismic load characterization and critical combination of thermal and seismic loadings
Models and Rock Properties for Subsurface Facility Design	Pending	RDTME.3.04 RDTME.3.05 RDTME.3.07 RDTME.3.08 RDTME.3.10 RDTME.3.13	Rock properties and data sufficiency, rock strength, and fracture pattern analyses
Subsurface Ground-Support Systems Design	Pending	RDTME.3.06 RDTME.3.09	Drift invert stability and rock support system analyses
Subsurface Ventilation System Design	Pending	RDTME.3.14	Ventilation modeling and validation
Subsurface Power and Power Distribution Systems Design	Pending	†	Staff Review Incomplete
Maintenance Plan for Subsurface Facility	Pending	†	Staff Review Incomplete
Waste Package and Engineered Barrier Subsystem Design	Pending	PRE.07.01 through PRE.07.05	Criticality analysis, finite element modeling, weld filler material compatibility, nondestructive evaluation methods, and mechanical properties after welding

*Related DOE and NRC agreements are associated with one or more acceptance criteria.
†Not discussed at the first DOE and NRC Technical Exchange and Management Meeting on Preclosure Safety, July 24–26, 2001, Las Vegas, Nevada.

2.1.7.5 References

American Society of Mechanical Engineers. “Rules for Construction of Nuclear Power Plant Components, Division 1, Subsection NB, Class 1 Components.” Section III. *American Society for Mechanical Engineers Boiler and Pressure Vessel Code.* New York, New York: American Society of Mechanical Engineers. 1995a.

———. "Materials." Section II. *American Society for Mechanical Engineers Boiler and Pressure Vessel Code.* New York, New York: American Society of Mechanical Engineers. 1995b.

———. "Welding and Brazing Qualifications." Section IX. *American Society for Mechanical Engineers Boiler and Pressure Vessel Code.* New York, New York: American Society of Mechanical Engineers. 1995c.

———. "Nondestructive Examination." Section V. *American Society for Mechanical Engineers Boiler and Pressure Vessel Code*. New York, New York: American Society of Mechanical Engineers. 1995d.

———. "Rules for Inservice Inspection of Nuclear Power Plant Components." Section XI. *American Society for Mechanical Engineers Boiler and Pressure Vessel Code.* New York: American Society of Mechanical Engineers. 1995e.

———. "Rules for Construction of Nuclear Power Plant Components, Division 1, Subsection NB, Class 1 Components." Section III. *American Society for Mechanical Engineers Boiler and Pressure Vessel Code.* New York, New York: American Society of Mechanical Engineers. 2001.

American Society for Testing and Materials International. "Standard Test Methods of Detecting Susceptibility to Intergranular Attack in Wrought, Nickel-Rich, Chromium-Bearing Alloys." G28-85. *Annual Book of Standards*. West Conshohocken, Pennsylvania: American Society for Testing and Materials. 1999.

Barton, N., R. Lien, and J. Lunde. "Engineering Classification of Rock Masses for the Design of Tunnel Support." *Rock Mechanics.* Vol. 6. pp. 189–236. 1974.

Bieniawski, Z.T. "The Geomechanics Classification in Rock Engineering Applications." Proceedings of the 4th International Congress on Rock Mechanics. Montreaux, Switzerland. Vol. 2. Rotterdam, The Netherlands: A.A. Balkema Publishers. pp. 41–48. 1979.

Brekke, T.L., E.J. Cording, J. Daemen, R.D. Hart, J.A. Hudson, P.K. Kaiser, and S. Pelizza. "Panel Report on the Drift Stability Workshop, Las Vegas, Nevada, 9–11 December 1998." Las Vegas, Nevada: DOE, Yucca Mountain Site Characterization Project. 1999.

Carlos, B.A., S.J. Chipera, and D.L. Bish. "Distribution and Chemistry of Fracture-Lining Minerals at Yucca Mountain, Nevada." LA–12977–MS, UC–814. Los Alamos, New Mexico: Los Alamos National Laboratory. 1995.

Cieslak, M.J., T.J. Headley, and A.D. Romig. "The Welding Metallurgy of Hastelloy Alloys C-4, C-22, C-276." *Metallurgical Transactions.* Vol. 17A. pp. 2,035–2,047. 1986a.

Cieslak, M.J., G.A. Knorovsky, T.J. Headley, and A.D. Romig. "The Use of PHACOMP in Understanding the Solidification Microstructure of Nickel Base Alloy Weld Metal." *Metallurgical Transactions.* Vol. 17A. pp. 2,107–2,116. 1986b.

CRWMS M&O. "Confirmation of Empirical Design Methodologies." BABEE0000–01717–5705–00002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 1997a.

———. "Yucca Mountain Site Geotechnical Report." B00000000–01717–5705–00043. Revision 01. Las Vegas, Nevada: CRWMS M&O. 1997b.

———. "Total System Performance Assessment–Site Recommendation Methods and Assumptions." TDR–MGR–MD–000001. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 1999a.

———. "Monitored Geologic Repository Project Description Document." B00000000–01717–170500003. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 1999b.

———. "Subsurface Facility System Description Document." SDD–SFS–SE–000001. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000a.

———. "Waste Emplacement/Retrieval System Description Document." SDD–WES–SE–000001. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000b.

———. "Subsurface Ventilation System Description Document." SDD–SVS–SE–000001. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000c.

———. "Ground Control System Description Document." SDD–GCS–SE–000001. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000d.

———. "Design Analysis for UCF Waste Packages." ANL–UDC–MD–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000e.

———. "Update to the EIS Engineering File for the Waste Package in Support of the Final EIS." TDR–EBS–MD–000010. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000f.

———. "Design Analysis for the Ex-Container Components." ANL–XCS–ME–00001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000g.

———. "Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations." TDR–WIS–RL–000001. Revision 04 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000h.

———. "Emplacement Drift System Description Document." SDD–EDS–SE–000001. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000i.

———. "Longevity of Emplacement Drift Ground Support Materials." ANL–EBS–GE–000003. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000j.

———. “Ground Control for Emplacement Drifts for Site Recommendation. ANL–EBS–GE–000002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000k.

———. “Drift Scale Thermal Analysis.” CAL–WIS–TH–000002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000l.

———. “Site Recommendation Subsurface Layout.” ANL–SFS–MG–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000m.

———. “Fracture Geometry Analysis for the Stratigraphic Units of the Repository Horizon.” ANL–EBS–GE–000006. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000n.

———. “Drift Degradation Analysis.” ANL–EBS–MD–000027. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000o.

———. “Performance Confirmation Plan.” TDR–PCS–SE–000001. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000p.

———. “Structural Calculation of an Emplacement Pallet Statically Loaded by a Waste Package.” CAL–WER–ME–000003. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000q.

———. “Structural Calculations for the Lifting of a Loaded Emplacement Pallet.” CAL–WER–ME–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000r.

———. “Aging and Phase Stability of Waste Package Outer Barrier.” ANL–EBS–MD–000002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000s.

———. “Waste Package Operations Fabrication Process Report.” TDR–EBS–ND–000003. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000t.

———. “Disposal Container Handling System Description Document.” SDD–DCH–SE–000001. Revision 01 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000u.

———. “Corner Drop of 21-PWR Waste Packages.” CAL–UDC–ME–000008. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000v.

———. “Waste Package Operations Fabrication Process Report.” TDR–EBS–ND–000003. Revision 02. Las Vegas, Nevada: CRWMS M&O. 2001a.

———. “Waste Package Project FY-01 Closure Methods Report.” TDR–EBS–ND–000006. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2001b.

Daily, W., W. Lin, and T. Buscheck. “Hydrological Properties of Topopah Spring Tuff: Laboratory Measurements.” *Journal of Geophysical Research*. Vol. 92, No. B8. pp. 7,854–7,864. 1987.

DOE. "Topical Report: Preclosure Seismic Design Methodology for a Geologic Repository at Yucca Mountain." YMP/TR–003–NP. Revision 2. Las Vegas, Nevada: DOE, Yucca Mountain Site Characterization Project. 1997.

———. "*Q-List*." YMP/90–55Q. Revision 6. Las Vegas, Nevada: DOE, Yucca Mountain Site Characterization Project. 2000.

———. "Yucca Mountain Science and Engineering Report—Technical Information Supporting Site Recommendation Consideration." Las Vegas, Nevada: CRWMS M&O. 2001a.

———. "Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation." TDR–MGR–SE–000009. Revision 00 ICN 03. Las Vegas, Nevada: DOE. 2001b.

Heubner, U.L., E. Altpeter, M.B. Rockel, and E. Wallis. "Electrochemical Behavior and Its Relation to Composition and Sensitization of Ni-Cr-Mo Alloys in ASTM G28 Solution." *Corrosion*. Vol. 45, No. 3. pp. 249–259. 1989.

Hoek, E. and E.T. Brown. *Underground Excavations in Rock*. London, England: The Institution of Mining and Metallurgy. 1980.

———. "Practical Estimates of Rock Mass Strength." *International Journal of Rock Mechanics and Mining Sciences and Geomechanics Abstracts*. Vol. 34, No. 8. pp. 1,165–1,186. 1997.

Hsiung S.M., G.H. Shi, and A.H. Chowdhury. "Assessment of Seismically Induced Rockfall in the Emplacement Drifts of the Proposed Repository at Yucca Mountain." San Antonio, Texas: CNWRA. 2001.

Kenney, T.C. "The Influence of Mineralogical Composition on the Residual Strength of Natural Soils." Proceedings of the Oslo Geotechnical Conference on the Shear Strength Properties of Natural Soils and Rocks, I. Oslo, Norway: Norwegian Geotechnical Institute. pp. 123–129. 1967.

Lajtai, E.Z. and R.H. Schmidtke. "Delayed Failure in Rock Loaded in Uniaxial Compression." *Rock Mechanics and Rock Engineering*. Vol. 19. pp. 11–25. 1986.

Levy, S.S., D.T. Vaniman, D.L. Bish, G. WoldeGabriel, S.J. Chipera, J.W. Carey, P. Dixon, S.J. Goldstein, and M.T. Murrell. "Summary and Synthesis Report on Mineralogy and Petrology Studies for the Yucca Mountain Site Characterization Project, Volume II: History of Mineralogic and Geochemical Alteration of Yucca Mountain." Los Alamos, New Mexico: Los Alamos National Laboratory. 1996.

Lin, W. and W. Daily. "Transport Properties of Topopah Spring Tuff." UCRL–53602. Livermore, California: Lawrence Livermore National Laboratory. 1984.

Martin, C.D. and N.A. Chandler. “The Progressive Fracture of Lac du Bonnet Granite.” *International Journal of Rock Mechanics and Mining Sciences and Geomechanics Abstracts.* Vol. 31, No. 6. pp. 643–659. 1994.

Matthews, S.J. “Thermal Stability of Solid Solution Strengthened High Performance Alloys.” Superalloys: Metallurgy and Manufacture, Proceedings of the Third International Symposium. Baton Rouge, Louisiana: Claitor’s Publishing Division. pp. 215–226. 1976.

Matyskiela, W. “Silica Redistribution and Hydrologic Changes in Heated Fractured Tuff.” *Geology*. Vol. 25, No. 12. pp. 1,115–1,118. 1997.

McCright, R.D. “Corrosion Data and Modeling Update for Viability Assessment.” Engineered Materials Characterization Report. Vol. 3. Revision 1.1. Livermore, California: Lawrence Livermore National Laboratory. 1998.

Mitchell, J.K. *Fundamentals of Soil Behavior*. New York, New York: John Wiley and Sons. 1976.

Murphy, W.M. “Geochemical Models for Gas-Water-Rock Interactions in a Proposed Nuclear Waste Repository at Yucca Mountain, Nevada.” Proceedings of the Topical Meeting on Site Characterization and Model Validation, Focus ’93, Las Vegas, Nevada, September 26–29, 1993. La Grange Park, Illinois: American Nuclear Society. pp. 115–121. 1993.

NRC. Regulatory Guide 3.71, “Nuclear Criticality Safety Standards for Fuels and Material Facilities.” Washington, DC: NRC, Office of Standards Development. 1998.

———. “Issue Resolution Status Report, Key Technical Issue: Repository Design and Thermal-Mechanical Effects.” Revision 3. Washington, DC: NRC. 2000a.

———. “Issue Resolution Status Report, Key Technical Issue: Total System Performance Assessment and Integration.” Revision 3. Washington, DC: NRC. 2000b.

———. “Issue Resolution Status Report, Key Technical Issue: Container Life and Source Term.” Revision 3. Washington, DC: NRC. 2001.

———. NUREG–1804, “Yucca Mountain Review Plan—Draft Report for Comment.” Revision 2. Washington, DC: NRC. March 2002.

Ofoegbu, G.I. “Variations of Drift Stability at the Proposed Yucca Mountain Repository.” Rock Mechanics for Industry: Proceedings of the 37th U.S. Rock Mechanics Symposium. Rotterdam, The Netherlands: A.A. Balkema Publishers. 1999.

———. “Thermal-Mechanical Effects on Long-Term Properties at the Proposed Yucca Mountain Nuclear Waste Repository.” CNWRA 2000-03. San Antonio, Texas: CNWRA. 2000.

———. “Hydrological Implications of Thermally Induced Geomechanical Responses at Yucca Mountain, Nevada.” Rock Mechanics in the National Interest: Proceedings of the 38th U.S. Rock Mechanics Symposium, Washington, DC, 7–10 July 2001. Lisse, The Netherlands: A.A. Balkema Publishers. 2001.

Ofoegbu, G.I. and J.H. Curran. “Rotation of Principal Stresses Near a Heated Fracture in a Bituminous Sand.” *Canadian Geotechnical Journal*. Vol. 24. pp. 357–365. 1987.

Ofoegbu, G.I., S. Painter, R. Chen, R.W. Fedors, and D.A. Ferrill. “Geomechanical and Thermal Effects on Moisture Flow at the Proposed Yucca Mountain Nuclear Waste Repository.” *Nuclear Technology*. Vol. 134. pp. 241–262. 2001.

Palmstrom, A. “Characterizing Rock Masses by the RMI for Use in Practical Rock Engineering. Part 2: Some Practical Applications of the Rock Mass Index (RMI).” *Tunnelling and Underground Space Technology*. Vol. 11, No. 3. pp. 287–303. 1996.

Raghavan, M., R.R. Mueller, G.A. Vaughn, and S. Floreen. “Determination of Isothermal Sections of Nickel Rich Portion of Ni-Cr-Mo System by Analytical Electron Microscopy.” *Metallurgical Transactions*. Vol. A. pp. 783–792. 1984.

Rebak, R.B., T.S.E. Summers, and R.M. Carranza. “Mechanical Properties, Microstructure, and Corrosion Performance of C-22 Alloy Aged at 260 °C to 800 °C.” Scientific Basis for Nuclear Waste Management XXIII. R.W. Smith and D.W. Shoesmith, eds. Symposium Proceedings 608. Warrendale, Pennsylvania: Materials Research Society. 2000.

Serafim, J.L. and J.P. Pereira. “Considerations on the Geomechanical Classification of Bieniawski.” Proceedings, International Symposium on Engineering Geology and Underground Construction. Vol. II. Lisbon, Portugal: Laboratorio Nactional De Engenharia Civil. pp. II.33–II.42. 1983.

Stephenson, N. “Versatility of Highly Alloyed Ni-Cr-Mo Welding Consumables—Part I.” *Welding and Metal Fabrication*. August/September. Vol. 58, No. 7. pp. 376–384. 1990.

Stock, J.M., J.H. Healy, S.H. Hickman, and M.D. Zoback. “Hydraulic Fracturing Stress Measurements at Yucca Mountain, Nevada.” *Journal of Geophysical Research*. Vol. 90, No. B10. pp. 8,691–8,706. 1985.

Szklarska-Smialowska, Z. *Pitting Corrosion of Metals.* Houston, Texas: National Association of Corrosion Engineers. 1986.

Tawancy, H.M. “Precipitation Characteristics of µ-Phase in Wrought Nickel-Base Alloys and Its Effect on Their Properties.” *Journal of Materials Science.* Vol. 31. pp. 3,929–3,936. 1996.

Thoma, S.G., D.P. Gallegos, and D.M. Smith. “Impact of Fracture Coatings on Fracture/Matrix Flow Interactions in Unsaturated, Porous Media.” *Water Resources Research.* Vol. 28, No. 5. pp. 1,357–1,367. 1992.

2.1.8 Meeting the 10 CFR Part 20 As Low As Is Reasonably Achievable Requirements for Normal Operations and Category 1 Event Sequences

Text in this section will be provided at a later date.

2.2 Plans for Retrieval and Alternate Storage of Radioactive Wastes

Text in this section will be provided at a later date.

2.3 Plans for Permanent Closure and Decontamination, or Decontamination, and Dismantlement of Surface Facilities

Text in this section will be provided at a later date.

2.4 Status of Preclosure Issue Resolution and Path Forward

Based on 10 CFR Part 63 and its review of the DOE preliminary preclosure safety assessment report (CRWMS M&O, 2001), the repository safety strategy (CRWMS M&O, 2000), and other support documents, NRC staff preliminarily identified 10 preclosure topics that DOE should address in any future license application regarding the potential high-level waste repository at Yucca Mountain.

(1) Site Description As It Pertains to Preclosure Safety Analysis

(2) Description of Structures, Systems, Components, Equipment, and Operational Process Activities

(3) Identification of Hazards and Initiating Events

(4) Identification of Event Sequences

(5) Consequence Analyses

(6) Identification of Structures, Systems, and Components Important to Safety; Safety Controls; and Measures to Ensure Availability of the Safety Systems

(7) Design of Structures, Systems, and Components Important to Safety and Safety Controls

(8) Meeting the 10 CFR Part 20 As Low As Is Reasonably Achievable Requirements for Normal Operations and Category 1 Event Sequences

(9) Plans for Retrieval and Alternate Storage of Radioactive Wastes

(10) Plans for Permanent Closure and Decontamination, or Decontamination and Dismantlement of Surface Facilities

Resolution of concerns related to these preclosure topics (8), (9), and (10) has not been initiated. Therefore, no progress toward these three areas is documented in this issue resolution status report. Identification and resolution of concerns in the remaining subject areas are at various stages of progress.

2.4.1 Progress on Preclosure Topics

Identification of technical concerns associated with preclosure topics (1) through (7) is at various stages of development. Subtopics for the various technical areas identified for these seven preclosure topics, as of the cutoff date for this issue resolution report, are discussed in this subsection (Table 2.4-1). The list is not complete at this time, and technical concerns will continue to be identified and clarified as the review of DOE documents proceeds. It should also be noted that not all the preclosure technical concerns identified were addressed in the

July 2001 Technical Exchange Meeting on Preclosure Safety.[1] Additional information about the status of seismic design and thermal-mechanical effects on underground facility design related to preclosure topic (7) is discussed in Section 2.4.2.

Detailed discussions and agreements reached regarding the technical concerns are provided in appropriate sections of this issue resolution report. Table 2.4-1 provides the status of preclosure technical concerns. The table also enumerates the related DOE and NRC agreements pertaining to the preclosure technical areas. Note that the status of all key technical issues are provided in Table 1.1-3. In addition, all agreements pertaining to the key technical issues and preclosure subtopics are provided in Appendix A.

2.4.2 Progress on Preclosure Concerns Addressed in the Repository Design and Thermal-Mechanical Effects Key Technical Issue

In the Repository Design and Thermal-Mechanical Effects Key Technical Issue, three subissues are relevant to preclosure topic (7): Subissue 1, Implementation of an Effective Design Control Process Within the Overall Quality Assurance Program; Subissue 2, Design of the Geological Repository Operations Area for the Effects of Seismic Events and Direct Fault Disruption; and Subissue 3, Thermal-Mechanical Effects on Underground Facility Design and Performance.

Table 2.4-1. Related Technical Concerns and Agreements

Preclosure Topics and Key Technical Issue	Concerns or Subissues	Status	Related Agreements
Site Description As It Pertains to Preclosure Safety Analysis	Geotechnical Investigation for Surface Facility	Not Addressed	None
	Design Basis Ash Fall	Not Addressed	None
Description of Structures, Systems, Components, Equipment, and Operational Process Activities	High-Level Waste Characterization	Not Addressed	None
Identification of Hazards and Initiating Events	Aircraft Hazards	Addressed	PRE.03.01
	Tornado Missile Hazards	Addressed	PRE.03.02
	Nearby Military Facilities Hazards	Not Addressed	None

[1]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001).” Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

Table 2.4-1. Related Technical Concerns and Agreements (continued)			
Preclosure Topics and Key Technical Issue	**Concerns or Subissues**	**Status**	**Related Agreements**
Identification of Hazards and Initiating Events	Operational Hazards Including Human Reliability	Not Addressed	None
	Earthquake as an Initiating Event	Addressed	RDTME.2.01 RDTME.2.02
	Fire Hazards	Not Addressed	None
Identification of Event Sequences	Events Screened Out by Design	Addressed	Agreement Summary*
	Justification of Probability Estimates	Addressed	Agreement Summary*
Consequence Analyses	Dose Calculation Methodology for Category 1 Event Sequences	Addressed	None†
	Dose Calculation Methodology for Category 2 Event Sequences	Not Addressed‡	None
Identification of Structures, Systems, and Components Important to Safety; Safety Controls; and Measures to Ensure Availability of the Safety Systems	Q-List Methodology	Addressed	PRE.06.01 PRE.06.02
	Quality Level Categorization	Addressed	PRE.06.01 PRE.06.02
Design of Structures, Systems, and Components Important to Safety and Safety Controls	Level of Design Details	Addressed	None§
	Engineered Barrier Subsystem and Fabrication	Addressed	PRE.07.02 through PRE.07.05
	Burnup Credit and Criticality	Addressed	PRE.07.01
	Soil-Structure Interaction	Not Addressed	None
	Ventilation Design	Not Addressed	None
	Fire Protection Design	Not Addressed	None

Table 2.4-1. Related Technical Concerns and Agreements (continued)			
Preclosure Topics and Key Technical Issue	**Concerns or Subissues**	**Status**	**Related Agreements**
Meeting the 10 CFR Part 20 As Low As Is Reasonably Achievable Requirements for Normal Operations and Category 1 Event Sequences	Not Yet Identified	Review Not Initiated	None
Plans for Retrieval and Alternate Storage of Radioactive Wastes	Not Yet Identified	Review Not Initiated	None
Plans for Permanent Closure and Decontamination, or Decontamination and Dismantlement of Surface Facilities	Not Yet Identified	Review Not Initiated	None
Repository Design and Thermal-Mechanical Effects	Subissue 1—Implementation of an Effective Design Control Process Within the Overall Quality Assurance Program	Closed	None
	Subissue 2—Design of the Geological Repository Operations Area for the Effects of Seismic Events and Direct Fault Disruption	Closed-Pending	RDTME.2.01 RDTME.2.02
	Subissue 3—Thermal-Mechanical Effects on Underground Facility Design and Performance	Closed-Pending	RDTME.3.01 through RDTME.3.14

* Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.
† Common understanding with DOE was reached.
‡ No significant uncertainties because well-established methods are available.
§ A draft position paper was provided to DOE.

Historically, DOE implementation of a design control process for design, construction, and operation of the geologic repository operations area has been one of the NRC major concerns. The staff conducted a series of interactions and reviews and an in-field verification to evaluate the effectiveness of the DOE design control process. Through these interactions, deficiencies covering a wide spectrum of the design control process, including data traceability, management, qualification, and software control, were identified [for a detailed discussion, refer to NRC (2000)]. In responding to the NRC concerns, DOE developed and implemented new administrative procedures to replace the existing quality assurance procedures. The new administrative procedures extend to the contractors. The staff believe these new administrative procedures simplify the document hierarchy that controls the design and analysis activities. As a result, transparency and traceability of the flowdown from the regulatory requirements to design bases and criteria are improved. The staff consider this simplified design control process acceptable, and Key Technical Issue Subissue 1, Implementation of an Effective Design Control Process Within the Overall Quality Assurance Program, is closed with respect to issue resolution. The implementation of the design control process, however, will continue to be monitored through observation of DOE audits or NRC independent audits and inspections of DOE activities.

DOE proposed three topical reports to address Key Technical Issue Subissue 2, Design of the Geological Repository Operations Area for the Effects of Seismic Events and Direct Fault Disruption. NRC staff reviewed and accepted the first and second topical reports (DOE, 1994, 1996). NRC will review the third topical report, Design of the Geological Repository Operations Area for the Effects of Seismic Events and Direct Fault Disruption, once it is submitted.

Key Technical Issue Subissue 3, Thermal-Mechanical Effects on Underground Facility Design and Performance, was discussed during the technical exchange meeting with DOE about the Repository Design and Thermal-Mechanical Effects Key Technical Issue.[2] Agreements on various aspects of the subissues were reached during the meeting. Consequently, Subissues 2 and 3 are currently closed-pending. Detailed discussions about concerns are provided in Section 2.1.7 of this issue resolution report. Table 2.4-1 provides the status of Subissues 2 and 3 and related DOE and NRC agreements pertaining to the Repository Design and Thermal-Mechanical Effects Key Technical Issue. The status and detailed agreements pertaining to all key technical issues are provided in Table 1.1-3 and Appendix A.

2.4.3 Path Forward

The path forward for addressing the preclosure-related concerns includes four parts:

(1) Conducting Appendix 7 meetings with DOE to monitor the progress of addressing the agreements reached during the previous technical exchange meetings

(2) Continuing the review of DOE preclosure-related documents when they become available and the identification of technical concerns, if any

(3) Conducting a technical exchange meeting to discuss the remaining preclosure concerns listed in Section 2.4.1 and new concerns identified so far through reviewing DOE preclosure-related documents

(4) Conducting limited independent preclosure safety analyses to identify vulnerabilities in the DOE design and related safety case

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[2]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Repository Design and Thermal-Mechanical Effects (February 6–8, 2001)." Letter (February 28) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

2.4.4 References

CRWMS M&O. “Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations.” TDR–WIS–RL–000001. Revision 04 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000.

———. “Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation.” TDR–MGR–SE–000009. Revision 00 ICN 03. Las Vegas, Nevada: CRWMS M&O. 2001.

DOE. “Methodology to Assess Fault Displacement and Vibratory Groundmotion Hazards at Yucca Mountain.” YMP/TR–002–NP. Revision 0. Washington, DC: DOE. 1994.

———. “Seismic Design Methodology for a Geologic Repository at Yucca Mountain.” YMP/TR–003–NP. Revision 1. Washington, DC: DOE. 1996.

NRC. “Issue Resolution Status Report—Key Technical Issues: Repository Design and Thermal-Mechanical Effects.” Revision 3. Washington, DC: NRC, Division of Waste Management. 2000.

3 REPOSITORY SAFETY AFTER PERMANENT CLOSURE

3.1 System Description and Demonstration of Multiple Barriers

3.1.1 Description of Issue

Postclosure performance objectives specified in 10 CFR Part 63 require a system of multiple barriers (at least one engineered and one natural). As defined in the regulations, barriers are materials or structures that prevent or substantially delay movement of water or radionuclides. Thus, a key element of the safety case is the identification and description of the capabilities of the repository barriers. Examples of natural barriers at Yucca Mountain include the unsaturated and saturated volcanic and alluvial rock units that control movement of radionuclides by processes such as infiltration, matrix diffusion, and sorption. Engineered barriers DOE has considered in design options include a titanium drip shield, a double-walled container for waste packages, fuel cladding, and invert materials. Each barrier provides additional assurance that the postclosure performance objectives can be met. The description of each barrier capability provides an overall understanding of the DOE safety case and how the diversity of the barriers enhances the resiliency of the repository system.

As provided in 10 CFR Part 63, DOE is required to identify the barriers in the safety case, describe the capabilities of each of the barriers, and provide the technical basis for the capability of the barriers (the technical basis is to be consistent with the technical basis used to support the total system performance assessment). In general, staff will review the potential Total System Performance Assessment–License Application to ensure that DOE identifies all barriers in its safety case; describes the capability of the barriers consistent with the parameter, models, and assumptions in the total system performance assessment; and provides a technical basis consistent with that used for the total system performance assessment.

The following summaries are excerpted from 10 CFR Part 63.

10 CFR 63.113—Performance objectives for the geologic repository after permanent closure.

- The geologic repository must include multiple barriers, consisting of both natural barriers and an engineered barrier subsystem.

- The engineered barrier subsystem must be designed so that, working in combination with natural barriers, radiological exposures to the reasonably maximally exposed individual are within the limits specified in 10 CFR 63.311 of Subpart L. Compliance with this paragraph must be demonstrated through a total system performance assessment (that meets the requirements specified in 10 CFR 63.114 of this subpart, and 10 CFR 63.303, 63.305, 63.312, and 63.342 of Subpart L).

- The engineered barrier subsystem must be designed so that, working in combination with natural barriers, radionuclides released into the accessible environment are within the limits specified in 10 CFR 63.331 of Subpart L. Compliance with this paragraph must be demonstrated through a total system performance assessment (that meets the requirements specified in 10 CFR 63.114 of this subpart and 10 CFR 63.303, 63.332, and 63.342 of Subpart L).

10 CFR 63.115—Requirements for multiple barriers. Demonstration of compliance with 10 CFR 63.113 must

- Identify those design features of the engineered barrier subsystem, and natural features of the geologic setting, considered barriers important to waste isolation.

- Describe the capability of barriers identified as important to waste isolation to isolate waste, taking into account uncertainties in characterizing and modeling the behavior of the barriers.

- Provide technical basis for description of the capability of barriers identified as important to waste isolation to isolate waste. The technical basis for each barrier's capability shall be based on and consistent with the technical basis for the total system performance assessments used to demonstrate compliance with 10 CFR 63.113(b) and (c).

Consistent with 10 CFR Part 63, the Multiple Barriers Subissue in NRC (2002) focuses on the demonstration of multiple barriers and includes (i) identification of design features of the engineered barrier subsystem and natural features of the geologic setting considered barriers important to waste isolation, (ii) descriptions of the capability of barriers to isolate waste, and (iii) technical basis for each barrier capability. In addition, the review plan (NRC, 2002) addresses the staff expectation of the contents of the DOE total system performance assessment and supporting documents. Specifically, it focuses on those aspects of the total system performance assessment that will allow for an independent review of the results.

NRC staff will review the potential Total System Performance Assessment–License Application to ensure that multiple barrier considerations satisfy the requirements of 10 CFR 63.113(a). Staff will ensure that an engineered barrier subsystem has been designed that, working in combination with natural barriers, satisfies the requirement for a system of multiple barriers and complies with postclosure performance standards.

NRC staff will review the potential Total System Performance Assessment–License Application to ensure that multiple barrier considerations satisfy the requirements at 10 CFR 63.115(a)–(c). Staff will ensure that those design features of the engineered barrier subsystem and natural features of the geologic setting considered barriers important to waste isolation have been identified. A description has been provided of the capabilities of barriers identified as important to waste isolation, taking into account uncertainties in characterizing and modeling the barriers. The technical basis provided for this description is based on and consistent with the technical basis for the total system performance assessment.

This section provides a review of the multiple barrier analysis presented in the DOE total system performance assessment, a discussion of the NRC review, and agreements reached with the DOE. NRC review was limited to the methodology portion of multiple barriers. Compliance with the standards in 10 CFR Part 63 for individual and groundwater protection and human intrusion is not considered in prelicensing issue resolution. The comments describe the staff expectation of the contents of the DOE total system performance assessment, and the supporting documents define those aspects that will allow an independent review of the total system performance assessment results.

3.1.2 Relationship to Key Technical Issue Subissues

All key technical issue subissues contribute to (i) identification of design features of the engineered barrier subsystem and natural features of the geologic setting, (ii) descriptions of the capability of barriers, and (iii) technical basis for each barrier capability.

3.1.3 Importance to Postclosure Performance

If the repository system is made up of multiple barriers, it will be more tolerant of unanticipated failures and external challenges. Understanding the capability of the system component barriers provides an understanding of the repository system, which can increase confidence that the postclosure performance objectives will be met.

The description of barrier capability provides information that helps interpret the total system performance assessment results and provides information independent from the condition of the other barriers, so that insights can be gained into total system performance assessment results. Such information illustrates the resilience or lack of resilience of the repository to unanticipated failures or external challenges.

The evaluation of a first-of-a-kind repository for an extended time period (i.e., 10,000 years) results in uncertainty in characterizing the natural system being included in the total system performance assessment. Besides, those materials used in the engineered barrier subsystem that are relatively new (i.e., without a long history of use), have uncertainty in their life prediction. Consideration of multiple barriers as a part of total system performance assessment compensates for such residual uncertainties in estimating performance and increases confidence that postclosure performance objectives will be met.

The description of each barrier capability provides the reviewer flexibility to consider the nature and extent of conservatism in the evaluations used for compliance demonstration and to decide whether there is a need to require DOE to reduce uncertainties in the assessment (e.g., collecting more site data) or to include further mitigative measures.

3.1.4 Technical Basis

NRC has developed a review plan (NRC, 2002) consistent with acceptance criteria and review methods found in previous issue resolution status reports. This section briefly describes the DOE approach and the NRC staff review of that approach. Finally, this section presents agreements DOE and NRC reached to address the staff concerns.

Overall, the current information, along with agreements reached between DOE and NRC, is sufficient to conclude that the necessary information will be available on the identification of barriers, description of barrier capability, and technical basis for barrier capability either before or at the time of a potential license application.

The NRC comments on the DOE multiple barrier analysis and the resulting agreements that led to the closed-pending status for this subissue are based on the information provided in

CRWMS M&O (2000a,b). A presentation titled, Total System Performance Assessment and Integration Key Technical Issue Subissue 1—Multiple Barriers, made at the technical exchange held in Las Vegas, Nevada, during August 6–10, 2001,[1] provided additional understanding of the DOE multiple barriers approach and future plan to support the DOE total system performance assessment. The staff also used their experience from the past independent research, information in open literature, review of previous DOE total system performance assessments, information learned during meetings with DOE, the approach used in the NRC TPA Version 4.0 code (Mohanty, et al., 2002), acceptance criteria and review methods in NRC (2002), and technical bases for these acceptance criteria contained in the Revision 3 Issue Resolution Status Reports of other key technical issues. In addition, insight gained from sensitivity studies using the NRC TPA Version 3.2 code (Mohanty, et al., 1999) has been incorporated to the extent feasible.

The DOE Approach

DOE documented its approach to identifying natural and engineered barriers in CRWMS M&O (2000a,b). DOE identified four natural barriers and five engineered barriers. Natural barriers consisted of (i) surficial soils and topography, (ii) unsaturated zone rocks above the repository, (iii) unsaturated zone rocks below the repository horizon, and (iv) tuff and alluvial aquifers. Engineered barriers consisted of (i) the titanium drip shield, (ii) the C–22 waste canister, (iii) commercial spent nuclear fuel cladding, (iv) the waste form (e.g., high-level waste glass), and (v) a drift invert (e.g., crushed tuff).

In CRWMS M&O (2000a,b) and the DOE presentation,[2] DOE stated that barrier importance analysis is used in conjunction with sensitivity analysis to demonstrate barrier capability. Barrier importance analysis encompasses[3] (i) evaluation of significance of parameter and model uncertainty, (ii) evaluation of robustness of system performance using low probability scenarios within the framework of the total system performance assessment, and (iii) quantification of the capability of the barrier to isolate waste. Two types of analyses have been performed: degraded barrier importance analysis and neutralized barrier importance analysis. The degraded barrier importance analysis has been performed by fixing several parameters associated with a barrier at the 95th percentile (or 5th percentile if that leads to maximizing the dose rate) values in the total system performance assessment model and rerunning the probabilistic analyses. For the neutralized barrier importance analysis, the function of a barrier is eliminated by setting selected parameters in a way that correspond to omission

[1]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[2]DOE. "Total System Performance Assessment and Integration." Presentation to DOE/NRC Technical Eaxchange on the *Total System Performance Assessment and Integration Key Technical Issue, August 6–9, 2001, Las Vegas, Nevada*. Las Vegas, Nevada: DOE, Yucca Mountain Site Characterization Office. 2001.

[3]Andrews, R.W. "Sensitivity and Barrier Importance Analyses for TSPA–SR." *Presentation to DOE/NRC Technical Exchange on Total System Performance Assessment (TSPA) for Yucca Mountain, June 6–7, 2000, San Antonio, Texas.* Washington, DC: DOE, Office of Civilian Radioactive Waste Management. 2000.

(i.e., neutralization) of a process model factor or equivalently (in most cases), a barrier. DOE points out that the neutralization of a barrier (compared to the degradation of a barrier, which is within the total system performance assessment parameter range) permits gaining insights into total system performance assessment and provides insights into barrier redundancy. In the degraded barrier importance analysis, DOE assumes that various natural and engineered barriers are degraded either individually or in combination. DOE recognizes that because the degraded barrier importance analysis necessarily stays within the basecase uncertainty ranges of individual analyses, it cannot elevate in importance any barrier having a restricted range of uncertainty.

DOE examined the relative contribution of each barrier by comparing the nominal performance results (i.e., dose curves) with the degraded performance results for radionuclides within and beyond the compliance period. The contribution of individual barriers has been compared to the overall performance objective.

The NRC review of the two DOE documents describing the demonstration of multiple barriers, in CRWMS M&O (2000a,b), resulted in several concerns, primarily in the areas of description of barrier capability and technical basis for barrier capability. The staff believe that barrier capability needs to be described consistent with the definitions in 10 CFR Part 63 (i.e., prevents or substantially reduces movement of water or radionuclides). The concerns that led to reaching an agreement with DOE are listed next. The concerns that did not require agreements because the DOE clarifications addressed the issue can be found in the handouts provided at the DOE and NRC Technical Exchange on Total Systerm Performance Assessment.[4]

- DOE states the capabilities of barriers include (i) limiting contact of water on waste packages by reducing infiltration, (ii) prolonging waste package lifetimes, (iii) limiting radionuclide mobility and release, and (iv) slowing transport away from the repository. The NRC staff found that DOE presented the capability of barriers primarily in terms of dose. For example, CRWMS M&O (2000a, pp. 2–5) describes barrier capability, but no diagrams are presented to support the discussion. Although CRWMS M&O (2000a) asserts the barriers limit water and radionuclide movements, the results from barrier neutralization importance analyses and degraded barrier importance analyses (see figures in Chapter 3 of CRWMS M&O, 2000a) are based only on dose, and not on barrier capability, to prevent or delay movement of water or radionuclides. To understand the barrier capability, the NRC staff should be able to understand how the total system performance assessment results can be explained through barrier capability (e.g., retardation of radionuclides in the saturated zone, waste package lifetime, and matrix diffusion in the unsaturated zone). Understanding the way natural

[4]DOE. "Total System Performance Assessment and Integration." Presentation to DOE/NRC Technical Eaxchange on the *Total System Performance Assessment and Integration Key Technical Issue, August 6–9, 2001, Las Vegas, Nevada*. Las Vegas, Nevada: DOE, Yucca Mountain Site Characterization Office. 2001.

and engineered barriers isolate waste or delay radionuclide release will increase confidence in the total system performance assessment objectives specified at 10 CFR 63.11(b).

- The methods used to differentiate the contributions of barriers that perform similar functions need to be explained. Barriers that perform similar functions could include components of natural and engineered systems (e.g., the combination of the natural system above the repository and the drip shield) along important boundaries. The discussion of barrier capabilities needs to differentiate between the independent and the interdependent contributions of the individual barriers.

- The uncertainty associated with particular barriers has not been described. The description needs to include model uncertainty (such as the performance of the barrier, assuming alternative conceptual models) and uncertainty in the attributes of the barrier (e.g., parameter uncertainty). The performance needs to be discussed in light of barrier capability to prevent or delay movement of water or radionuclides and, consequently, to limit the expected annual dose.

- The DOE analyses do not describe the interdependence of barriers and also the treatment of combinations of barriers appears to be inconsistent. For example, the combination of barriers treated in CRWMS M&O (2000a) for the degraded barrier importance analyses is different from that used in the barrier neutralization importance analyses. Similarly, the combination of barriers presented in CRWMS M&O (2000b) is different from the combinations presented in CRWMS M&O (2000a) for degraded barrier importance analyses and barrier neutralization analyses. It is difficult to understand the results from the degraded barrier importance analyses and the barrier neutralization importance analyses for identifying barrier importance, without a discussion of the independent and interdependent contributions of the barriers.

  Example 1: The presence of the drip shield in the degraded waste package analyses (CRWMS M&O, 2000a) could mask the effect of the waste package on radionuclide transport during the early period or at least until the drip shield fails. Although such analyses (i.e., in the presence of the drip shield) shows the protection afforded by the drip shield even after the waste package fails, the actual protection provided by each individual barrier in 10,000 years is not clearly identified.

  Example 2: It is not clear why performance improved for the degraded radionuclide concentration limits case, which represents nonmechanistic juvenile failure scenario-sensitivity to radionuclide concentration limits, between 2,000 and 8,000 years (CRWMS M&O, 2000a, Figure 3-20, p. 3-18).

- The description of the capability for individual barriers to prevent or substantially delay movement of water or radionuclide materials needs to include a discussion of the changes in barrier capability during time (throughout the 10,000-year compliance period). The discussion should include the extent to which the conceptual models of the barriers consider cumulative degradation processes during time, processes that may significantly affect the performance of the barrier, and temporal changes within the repository system. As examples, time-dependent environmental or physical-chemical variabilities of the system (e.g., pressure, temperature, or spatial changes before, during, and after the thermal pulse); dynamic conditions (e.g., boiling zone/refluxation; calcite-opal mobilization and precipitation in fractures, lithophysae, and matrix pores; and drift collapse induced by thermal-mechanical stresses) may need to be discussed to appropriately describe the performance of particular barriers.

- The description of barrier capabilities needs to include a discussion of the effects of spatial variability on the ability of the barrier to prevent or substantially delay movement of water or radionuclide materials, including a discussion of the spatial resolution in the models and data used to evaluate the performance of the barriers. For example, assume 50 percent of the Calico Hills nonwelded vitric unit is strongly sorbing and 50 percent is not. As another example, in the what-if analysis of the nonmechanistic juvenile failure scenario in CRWMS M&O (2000a, pp. 3–15), one waste package was artificially set to fail after 100 years. The consequences associated with the failed waste package are influenced by the location of the failed waste package (e.g., the characteristics of radionuclide release, water flow, and radionuclide transport in the vicinity of the failed waste package, where these characteristics may be affected by spatial heterogeneity and its representation in the model used in the analysis).

NRC presented the previously mentioned concerns to DOE, and general agreements were reached at the DOE and NRC Total System Performance Assessment and Integration Issue Resolution Meeting, August 6–10, 2001.[5] DOE agreed to provide (i) enhanced descriptive treatment for presenting barrier capabilities in its final approach for demonstrating multiple barriers and (ii) a discussion of the capabilities of individual barriers, in light of existing parameter uncertainty (e.g., in barrier and system characteristics) and model uncertainty.

DOE also agreed to provide a discussion of the following when documenting barrier capabilities and the corresponding technical bases: (i) parameter uncertainty, (ii) model uncertainty (i.e., the effect of viable alternative conceptual models), (iii) spatial and temporal variabilities in the performance of the barriers, (iv) independent and interdependent capabilities of the barriers (e.g., including a differentiation of the capabilities of barriers performing similar functions), and (v) barrier effectiveness with regard to individual radionuclides. DOE will analyze and document barrier capabilities, in light of existing data and analyses of the performance of the repository system.

3.1.5 Status and Path Forward

The status of the System Description and Demonstration of Multiple Barriers Subissue of the Total System Performance Assessment and Integration Key Technical Issue is provided in Table 3.1-1. This subissue is considered closed-pending by the NRC staff as documented following the DOE and NRC Technical Exchange on Total System Performance Assessment

[5]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

and Integration.[6] The proposed DOE approach, together with the DOE agreements to provide NRC with additional information, acceptably addresses the NRC questions so that no information beyond that already provided, or agreed to be provided, will likely be required at the time of a potential license application.

It should be noted that the NRC review to date has been limited to the methodology portion of multiple barriers, and NRC is not addressing whether DOE has adequately identified multiple barriers or if DOE has demonstrated multiple barriers are present. The status and the detailed agreements (path forward) pertaining to all key technical issue subissues are provided in Table 1.1-3 and Appendix A.

Table 3.1-1. Status of Resolution of the System Description and Demonstration of Multiple Barriers Subissue

Key Technical Issue	Subissue	Status	Related Agreements*
Container Life and Source Term	Subissue 3—The Rate at Which Radionuclides in Spent Nuclear Fuels Are Released from the Engineered Barrier Subsystem through the Oxidation and Dissolution of Spent Nuclear Fuel	Closed-Pending	CLST.3.01
	Subissue 4—The Rate at Which the Radionuclides in High-level Waste Glass Are Leached and Released from the Engineered Barrier Subsystem	Closed-Pending	CLST.4.01
Total System Performance Assessment and Integration	Subissue 1—System Description and Demonstration of Multiple Barriers	Closed-Pending	TSPAI.1.01 TSPAI.1.02

*Related DOE and NRC agreements are associated with one or all acceptance criteria.

3.1.6 References

CRWMS M&O. "Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations." TDR–WIS–RL–000001. Revision 04 ICN 01. North Las Vegas, Nevada: DOE, Yucca Mountain Site Characterization Office. 2000a.

[6]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

———. “Total System Performance Assessment for the Site Recommendation.” TDR–WIS–PA–000001. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000b.

Mohanty, S., T.J. McCartin, and D. Esh. “Total-system Performance Assessment (TPA) Version 4.0 Code: Module Descriptions and User’s Guide.” San Antonio, Texas: CNWRA. 2002.

Mohanty, S., R. Codell, R.W. Rice, J. Weldy, Y. Lu, R.M. Byrne, T.J. McCartin, M. Jarzemba, and G.W. Wittmeyer. “System-Level Repository Sensitivity Analyses Using TPA Version 3.2 Code.” San Antonio, Texas: CNWRA. 1999.

NRC. NUREG–1804, “Yucca Mountain Review Plan.” Draft Report for Comment. Revision 2. Washington, DC: NRC. March 2002.

3.2 Scenario Analysis and Event Probability

3.2.1 Scenario Analysis

3.2.1.1 Description of Issue

A complete safety evaluation of a geologic repository for high-level waste requires consideration of potential future conditions affecting its behavior during the period of regulatory concern. This safety evaluation may be accomplished through scenario analysis, which is the systematic enumeration of features, events, and processes that can reasonably occur in the repository system. Scenario analysis facilitates identifying the possible ways in which the geologic repository environment can evolve so a defensible representation of the system can be included in the total system performance assessment.

A scenario is defined as the plausible future evolution of the repository system during the period of regulatory concern. A scenario includes a postulated sequence (or absence) of events and assumptions about initial and boundary conditions. A scenario analysis is composed of four steps: (i) identification of features, events, and processes relevant to the proposed high-level waste geologic repository; (ii) selection or screening of features, events, and processes important to estimating dose risk to a reasonably maximally exposed individual during the period of regulatory concern; (iii) formation of scenario classes from a screened or reduced collection of features, events, and processes; and (iv) selection or screening of the scenario classes for actual implementation into a total system performance assessment.

This section provides a review of the scenario analysis methodology implemented by DOE. Technical bases for scenario analysis are documented in analysis and model reports, CRWMS M&O (2000a), and other technical reports. The scenario analysis review is documented in two parts, one referring to the identification of features, events, and processes that affect compliance with the overall performance objective and the other referring to the identification of events with probabilities greater than 10^{-8} per year.

3.2.1.2 Relationship to Key Technical Issue Subissues

The identification of features, events, and processes important to repository safety is pertinent to all the key technical issue subissues. The subsequent sections incorporate applicable portions of these technical issue subissues, however, no effort was made to explicitly identify each subissue in the text. Features, events, and processes incorporated into the performance assessment are reviewed under the appropriate integrated subissues under model abstraction.

3.2.1.3 Importance to Postclosure Performance

A scenario analysis attempts to identify all features, events, and processes that could influence, directly or indirectly, dose risk from the proposed high-level waste repository to a reasonably maximally exposed individual. A well-implemented process for identification of these features, events, and processes helps to ensure relevant aspects of the proposed high-level waste repository, and associated implications to the dose risk, are studied. Appropriate identification and screening of scenario classes are intended to guarantee that all relevant sequences of events and processes are accounted for in the dose risk assessment. A well-documented compendium of features, events, and processes facilitates identification of the aspects analyzed in the evaluation of the repository safety and serves as a road map to the location of the analyses and their conclusions. Therefore, the goal of scenario analysis is to ensure that no aspect of the proposed high-level waste repository is overlooked in the evaluation of its safety.

3.2.1.4 Technical Basis

NRC developed a plan (2002) consistent with the acceptance criteria and review methods found in previous issue resolution status reports. A review of DOE approaches for development of a scenario analysis to support the total system performance assessment is provided in the following subsections. The review is organized according to the four acceptance criteria: (i) The Identification of an Initial List of Features, Events, and Processes Is Adequate; (ii) Screening of the Initial List of Features, Events, and Processes Is Appropriate; (iii) Formation of Scenario Classes Using the Reduced Set of Events Is Adequate; and (iv) Screening of Scenario Classes Is Appropriate.

3.2.1.4.1 The Identification of an Initial List of Features, Events, and Processes Is Adequate

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.2.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the adequacy of the identification of an initial list of features, events, and processes.

The process used to construct the initial list of features, events, and processes is detailed in CRWMS M&O (2000a, 2001a). DOE compiled a database of features, events, and processes potentially relevant to the proposed high-level waste repository (the Yucca Mountain Project Database of Features, Events, and Processes, hereon referred to as the database). This database is a collection of features, events, and processes from other radioactive waste disposal programs cataloged by the Nuclear Energy Agency of the Organization for Economic Co-operation and Development. This list was supplemented with entries from Yucca Mountain project literature; brainstorming and iterative reviews from experts; and feedback from DOE and NRC technical exchanges, Appendix 7 meetings, and NRC issue resolution status reports (CRWMS M&O 20001a). DOE acknowledges that construction of the list of features, events, and processes is an iterative process subject to refinement (CRWMS M&O, 2000a). DOE stated this list is open and may continue to expand if additional features, events, and processes are identified during the site recommendation process or the development of a potential license application (CRWMS M&O, 2000a).

A total of 1,808 entries, identified as primary, secondary, or classification, has been cataloged in the CRWMS M&O (2001b). Only primary and secondary entries correspond to actual features, events, and processes. Classification entries are intended to enhance the organization of the database. Primary entries have been given broad definitions so they encompass multiple secondary entries. It is expected that, by developing screening arguments for primary features, events, and processes, screening rationales for secondary features, events, and processes would follow. A total of 328 primary features, events, and processes has been identified in the database (CRWMS M&O, 2001a).

DOE argues that the list of features, events, and processes is comprehensive because these (i) have been identified from diverse backgrounds (from several international waste disposal programs) using a variety of methods (expert judgment, informal elicitation, event tree analysis, and stakeholder review) and (ii) have been subjected to iterative discussions and systematic classification (CRWMS M&O, 2000a). Also, DOE stated this list of features, events, and processes is indeed comprehensive (CRWMS M&O, 2001a) because few new elements have been identified in recent iterative reviews.

According to CRWMS M&O (2001a), the database may be updated by DOE through a systematic review of NRC issue resolution status reports, a review of a newer version (Version 1.2) of the Nuclear Energy Agency database, and the resolution of any outstanding NRC near-field environment audit issues identified in Pickett and Leslie (1999) and outstanding issues in NRC (2000).

NRC staff evaluated the list of features, events, and processes reported in several analysis and model reports and in the CRWMS M&O (2001b) and concluded that some aspects of the proposed high-level waste repository are not described in this list. For example, no item is listed in the database addressing response of the drip shield to static loads and seismic excitation. The database should contain elements to account for degradation of the drip shield caused by the interaction of seismic excitation with dead loads (e.g., rockfall or drift collapse), either for the screening argument of an existing feature, event, and process in the database or for a new entry. Entry 1.2.03.02.00 (Seismic Vibration Causes Container Failure)[1] assesses the effect of ground motion on the waste package and drip shield, without consideration of possible preexisting static loads (CRWMS M&O, 2000b, 2001c). Part of the screening argument for 2.1.06.06.00 (Effects and Degradation of Drip Shield) in CRWMS M&O (2001c) is based on an assumption that does not account for the possibility of static loads affecting the drip shield and, possibly, the waste package.

The database does not address the effect of trace metal cations on Alloy 22 and titanium corrosion and stress corrosion cracking, which is a possibility according to results recently reported by Barkatt and Gorman.[2]

At issue is the comprehensiveness of the list of features, events, and processes. For the issues identified in the previous two paragraphs, DOE and NRC have agreements on technical aspects that address outstanding concerns (e.g., Subissue 1 of Container Life and Source

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[1]In this chapter, features, events, and processes listed in the Yucca Mountain Project Database are referred to by the database entry number and title enclosed by parentheses [e.g., 2.1.07.02.00 (Mechanical Degradation or Collapse of Drift)]. The meaning of the database entry number in the form X.X.YY.ZZ.WW is described in CRWMS M&O (2001a).

[2]Barkatt, A. and J.A. Gorman. "Tests to Explore Specific Aspects of the Corrosion Resistance of C–22." *Nuclear Waste Technical Review Board Meeting, August 1, 2000*. Carson City, Nevada. 2000.

Term Key Technical Issue Agreement 14[3] and Subissue 3 of Evolution of the Near-Field Environment Key Technical Issue Agreement 4[4]). DOE agreed to revise descriptions and screening arguments of adequate features, events, and processes to enclose the two items listed previously.[5]

The definition of some primary features, events, and processes is too broad and nondescript to permit easy identification of those aspects included. For example, detailed processes related to the interaction of the ascending dike with the repository drift are not identified as features, events, and processes in the database. Instead, the database includes only general categories such as 1.2.04.04.00 (Magma Interacts with Waste) and 1.2.04.01.00 (Igneous Activity). This high-level definition of features, events, and processes may cause elements relevant to repository and dike interactions and interactions between magma and waste packages and spent nuclear fuel to be overlooked. Features, events, and processes related to magma/repository interactions that do not appear to be explicitly listed in the database include solid and fluid dynamics at the dike tip, vesiculation, plume dynamics, effect of drip shield on magma/repository interactions, geologic factors, threshold flow characteristics, gas segregation, alternate models of vent formation, effects of air shafts and drifts, consideration of flow segregation, localization of magma, recirculation of magma, and evolution of flow conditions. Canister/magma interactions that appear to have been missed include hoop stresses caused by differential expansion of the inner and outer waste packages, melting of materials, thermal shock, and phase changes in Alloy 22 because of the long-term exposure to elevated temperatures. Spent nuclear fuel/magma interactions that may have been missed include cladding response to high temperatures, cladding/fuel chemical reactions causing damage to the waste form (no credit is currently taken for the presence of cladding), mechanical shear, oxidation (during and posteruption), reworking of magma-borne spent nuclear fuel in tunnels and adits, and evolution of flow conditions.

In addition to the difficulty in outlining detailed items addressed by features, events, and processes with broad definitions, the broad definitions produce overlap among database entries, adding complexity to the identification of those aspects addressed by the list of features, events, and processes. Examples of features, events, and processes with broad definitions include (without being exhaustive)

- 1.1.12.01.00 (Accidents and Unplanned Events During Operation)—The entry 1.1.02.01.00 (Site Flooding During Construction and Operation) is explicitly identified in its definition as a particular instance of the former.

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[3]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000)." Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[4]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9-12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[5]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration–Features, Events, and Processes (May 15–17, 2001)." Letter (May 30) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

- 1.2.03.01.00 (Seismic Activity)—The entry 1.2.03.02.00 (Seismic Vibration Causes Container Failure) seems a particular instance of the former.

- 2.2.12.00.00 [Undetected Features (in Geosphere)]—This item is too broad for a clear screening argument to be developed. Undetected features relevant to repository performance may be considered in uncertainty and hazard estimates as suggested in the screening argument (CRWMS M&O, 2001c). Multiple features, events, and processes are related to features in the geosphere. For example, features at the repository horizon are also addressed in 1.1.07.00.00 (Repository Design). Thus, the precise scope of this database entry is not clear.

- 2.3.13.01.00 (Biosphere Characteristics)—The broad span of this item causes the scope to be unclear. For example, 2.3.13.02.00 (Biosphere Transport), 2.3.11.01.00 (Precipitation), and 2.4.09.02.00 (Animal Farms and Fisheries) seem to be instances of this entry.

Questions about the scope of several primary features, events, and processes and the differing levels of detail encompassed by them were presented to DOE at the May 15–17[6] and August 6–10,[7] 2001, DOE and NRC Technical Exchanges and Management Meetings on Total System Performance Assessment and Integration. At the May 15–17 meeting, NRC observed that 10 CFR Part 63 requires a systematic analysis of features, events, and processes that might affect the performance of a potential geologic repository at Yucca Mountain. Although it does not specify the manner by which features, events, and processes should be investigated, 10 CFR Part 63 requires that DOE "... provide the technical basis for either inclusion or exclusion of specific features, events, and processes... ." NRC is interested in a transparent, traceable, and technically defensible investigative process leading to a clear understanding of the DOE basis for consideration of features, events, and processes in a total system performance assessment. The varying levels of information used to describe the scope of primary features, events, and processes make it difficult to judge the comprehensiveness of the database.[8] Based on the documentation available, it was not possible for NRC to determine what aspects that might affect the performance of a potential geologic repository at Yucca Mountain were considered by DOE, and where particular features, events, and processes were addressed. Also, it was not evident that the list of features, events, and processes was consistent with transparency and traceability requirements (i.e., it was not evident that the list could be audited).

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[6]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration–Features, Events, and Processes (May 15–17, 2001)." Letter (May 30) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[7]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[8]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration–Features, Events, and Processes (May 15–17, 2001)." Letter (May 30) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

DOE stated that the list of secondary features, events, and processes is not intended to specify details of primary entries. The definitions of primaries enclose the secondary entries but, in general, have broader scopes. Secondary features, events, and processes are listed in the database to enable traceability and to identify the origin of the primary entry, not to enumerate all aspects addressed by the collection of primary features, events, and processes. DOE stated that the set of primary features, events, and processes should be judged for completeness and comprehensiveness.[9] If DOE adopts aspects of the Nuclear Energy Agency database, then DOE should justify the appropriateness and applicability to the proposed geologic repository at Yucca Mountain. Such information is not available in current DOE documentation.

At the August 6–10, 2001, meeting, DOE stated that it would revise the descriptions of all of the features, events, and processes to (i) better identify all components included in a feature, event, and process; (ii) ensure full incorporation of relevant aspects of a feature, event, and process; (iii) eliminate use of secondary entry terminology, yet retain traceability to the Nuclear Energy Agency database or other source documents; and (iv) make the level-of-detail more consistent, where possible, with a clear differentiation between features, events, and processes and modeling aspects. DOE stated that it would be developing level of detail criteria and refining entries in the database consistent with these criteria. Finally, DOE stated that, besides revising screening arguments for excluded features, events, and processes to improve technical basis descriptions, it will clarify how features, events, and processes screened for inclusion are addressed in the total system performance assessment.[10]

Various agreements addressing the issues highlighted in Section 3.2.1.4.1 were reached at the May 15–17 and August 6–10, 2001, DOE and NRC Technical Exchanges and Management Meetings on Total System Performance Assessment and Integration, and are listed in Section 3.2.1.5.

3.2.1.4.2 Screening of the Initial List of Features, Events, and Processes Is Appropriate

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.2.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the appropriateness of the screening of the initial list of features, events, and processes.

DOE classified the 328 primary features, events, and processes in CRWMS M&O (2001b) into process model subject areas. Eleven analysis and model reports discuss developing screening arguments for features, events, and processes, which are listed in Table 3.2.1-1. Database entries were assigned to more than one analysis and model report because, in general, the

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[9]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration–Features, Events, and Processes (May 15–17, 2001)." Letter (May 30) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[10]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

Table 3.2.1-1. Set of Features, Events, and Processes Analysis and Model Reports for Developing Screening Arguments

Analysis and Model Report Title	Control Identification	Revision/ICN	Year
Features, Events, and Processes in Unsaturated Zone Flow and Transport	ANL–NBS–MD–000001	01/00	2001
Features, Events, and Processes in Saturated Zone Flow and Transport	ANL–NBS–MD–000002	01/00	2000
Evaluation of the Applicability of Biosphere-Related Features, Events, and Processes	ANL–MGR–MD–000011	01/00	2001
Features, Events, and Processes: Screening for Disruptive Events	ANL–WIS–MD–000005	00/01	2000
Features, Events, and Processes: Screening of Processes and Issues in Drip Shield and Waste Package Degradation	ANL–EBS–PA–000002	01/00	2001
Miscellaneous Waste-Form Features, Events, and Processes	ANL–WIS–MD–000009	00/01	2000
Clad Degradation—Features, Events, and Processes Screening Arguments	ANL–WIS–MD–000008	00/01	2000
Colloid-Associated Concentration Limits: Abstraction and Summary	ANL–WIS–MD–000012	00/01	2000
Features, Events, and Processes in Thermal Hydrology and Coupled Processes	ANL–NBS–MD–000004	01/00	2001
Engineered Barrier Subsystem Features, Events, and Processes/Degradation Models Abstraction	ANL–WIS–PA–000002	01/00	2001
Features, Events, and Processes: System Level and Criticality	ANL–WIS–MD–000019	00/00	2000

entries are relevant to more that one process model subject area. Entries addressed by more than one analysis and model report are denoted as shared features, events, and processes. Within an analysis and model report, the terms included and excluded are used to conclude if a feature-event process is relevant or irrelevant (with respect to the dose risk of the proposed high-level waste repository) to a given process-level model. Thus, shared features, events, and processes were given several screening assignments (e.g., included/excluded) by the various analysis and model reports. These screening decisions have not yet been integrated into a single screening decision, but DOE is intending to do so (CRWMS M&O, 2000a).

Each primary database entry was screened as included or excluded on the basis of three criteria developed in the DOE Interim Guidance.[11] These criteria are regulatory, probability, and consequence (CRWMS M&O, 2000a). The Regulatory Criterion refers to the exclusion of primary features, events, and processes from the performance assessment because they are not in accordance with the regulatory guidance[12] or are not applicable by regulation. The Probability Criterion states that features, events, and processes with a probability of occurrence of less than 10^{-4} in 10,000 years can be excluded from consideration in the total system performance assessment. Finally, the Consequence Criterion states that features, events, and processes whose exclusion would not significantly change the expected annual dose may be excluded from the total system performance assessment (CRWMS M&O, 2000a). A summary of the screening decisions (e.g., included/excluded) and the basis (regulatory, probability, or consequence) for the 328 primary features, events, and processes is available in CRWMS M&O (2000a), and the electronic version (in Microsoft® Access) is available in CRWMS M&O (2001b).

DOE plans to update screening arguments and screening decisions in analysis and model reports in accordance with a lower thermal load design [current screening discussions are based on a reference repository design described in CRWMS M&O (2000a)]. Additional effort will focus on integration of screening information and primary descriptions for shared features, events, and processes, and explicit identification of the scenario class (nominal, disruptive, or human intrusion) for each of the elements in the list of features, events, and processes screened as included. Screening arguments will be revised to be entirely consistent with the Interim Guidance[13] (CRWMS M&O, 2001a). As mentioned in Section 3.2.1.4.1, it is also expected that DOE will refine the feature, event, and process descriptions to address NRC concerns per the agreements reached during the May 15–17 and August 6–10, 2001, DOE and NRC Technical Exchanges and Management Meetings on Total System Performance Assessment and Integration.

Staff evaluated screening arguments in analysis and model reports listed in Table 3.2.1-1. Screening arguments in some analysis and model reports depend on assumptions yet to

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[11]Dyer, J.R. "Revised Interim Guidance Pending Issuance of New U.S. Nuclear Regulatory Commission (NRC) Regulations (Revision 01, July 22, 1999), for Yucca Mountain Nevada." Letter (September 3) to D.R. Wilkins, CRWMS M&O. Washington, DC: DOE. 1999.

[12]Ibid.

[13]Ibid.

be verified (CRWMS M&O, 2000c, 2001d,e). Some screening arguments are indicated to be preliminary {e.g., 2.1.07.01.00 [Rockfall (Large Block)]; 1.2.02.01.00 (Fractures); 1.2.02.02.00 (Faulting); 1.2.03.01.00 (Seismic Activity) in CRWMS M&O (2000b); 2.1.14.14.00 (Out-of-Package Criticality, Fuel/Magma Mixture) in CRWMS M&O (2000d); and items listed in Attachment I in CRWMS M&O (2001f)}. It is acknowledged that to-be-verified assumptions are properly tracked by DOE, that work reported in the cited analysis and model reports constitutes work in progress, and that these documents will be revised to disclose more definite screening arguments, as discussed at the May 2001 technical exchange.[14]

A summary of the detailed evaluation of the screening arguments is contained in Table 3.2.1-2, which lists the 328 primary features, events, and processes of CRWMS M&O (2001a), in ascending order of database tracking numbers. In Table 3.2.1-2, features, events, and processes have been classified in accordance with the integrated subissue structure. Elements not pertinent to a given integrated subissue are indicated by a long dash (–). Features, events, and processes not clearly belonging to any of the integrated subissues are listed in the Orphan column. The DOE screening decision is symbolized by I and E (included and excluded), and the initial staff evaluation is labeled as S or U (satisfactory or unsatisfactory). Those items classified with U were discussed at the May 15–17,[15] August 6–10,[16] and September 5,[17] 2001, DOE and NRC Technical Exchanges and Management Meetings, and agreements are available. The column labeled Technical Exchange in Table 3.2.1-2 contains tracking numbers used at these technical exchanges and management meetings to identify the NRC comments. The same tracking numbers are used in Appendix B. A notation of I/U has been used in Table 3.2.1-2 to denote screening arguments where inconsistencies have been identified. The symbol I/U is not intended as a criticism to the way the features, events, and processes have been included in the model abstraction. An isolated U (i.e., not accompanied by I or E) in Table 3.2.1-2 indicates a feature, event, and process not evaluated in a suggested integrated subissue scope. Additional details on the evaluation of screening arguments are available in Appendix B. The symbol RF identifies those features, events, and processes with screening arguments that appeal to requirements in 10 CFR Part 63 and appearing adequate. The symbol QA highlights those features, events, and processes with screening arguments invoking the implementation of quality assurance procedures. These screening arguments appear adequate pending the development of quality assurance procedures with

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[14]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration–Features, Events, and Processes (May 15–17, 2001)." Letter (May 30) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[15]Ibid.

[16]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[17]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

Table 3.2.1-2. Summary of Features, Events, and Processes Screening Argument Evaluation

Database Tracking Number	Feature, Event, and Process Name	ENG1*	ENG2	ENG3	ENG4	UZ1	UZ2	UZ3	SZ1	SZ2	Direct1	Direct2	Dose1	Dose2	Dose3	Orphan	Technical Exchange
0.1.02.00.00	Timescales of concern	–	–	–	–	–	–	–	–	–	–	–	–	–	–	I	–
0.1.03.00.00	Spatial domain of concern	–	–	–	–	–	–	–	–	–	–	–	–	–	–	I	–
0.1.09.00.00	Regulatory requirements and exclusions	I	I	I	I	I	I	I	I	I	I	I	I	I	I	–	–
0.1.10.00.00	Model and data issues	–	–	–	–	–	–	–	–	–	–	–	–	–	–	I E/S	–
1.1.01.01.00	Open-site investigation boreholes	–	–	–	–	E/QA	E/QA	–	–	–	–	–	–	–	–	–	–
1.1.01.02.00	Loss of integrity of borehole seals	–	–	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–
1.1.02.00.00	Excavation/construction	–	I E/S	–	–	–	I E/S	–	–	–	U †	–	–	–	–	–	75
1.1.02.01.00	Site flooding (during construction and operation)	–	–	–	–	E/QA	–	–	–	–	–	–	–	–	–	–	–
1.1.02.02.00	Effects of preclosure ventilation	–	I	–	–	–	I	–	–	–	–	–	–	–	–	–	–
1.1.02.03.00	Undesirable materials left	E/A	–	–	E/U	–	E/S	E/U	–	–	–	–	–	–	–	–	57
1.1.03.01.00	Error in waste or backfill emplacement	–	E/QA	–	–	–	–	–	–	–	–	–	–	–	–	–	–
1.1.04.01.00	Incomplete closure	–	–	–	–	–	E/S	–	–	–	U	–	–	–	–	–	75
1.1.05.00.00	Records and markers, repository	–	–	–	–	–	–	–	–	–	–	–	–	–	S	–	–
1.1.07.00.00	Repository design	–	I E/QA	I	I E/QA	–	I	–	–	–	I	–	–	–	–	–	–
1.1.08.00.00	Quality control	I E/QA	I E/QA	I	I E/QA	–	–	–	–	–	–	–	–	–	–	–	–
1.1.09.00.00	Schedule and planning	–	–	–	–	–	–	–	–	–	–	–	–	–	–	E/S	–
1.1.10.00.00	Administrative control, repository site	–	–	–	–	–	–	–	–	–	–	–	–	–	–	I	–
1.1.11.00.00	Monitoring of repository	–	–	–	–	–	–	–	–	–	–	–	–	–	–	E/S	–
1.1.12.01.00	Accidents and unplanned events during operation	E/QA	E/QA	–	E/QA	–	–	–	–	–	–	–	–	–	–	–	–
1.1.13.00.00	Retrievability	–	–	I	I	–	–	–	–	–	–	–	–	–	–	–	–
1.2.01.01.00	Tectonic activity, large scale	–	E/S	–	–	–	–	–	–	–	E/S	–	–	–	–	–	–
1.2.02.01.00	Fractures	–	I E/S	–	–	E/A	I E/A	–	E/S	–	–	–	–	–	–	–	68
1.2.02.02.00	Faulting	–	I E/A	–	–	–	I E/A	–	I E/A	–	–	–	–	–	–	–	J-25
1.2.02.03.00	Fault movement shears waste container	–	E/A	–	–	–	–	–	–	–	–	–	–	–	–	–	J-25, J-26
1.2.03.01.00	Seismic activity	–	E/A	–	I	–	E/A	–	E/A	–	–	–	–	–	–	–	J-27
1.2.03.02.00	Seismic vibration causes container failure	–	I E/A	–	–	–	–	–	–	–	–	–	–	–	–	–	78, J-25
1.2.03.03.00	Seismicity associated with igneous activity	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–	–	–
1.2.04.01.00	Igneous activity	–	I	–	–	–	–	–	–	–	I E/S	I	I	I	I	–	–
1.2.04.02.00	Igneous activity causes changes to rock properties	–	E/S	–	–	–	E/S	E/U	E/S	E/S	E/U	–	–	–	–	–	J-22
1.2.04.03.00	Igneous intrusion into repository	–	I	–	–	–	–	–	–	–	I	–	–	–	–	–	–
1.2.04.04.00	Magma interacts with waste	I	I	–	I	–	–	–	–	–	I	–	–	–	–	–	–
1.2.04.05.00	Magmatic transport of waste	–	I E/S	–	–	–	–	–	–	–	I E/RF	I	–	–	–	–	–
1.2.04.06.00	Basaltic cinder cone erupts through the repository	–	I	–	–	–	–	–	–	–	I	I	–	–	–	–	–

Table 3.2.1-2. Summary of Features, Events, and Processes Screening Argument Evaluation (continued)

Database Tracking Number	Feature, Event, and Process Name	ENG1	ENG2	ENG3	ENG4	UZ1	UZ2	UZ3	SZ1	SZ2	Direct1	Direct2	Dose1	Dose2	Dose3	Orphan	Technical Exchange
1.2.04.07.00	Ashfall	–	–	–	–	–	–	–	E/U	–	–	–	I E/U	I E/U	U	–	8, 19
1.2.05.00.00	Metamorphism	–	–	–	–	–	–	–	–	–	–	–	–	–	–	E/S	–
1.2.06.00.00	Hydrothermal activity	–	–	E/A	–	–	E/U	–	E/S	E/A	–	–	–	–	–	–	4, J-23
1.2.07.01.00	Erosion/denudation	–	–	–	–	E/U	–	–	–	–	–	–	–	I	I E/S	–	J-16
1.2.07.02.00	Deposition	–	–	–	–	E/S	–	–	–	–	–	–	–	I	I	–	–
1.2.08.00.00	Diagenesis	–	–	–	–	–	–	E/S	–	–	–	–	–	–	–	–	–
1.2.09.00.00	Salt diapirism and dissolution	–	–	–	–	–	–	–	–	–	–	–	–	–	–	E/S	–
1.2.09.01.00	Diapirism	–	–	–	–	–	–	–	–	–	–	–	–	–	–	E/S	–
1.2.09.02.00	Large-scale dissolution	–	–	–	–	–	E/S	E/S	E/S	E/S	–	–	–	–	–	–	–
1.2.10.01.00	Hydrological response to seismic activity	–	–	–	–	E/S	E/S	–	E/S	–	–	–	–	–	–	–	J-17
1.2.10.02.00	Hydrologic response to igneous activity	–	–	–	–	E/U	E/S	–	E/S	–	–	–	–	–	–	–	–
1.3.01.00.00	Climate change, global	–	–	–	–	I	–	–	–	–	–	–	I	I	I	–	–
1.3.04.00.00	Periglacial effects	–	–	–	–	E/U	–	–	–	–	–	–	–	E/S	E/S	–	J-18
1.3.05.00.00	Glacial and ice sheet effects, local	–	–	–	–	E/S	–	–	–	–	–	–	E/S	E/S	E/S	–	–
1.3.07.01.00	Drought/water table decline	–	–	–	–	–	–	–	E/A	E/A	–	–	E/A	–	E/A	–	11
1.3.07.02.00	Water table rise	–	–	–	–	–	–	–	I	I	–	–	U	U	U	–	19
1.4.01.00.00	Human influences on climate	–	–	–	–	E/S	–	–	–	–	–	–	E/RF	–	E/RF	–	–
1.4.01.01.00	Climate modification increases recharge	–	–	–	–	I	I	–	–	–	–	–	I	–	–	–	–
1.4.01.02.00	Greenhouse gas effects	–	–	–	–	E/S	–	–	–	–	–	–	–	–	E/RF	–	–
1.4.01.03.00	Acid rain	–	–	–	–	E/RF	–	–	–	–	–	–	–	–	E/RF	–	–
1.4.01.04.00	Ozone layer failure	–	–	–	–	E/S	–	–	–	–	–	–	–	–	E/RF	–	–
1.4.02.01.00	Deliberate human intrusion	–	–	–	–	–	–	–	–	–	–	–	–	–	–	E/S	–
1.4.02.02.00	Inadvertent human intrusion	–	–	–	–	–	–	–	–	–	–	–	–	–	–	I	–
1.4.03.00.00	Unintrusive site investigation	–	–	–	–	–	–	–	–	–	–	–	–	–	–	E/S	–
1.4.04.00.00	Drilling activities (human intrusion)	–	–	–	–	–	–	–	–	–	–	–	–	–	–	I E/RF	–
1.4.04.01.00	Effects of drilling intrusion	–	–	–	–	–	–	–	–	–	–	–	–	–	–	I	–
1.4.04.02.00	Abandoned and undetected boreholes	–	–	–	–	E/S	E/S	–	–	–	–	–	–	–	–	–	–
1.4.05.00.00	Mining and other underground activities (human intrusion)	–	–	–	–	–	–	–	–	–	–	–	–	–	–	E/S	–
1.4.06.01.00	Altered soil or surface water chemistry	–	–	–	–	–	–	–	–	E/U	–	–	–	–	E/RF	–	7
1.4.07.01.00	Water management activities	–	–	–	–	–	–	–	I E/S	I E/S	–	–	I E/U	I E/U	I E/U	–	18
1.4.07.02.00	Wells	–	–	–	–	–	–	–	I	–	–	–	I E/RF	–	I E/RF	–	–
1.4.08.00.00	Social and institutional developments	–	–	–	–	–	–	–	–	–	–	–	E/RF	E/RF	E/RF	–	–
1.4.09.00.00	Technological developments	–	–	–	–	–	–	–	–	–	–	–	E/RF	E/RF	E/RF	–	–
1.4.11.00.00	Explosions and crashes (human activities)	–	–	–	–	–	–	–	–	–	–	–	–	E/RF	–	–	–
1.5.01.01.00	Meteorite impact	–	–	–	–	–	–	–	–	–	–	–	–	E/S	E/S	–	–
1.5.01.02.00	Extraterrestrial events	–	–	–	–	–	–	–	–	–	–	–	–	–	E/S	–	–
1.5.02.00.00	Species evolution	–	–	–	–	–	–	–	–	–	–	–	–	–	E/RF	–	–

Table 3.2.1-2. Summary of Features, Events, and Processes Screening Argument Evaluation (continued)

Database Tracking Number	Feature, Event, and Process Name	ENG1	ENG2	ENG3	ENG4	UZ1	UZ2	UZ3	SZ1	SZ2	Direct1	Direct2	Dose1	Dose2	Dose3	Orphan	Technical Exchange
1.5.03.01.00	Changes in the Earth's magnetic field	–	–	–	–	–	–	–	–	–	–	–	–	–	–	E/S	–
1.5.03.02.00	Earth tides	–	–	–	–	–	–	–	–	–	–	–	–	–	–	E/S	–
2.1.01.01.00	Waste inventory	–	–	–	I	–	–	–	–	–	I	–	I	I	–	–	–
2.1.01.02.00	Codisposal/co-location of waste	I	–	I	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.01.03.00	Heterogeneity of waste forms	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.01.04.00	Spatial heterogeneity of emplaced waste	E/A	–	–	E/A	–	–	–	–	–	–	–	–	–	–	–	48
2.1.02.01.00	Defense spent nuclear fuel degradation, alteration, and dissolution	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.02.00	Commercial spent nuclear fuel alteration, dissolution, and radionuclide release	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.03.00	Glass degradation, alteration, and dissolution	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.04.00	Alpha recoil enhances dissolution	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.05.00	Glass cracking and surface area	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.06.00	Glass recrystallization	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.07.00	Gap and grain release of Cs, I	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.08.00	Pyrophoricity	–	–	E/S	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.09.00	Void space (in glass container)	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.10.00	Cellulosic degradation	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.11.00	Waterlogged rods	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.12.00	Cladding degradation before YMP receives it	–	–	–	I E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.13.00	General corrosion of cladding	–	–	–	E/U	–	–	–	–	–	–	–	–	–	–	–	50
2.1.02.14.00	Microbial corrosion (MIC) of cladding	–	–	–	I/U	–	–	–	–	–	–	–	–	–	–	–	51
2.1.02.15.00	Acid corrosion of cladding from radiolysis	–	–	–	I/U	–	–	–	–	–	–	–	–	–	–	–	49,51
2.1.02.16.00	Localized corrosion (pitting) of cladding	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.17.00	Localized corrosion (crevice corrosion) of cladding	–	–	–	E/A	–	–	–	–	–	–	–	–	–	–	–	47
2.1.02.18.00	High dissolved silica content of waters enhances corrosion of cladding	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.19.00	Creep rupture of cladding	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.20.00	Pressurization from He production causes cladding failure	–	–	–	I/U	–	–	–	–	–	–	–	–	–	–	–	41
2.1.02.21.00	Stress corrosion cracking (SCC) of cladding	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.22.00	Hydride embrittlement of cladding	–	–	–	E/U	–	–	–	–	–	–	–	–	–	–	–	53
2.1.02.23.00	Cladding unzipping	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.24.00	Mechanical failure of cladding	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.25.00	Defense spent nuclear fuel cladding degradation	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.26.00	Diffusion controlled cavity growth	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.27.00	Localized corrosion perforation from fluoride	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.28.00	Various features of the approximately 250 Defense spent nuclear fuel types and grouping into waste categories	–	–	–	I E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.29.00	Flammable gas generation from Defense spent nuclear fuel	–	–	–	I E/S	–	–	–	–	–	–	–	–	–	–	–	–

Table 3.2.1-2. Summary of Features, Events, and Processes Screening Argument Evaluation (continued)

Database Tracking Number	Feature, Event, and Process Name	ENG1	ENG2	ENG3	ENG4	UZ1	UZ2	UZ3	SZ1	SZ2	Direct1	Direct2	Dose1	Dose2	Dose3	Orphan	Technical Exchange
2.1.03.01.00	Corrosion of waste containers	**I**	–	**I**	–	–	–	–	–	–	–	–	–	–	–	–	–
2.1.03.02.00	Stress corrosion cracking of waste containers	**I** **E/A**	**I** **E/A**	**I**	–	–	–	–	–	–	–	–	–	–	–	–	34
2.1.03.03.00	Pitting of waste containers	**I**	–	**I**	–	–	–	–	–	–	–	–	–	–	–	–	–
2.1.03.04.00	Hydride cracking of waste containers	**E/S**	–	**E/S**	–	–	–	–	–	–	–	–	–	–	–	–	–
2.1.03.05.00	Microbially mediated corrosion of waste container	**I** **E/A**	–	**I**	–	–	–	–	–	–	–	–	–	–	–	–	30
2.1.03.06.00	Internal corrosion of waste container	**E/S**	–	**I**	–	–	–	–	–	–	–	–	–	–	–	–	–
2.1.03.07.00	Mechanical impact on waste container	–	**E/S**	–	–	–	–	–	–	–	–	–	–	–	–	–	–
2.1.03.08.00	Juvenile and early failure of waste containers	**I** **E/A**	**I** **E/A**	–	–	–	–	–	–	–	–	–	–	–	–	–	35
2.1.03.09.00	Copper corrosion	**E/S**	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–
2.1.03.10.00	Container healing	**E/S**	–	**I**	–	–	–	–	–	–	–	–	–	–	–	–	–
2.1.03.11.00	Container form	**E/S**	**E/U**	**E/S**	–	–	–	–	–	–	**E/S**	–	–	–	–	–	J-1
2.1.03.12.00	Container failure (long-term)	**I**	**I**	**I**	–	–	–	–	–	–	**U**	–	–	–	–	–	75
2.1.04.01.00	Preferential pathways in backfill	–	–	**I**	–	–	–	–	–	–	–	–	–	–	–	–	–
2.1.04.02.00	Physical and chemical properties of backfill	–	**E/S**	**I**	**I**	–	–	–	–	–	–	–	–	–	–	–	–
2.1.04.03.00	Erosion or dissolution of backfill	–	**E/S**	**E/S**	**E/S**	–	–	–	–	–	–	–	–	–	–	–	–
2.1.04.04.00	Mechanical effects of backfill	–	**E/S**	–	–	–	–	–	–	–	**E/S**	–	–	–	–	–	–
2.1.04.05.00	Backfill evolution	–	**E/S**	**I**	**I**	–	–	–	–	–	–	–	–	–	–	–	–
2.1.04.06.00	Properties of bentonite	–	–	–	–	–	–	–	–	–	–	–	–	–	–	**E/S**	–
2.1.04.07.00	Buffer characteristics	–	–	–	–	–	–	–	–	–	–	–	–	–	–	**E/S**	–
2.1.04.08.00	Diffusion in backfill	–	–	–	**E/S**	–	–	–	–	–	–	–	–	–	–	–	–
2.1.04.09.00	Radionuclide transport through backfill	–	–	–	**E/S**	–	–	–	–	–	–	–	–	–	–	–	–
2.1.05.01.00	Seal physical properties	–	–	–	–	**E/U**	–	–	–	–	–	–	–	–	–	–	J-19
2.1.05.02.00	Groundwater flow and radionuclide transport in seals	–	–	–	–	**E/U**	**E/U**	–	–	–	–	–	–	–	–	–	J-19
2.1.05.03.00	Seal degradation	–	–	–	–	**E/U**	–	–	–	–	–	–	–	–	–	–	J-19
2.1.06.01.00	Degradation of cementitious materials in drift	–	**I**	**I** **E/A**	–	–	–	**U**	–	–	–	–	–	–	–	–	J-3
2.1.06.02.00	Effects of rock reinforcement materials	–	**I**	**I**	–	–	**I**	–	–	–	–	–	–	–	–	–	–
2.1.06.03.00	Degradation of the liner	–	**E/S**	**I**	**E/S**	–	–	**E/S**	–	–	–	–	–	–	–	–	–
2.1.06.04.00	Flow through the liner	–	–	**E/S**	**E/S**	–	–	–	–	–	–	–	–	–	–	–	–
2.1.06.05.00	Degradation of invert and pedestal	–	**E/U**	**E/S**	–	–	–	**E/U**	–	–	–	–	–	–	–	–	J-2, J-4
2.1.06.06.00	Effects and degradation of drip shield	**I** **E/U**	**I** **E/A**	**I** **E/A**	–	–	–	–	–	–	–	–	–	–	–	–	39
2.1.06.07.00	Effects at material interfaces	**E/A**	–	**I**	**E/S**	–	–	–	–	–	–	–	–	–	–	–	29
2.1.07.01.00	Rockfall (large block)	–	**E/A**	–	**E/A**	–	–	–	–	–	–	–	–	–	–	–	79
2.1.07.02.00	Mechanical degradation or collapse of drift	**E/A**	**E/A**	**E/A**	–	–	–	–	–	–	**U**	–	–	–	–	–	75, 77
2.1.07.03.00	Movement of containers	–	**E/S**	–	–	–	–	–	–	–	–	–	–	–	–	–	–
2.1.07.04.00	Hydrostatic pressure on container	–	**E/S**	–	–	–	–	–	–	–	–	–	–	–	–	–	–
2.1.07.05.00	Creeping of metallic materials in the engineered barrier system	–	**E/A**	–	–	–	–	–	–	–	–	–	–	–	–	–	37
2.1.07.06.00	Floor buckling	**E/A**	**E/A**	–	**E/A**	–	–	–	–	–	–	–	–	–	–	–	56

Table 3.2.1-2. Summary of Features, Events, and Processes Screening Argument Evaluation (continued)

Database Tracking Number	Feature, Event, and Process Name	ENG1	ENG2	ENG3	ENG4	UZ1	UZ2	UZ3	SZ1	SZ2	Direct1	Direct2	Dose1	Dose2	Dose3	Orphan	Technical Exchange
2.1.08.01.00	Increased unsaturated water flux at the repository	–	–	–	–	–	I	–	–	–	–	–	–	–	–	–	–
2.1.08.02.00	Enhanced influx (Philip's drip)	–	–	–	–	–	I	–	–	–	–	–	–	–	–	–	–
2.1.08.03.00	Repository dryout due to waste heat	–	–	–	–	–	I	–	–	–	–	–	–	–	–	–	–
2.1.08.04.00	Cold traps	–	–	E/A	–	–	E/A	–	–	–	–	–	–	–	–	–	59
2.1.08.05.00	Flow through invert	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.08.06.00	Wicking in waste and engineered barrier system	–	–	I	–	–	I	–	–	–	–	–	–	–	–	–	–
2.1.08.07.00	Pathways for unsaturated flow and transport in the waste and engineered barrier system	–	–	I E/A	I	–	–	–	–	–	–	–	–	–	–	–	42
2.1.08.08.00	Induced hydrological changes in the waste and engineered barrier system	–	–	I	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.08.09.00	Saturated groundwater flow in waste and engineered barrier system	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.08.10.00	Desaturation/dewatering of the repository	–	–	–	I	–	I	–	–	–	–	–	–	–	–	–	–
2.1.08.11.00	Resaturation of repository	–	–	I	–	–	I	–	–	–	–	–	–	–	–	–	–
2.1.08.12.00	Drainage with transport, sealing and plugging	–	–	E/S	–	–	E/S	–	–	–	–	–	–	–	–	–	–
2.1.08.13.00	Drains	–	–	E/S	–	–	E/S	–	–	–	–	–	–	–	–	–	–
2.1.08.14.00	Condensation on underside of drip shield	–	–	E/S	–	–	E/S	–	–	–	–	–	–	–	–	–	–
2.1.08.15.00	Waste-form and backfill consolidation	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.01.00	Properties of the potential carrier plume in the waste and engineered barrier system	–	–	I	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.02.00	Interaction with corrosion products	–	–	E/A	I	–	–	–	–	–	–	–	–	–	–	–	54
2.1.09.03.00	Volume increase of corrosion products	E/A	E/S	–	–	–	–	–	–	–	–	–	–	–	–	–	36
2.1.09.04.00	Radionuclide solubility, solubility limits, and speciation in the waste form and engineered barrier system	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.05.00	In-drift sorption	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.06.00	Reduction-oxidation potential in waste and engineered barrier system	E/S	–	I	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.07.00	Reaction kinetics in waste and engineered barrier system	I E/A	–	I E/A	I E/S	–	–	–	–	–	–	–	–	–	–	–	55
2.1.09.08.00	Chemical gradients/enhanced diffusion in waste and engineered barrier system	–	–	I	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.09.00	Electrochemical effects (electrophoresis, galvanic coupling) in waste and engineered barrier system	E/S	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.10.00	Secondary phase effects on dissolved radionuclide concentrations at the waste form	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.11.00	Waste-rock contact	–	–	E/S	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.12.00	Rind (altered zone) formation in waste, engineered barrier system, and adjacent rock	–	I E/A	I	I	E/A	I E/A	E/S	–	–	–	–	–	–	–	–	63
2.1.09.13.00	Complexation by organics in waste and engineered barrier system	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.14.00	Colloid formation in waste and engineered barrier system	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–

Table 3.2.1-2. Summary of Features, Events, and Processes Screening Argument Evaluation (continued)

Database Tracking Number	Feature, Event, and Process Name	ENG1	ENG2	ENG3	ENG4	UZ1	UZ2	UZ3	SZ1	SZ2	Direct1	Direct2	Dose1	Dose2	Dose3	Orphan	Technical Exchange
2.1.09.15.00	Formation of true colloids in waste and engineered barrier system	–	–	–	**I**	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.16.00	Formation of pseudo-colloids (natural) in waste and engineered barrier system	–	–	–	**I**	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.17.00	Formation of pseudo-colloids (corrosion products) in waste and engineered barrier system	–	–	–	**I**	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.18.00	Microbial colloid transport in the waste and engineered barrier system	–	–	–	**E/S**	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.19.00	Colloid transport and sorption in the waste and engineered barrier system	–	–	–	**E/S**	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.20.00	Colloid filtration in the waste and engineered barrier system	–	–	–	**E/S**	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.21.00	Suspensions of particles larger than colloids	–	–	–	**E/U**	–	–	**E/U**	**I**	**U**	–	–	**I**	–	–	–	J-5, 5
2.1.09.22.00	Colloid sorption at the air-water interface	–	–	–	–	–	–	**E/S**	–	**E/S**	–	–	–	–	–	–	–
2.1.09.23.00	Colloidal stability and concentration dependence on aqueous chemistry	–	–	–	–	–	–	**I**	–	**I**	–	–	–	–	–	–	–
2.1.09.24.00	Colloidal diffusion	–	–	–	–	–	–	**I**	–	**I**	–	–	–	–	–	–	–
2.1.09.25.00	Colloidal phases are produced by coprecipitation (in waste and engineered barrier system)	–	–	–	**I**	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.26.00	Colloid gravitational settling	–	–	–	**E/S**	–	–	**E/S**	–	–	–	–	–	–	–	–	–
2.1.10.01.00	Biological activity in waste and engineered barrier system	**I** **E/S**	–	**I**	**I**	–	–	–	–	–	–	–	–	–	–	–	–
2.1.11.01.00	Heat output/temperature in waste and engineered barrier system	**I**	**I**	**I**	**I**	–	**I**	–	–	–	–	–	–	–	–	–	–
2.1.11.02.00	Nonuniform heat distribution/edge effects in repository	–	–	**I**	**I**	–	**I** **E/A**	–	–	–	–	–	–	–	–	–	65
2.1.11.03.00	Exothermic reactions in waste and engineered barrier system	–	–	**E/S**	**E/S**	–	–	–	–	–	–	–	–	–	–	–	–
2.1.11.04.00	Temperature effects/coupled processes in waste and engineered barrier system	**I**	**I**	**I**	**I**	–	–	–	–	–	–	–	–	–	–	–	–
2.1.11.05.00	Differing thermal expansion of repository components	**I** **E/U**	**I** **E/U**	–	–	–	–	–	–	–	–	–	–	–	–	–	38
2.1.11.06.00	Thermal sensitization of waste containers increases fragility	**I**	**I**	–	–	–	–	–	–	–	–	–	–	–	–	–	–
2.1.11.07.00	Thermally induced stress changes in waste and engineered barrier system	–	**I**	–	**I** **E/S**	–	–	–	–	–	–	–	–	–	–	–	–
2.1.11.08.00	Thermal effects: chemical and microbiological changes in the waste and engineered barrier system	–	–	**I**	**I** **E/S**	–	–	–	–	–	–	–	–	–	–	–	–
2.1.11.09.00	Thermal effects on liquid or two-phase fluid flow in the waste and engineered barrier system	–	–	**I**	**I**	–	–	–	–	–	–	–	–	–	–	–	–
2.1.11.10.00	Thermal effects on diffusion (Soret effect) in waste and engineered barrier system	**E/S**	–	–	**E/S**	–	–	–	–	–	–	–	–	–	–	–	–
2.1.12.01.00	Gas generation	**E/S**	–	**E/U**	–	–	–	–	–	–	–	–	–	–	–	–	60
2.1.12.02.00	Gas generation (He) from fuel decay	**E/S**	–	–	**E/S**	–	–	–	–	–	–	–	–	–	–	–	–
2.1.12.03.00	Gas generation (H_2) from metal corrosion	**E/S**	–	**E/S**	–	–	–	–	–	–	–	–	–	–	–	–	–

Table 3.2.1-2. Summary of Features, Events, and Processes Screening Argument Evaluation (continued)

Database Tracking Number	Feature, Event, and Process Name	ENG1	ENG2	ENG3	ENG4	UZ1	UZ2	UZ3	SZ1	SZ2	Direct1	Direct2	Dose1	Dose2	Dose3	Orphan	Technical Exchange
2.1.12.04.00	Gas generation (CO_2, CH_4, H_2S) from microbial degradation	**E/S**	–	**E/S**	–	–	–	–	–	–	–	–	–	–	–	–	–
2.1.12.05.00	Gas generation from concrete	–	–	**E/U**	–	–	–	–	–	–	–	–	–	–	–	–	60
2.1.12.06.00	Gas transport in waste and engineered barrier system	**E/S**	–	**E/S**	**E/S**	–	–	–	–	–	–	–	–	–	–	–	–
2.1.12.07.00	Radioactive gases in waste and engineered barrier system	–	–	–	**E/S**	–	–	–	–	–	–	–	–	–	–	–	–
2.1.12.08.00	Gas explosions	–	–	–	**E/S**	–	–	–	–	–	–	–	–	–	–	–	–
2.1.13.01.00	Radiolysis	**E/A**	–	**E/U**	**E/U**	–	–	–	–	–	–	–	–	–	–	–	32
2.1.13.02.00	Radiation damage in waste and engineered barrier system	**E/S**	**E/S**	**E/S**	**E/S**	–	–	–	–	–	–	–	–	–	–	–	–
2.1.13.03.00	Mutation	–	–	–	–	–	–	–	–	–	–	–	–	–	–	**E/S**	–
2.1.14.01.00	Criticality in waste and engineered barrier system	–	–	–	**E/A**	–	–	–	–	–	–	–	–	–	–	–	74
2.1.14.02.00	Criticality *in situ*, nominal configuration, top breach	–	–	–	**E/A**	–	–	–	–	–	–	–	–	–	–	–	74
2.1.14.03.00	Criticality *in situ*, waste package internal structures degrade faster than waste form, top breach	–	–	–	**E/A**	–	–	–	–	–	–	–	–	–	–	–	74
2.1.14.04.00	Criticality *in situ*, waste package internal structures degrade at same rate as waste form, top breach	–	–	–	**E/A**	–	–	–	–	–	–	–	–	–	–	–	74
2.1.14.05.00	Criticality *in situ*, waste package internal structures degrade slower than waste form, top breach	–	–	–	**E/A**	–	–	–	–	–	–	–	–	–	–	–	74
2.1.14.06.00	Criticality *in situ*, waste form degrades in place and swells, top breach	–	–	–	**E/A**	–	–	–	–	–	–	–	–	–	–	–	74
2.1.14.07.00	Criticality *in situ*, bottom breach allows flow through waste package, fissile material collects at bottom of waste package	–	–	–	**E/A**	–	–	–	–	–	–	–	–	–	–	–	74
2.1.14.08.00	Criticality *in situ*, bottom breach allows flow through waste package, waste form degrades in place	–	–	–	**E/A**	–	–	–	–	–	–	–	–	–	–	–	74
2.1.14.09.00	Near-field criticality, fissile material deposited in near-field pond	–	–	–	**E/A**	–	–	–	–	–	–	–	–	–	–	–	74
2.1.14.10.00	Near-field criticality, fissile solution flows into drift lowpoint	–	–	–	**E/A**	–	–	–	–	–	–	–	–	–	–	–	74
2.1.14.11.00	Near-field criticality, fissile solution is adsorbed or reduced in invert	–	–	–	**E/A**	–	–	–	–	–	–	–	–	–	–	–	74
2.1.14.12.00	Near-field criticality, filtered slurry or colloidal stream collects on invert surface	–	–	–	**E/A**	–	–	–	–	–	–	–	–	–	–	–	74
2.1.14.13.00	Near-field criticality associated with colloidal deposits	–	–	–	**E/S**	–	–	–	–	–	–	–	–	–	–	–	74
2.1.14.14.00	Out-of-package criticality, fuel/magma mixture	–	–	–	–	–	–	–	–	–	–	–	–	–	–	**E/S**	–
2.2.01.01.00	Excavation and construction-related changes in the adjacent host rock	–	**I** **E/S**	–	–	–	**I** **E/A**	–	–	–	–	–	–	–	–	–	69
2.2.01.02.00	Thermal and other waste and engineered barrier system-related changes in the adjacent host rock	–	**E/A**	–	–	–	**E/A**	**E/S**	–	–	–	–	–	–	–	–	62
2.2.01.03.00	Changes in fluid saturations in the excavation disturbed zone	–	–	–	–	–	**E/S**	–	–	–	–	–	–	–	–	–	–
2.2.01.04.00	Elemental solubility in excavation disturbed zone	–	–	–	**E/S**	–	–	–	–	–	–	–	–	–	–	–	–

Table 3.2.1-2. Summary of Features, Events, and Processes Screening Argument Evaluation (continued)

Database Tracking Number	Feature, Event, and Process Name	ENG1	ENG2	ENG3	ENG4	UZ1	UZ2	UZ3	SZ1	SZ2	Direct1	Direct2	Dose1	Dose2	Dose3	Orphan	Technical Exchange
2.2.01.05.00	Radionuclide transport in excavation disturbed zone	–	–	–	–	–	–	**E/S**	–	–	–	–	–	–	–	–	–
2.2.03.01.00	Stratigraphy	–	–	–	–	**I**	**I**	**I**	**I**	**I**	**I**	–	–	–	–	–	–
2.2.03.02.00	Rock properties of host rock and other units	–	**I**	–	–	**I**	**I**	**I**	**I**	**I**	–	–	–	–	–	–	–
2.2.06.01.00	Changes in stress (due to thermal, seismic, or tectonic effects) change porosity and permeability of rock	–	**E/A**	**E/A**	–	–	**E/A**	–	**E/S**	–	–	–	–	–	–	–	66
2.2.06.02.00	Changes in stress (due to thermal, seismic, or tectonic effects) produce change in permeability of faults	–	–	–	–	–	**E/S**	–	**E/S**	–	–	–	–	–	–	–	–
2.2.06.03.00	Changes in stress (due to seismic or tectonic effects) alter perched water zones	–	–	–	–	–	**I**	–	**I**	–	–	–	–	–	–	–	–
2.2.06.04.00	Effects of subsidence	–	–	–	–	–	**E/S**	–	–	–	–	–	–	–	–	–	–
2.2.06.05.00	Salt creep	–	–	–	–	–	–	–	–	–	–	–	–	–	–	**E/S**	–
2.2.07.01.00	Locally saturated flow at bedrock/alluvium contact	–	–	–	–	**I**	–	–	–	–	–	–	–	–	–	–	–
2.2.07.02.00	Unsaturated groundwater flow in geosphere	–	–	–	–	**I**	**I**	–	–	–	–	–	–	–	–	–	–
2.2.07.03.00	Capillary rise	–	–	–	–	–	–	–	–	–	–	–	–	–	**E/RF**	–	–
2.2.07.04.00	Focusing of unsaturated flow (fingers, weeps)	–	–	–	–	**I**	**I**	–	–	–	–	–	–	–	–	–	–
2.2.07.05.00	Flow and transport in the unsaturated zone from episodic infiltration	–	–	–	–	**I**	**I** **E/A**	–	–	–	–	–	–	–	–	–	20
2.2.07.06.00	Episodic/pulse release from repository	–	–	–	**I**	–	–	–	–	–	–	–	–	–	–	–	–
2.2.07.07.00	Perched water develops	–	–	–	–	–	I	**I**	–	–	–	–	–	–	–	–	–
2.2.07.08.00	Fracture flow in the unsaturated zone	–	–	–	–	–	**I**	–	–	–	–	–	–	–	–	–	–
2.2.07.09.00	Matrix imbibition in the unsaturated zone	–	–	–	–	–	**I**	–	–	–	–	–	–	–	–	–	–
2.2.07.10.00	Condensation zone forms around drifts	–	–	**I**	–	–	I	–	–	–	–	–	–	–	–	–	–
2.2.07.11.00	Return flow from condensation cap/resaturation of dryout zone	–	–	**I**	–	–	**I**	–	–	–	–	–	–	–	–	–	–
2.2.07.12.00	Saturated groundwater flow	–	–	–	–	–	–	–	**I**	–	–	–	**I**	–	–	–	–
2.2.07.13.00	Water-conducting features in the saturated zone	–	–	–	–	–	–	–	**I**	–	–	–	**I**	–	–	–	–
2.2.07.14.00	Density effects on groundwater flow	–	–	–	–	–	–	–	**E/S**	–	–	–	**E/S**	–	–	–	–
2.2.07.15.00	Advection and dispersion	–	–	–	–	–	–	**U**	**I**	**I**	–	–	–	–	–	–	J-6
2.2.07.16.00	Dilution of radionuclides in groundwater	–	–	–	–	–	–	–	**I**	**I**	–	–	**I**	–	–	–	–
2.2.07.17.00	Diffusion in the saturated zone	–	–	–	–	–	–	–	–	**I**	–	–	**I**	–	–	–	–
2.2.07.18.00	Film flow into drifts	–	–	–	–	–	**I** **E/A**	–	–	–	–	–	–	–	–	–	USFIC-1
2.2.07.19.00	Lateral flow from Solitario Canyon fault enters potential waste emplacement drifts	–	–	–	–	–	**E/S**	–	–	–	–	–	–	–	–	–	–
2.2.08.01.00	Groundwater chemistry/composition in unsaturated zone and saturated zone	–	–	–	–	–	–	**I** **E/U**	–	**I**	–	–	–	–	**U**	–	19
2.2.08.02.00	Radionuclide transport occurs in a carrier plume in geosphere	–	–	–	–	–	–	**E/U**	**I**	**I**	–	–	**U**	**U**	**U**	–	J-8
2.2.08.03.00	Geochemical interactions in geosphere (dissolution, precipitation, weathering) and effects on radionuclide transport	–	–	–	–	–	**E/U**	**E/U**	–	**I**	–	–	–	–	–	–	J-9

Table 3.2.1-2. Summary of Features, Events, and Processes Screening Argument Evaluation (continued)

Database Tracking Number	Feature, Event, and Process Name	ENG1	ENG2	ENG3	ENG4	UZ1	UZ2	UZ3	SZ1	SZ2	Direct1	Direct2	Dose1	Dose2	Dose3	Orphan	Technical Exchange
2.2.08.04.00	Redissolution of precipitates directs more corrosive fluids to containers	–	–	**I**	–	–	**I**	–	–	–	–	–	–	–	–	–	–
2.2.08.05.00	Osmotic processes	–	–	–	–	–	–	**E/S**	–	–	–	–	–	–	–	–	–
2.2.08.06.00	Complexation in geosphere	–	–	–	–	–	–	**I** **E/U**	–	**I**	–	–	**I**	–	–	–	J-10
2.2.08.07.00	Radionuclide solubility limits in the geosphere	–	–	–	–	–	–	**E/U**	–	**I**	–	–	**I**	**U**	–	–	20, J-11
2.2.08.08.00	Matrix diffusion in geosphere	–	–	–	–	–	**I**	**I**	–	**I**	–	–	–	–	–	–	–
2.2.08.09.00	Sorption in unsaturated zone and saturated zone	–	–	–	–	–	–	**I**	–	**I**	–	–	–	–	–	–	–
2.2.08.10.00	Colloidal transport in geosphere	–	–	–	–	–	–	**I**	–	**I**	–	–	**I**	–	–	–	–
2.2.08.11.00	Distribution and release of nuclides from the geosphere	–	–	–	–	–	–	–	–	–	–	–	**U**	**I**	**U**	–	19
2.2.08.14.00	Condensation on underside of drip shield	**E/S**	–	**E/S**	–	–	**E/S**	–	–	–	–	–	–	–	–	–	–
2.2.09.01.00	Microbial activity in geosphere	–	–	–	–	–	–	**E/S**	–	**I**	–	–	–	–	–	–	–
2.2.10.01.00	Repository-induced thermal effects in geosphere	–	–	**I**	–	**I** **E/S**	**I**	**I** **E/U**	**E/S**	**E/U**	–	–	–	–	–	–	J-12
2.2.10.02.00	Thermal convection cell develops in saturated zone	–	–	–	–	–	–	–	**E/U**	–	–	–	–	–	–	–	13
2.2.10.03.00	Natural geothermal effects	–	–	–	–	–	**I/A**	–	**I/A**	**I/A**	–	–	–	–	–	–	3
2.2.10.04.00	Thermo-mechanical alteration of fractures near repository	–	**E/A**	**E/A**	–	–	**E/A**	–	–	–	–	–	–	–	–	–	70
2.2.10.05.00	Thermo-mechanical alteration of rocks above and below the repository	–	–	–	–	–	**E/U**	–	–	–	–	–	–	–	–	–	67
2.2.10.06.00	Thermo-chemical alteration (solubility, speciation, phase changes, precipitation/dissolution)	–	–	**I** **E/A**	–	–	–	**E/A** **E/U**	**E/U**	**E/U**	–	–	–	–	–	–	J-13, 9, 64
2.2.10.07.00	Thermo-chemical alteration of the Calico Hills unit	–	–	–	–	–	**E/U**	**E/U**	–	–	–	–	–	–	–	–	J-14
2.2.10.08.00	Thermo-chemical alteration of the saturated zone	–	–	–	–	–	–	–	**E/U**	**E/U**	–	–	–	–	–	–	9
2.2.10.09.00	Thermo-chemical alteration of the Topopah Spring basal vitrophyre	–	–	–	–	–	**E/U**	**E/U**	–	–	–	–	–	–	–	–	J-15
2.2.10.10.00	Two-phase bouyant flow/heat pipes	–	–	–	–	–	**I**	–	–	–	–	–	–	–	–	–	–
2.2.10.11.00	Natural air flow in unsaturated zone	–	–	–	–	**E/S**	**E/S**	–	–	–	–	–	–	–	–	–	–
2.2.10.12.00	Geosphere dryout due to waste heat	–	–	–	–	**U**	**I**	–	–	–	–	–	–	–	–	–	61
2.2.10.13.00	Density-driven groundwater flow (thermal)	–	–	–	–	–	**I**	–	**E/S**	**I** **E/A**	–	–	–	–	–	–	12
2.2.10.14.00	Mineralogic dehydration reactions	–	–	–	–	–	**E/S**	–	–	–	–	–	–	–	–	–	–
2.2.11.01.00	Naturally occurring gases in geosphere	–	–	–	–	–	–	**E/S**	**E/S**	**E/S**	–	–	–	–	–	–	–
2.2.11.02.00	Gas pressure effects	–	–	–	–	–	**E/U**	–	–	–	–	–	–	–	–	–	J-21
2.2.11.03.00	Gas transport in geosphere	–	–	–	–	–	–	**I** **E/S**	–	–	–	–	–	–	–	–	–
2.2.12.00.00	Undetected features (in geosphere)	–	–	–	–	**E/S**	**E/S**	–	**I**	**I**	–	–	**I**	–	–	–	–
2.2.14.01.00	Critical assembly forms away from repository	–	–	–	–	–	–	**E/A**	–	**E/A**	–	–	–	–	–	–	74
2.2.14.02.00	Far-field criticality, precipitation in organic reducing zone in or near water table	–	–	–	–	–	–	**E/A**	–	**E/A**	–	–	–	–	–	–	74
2.2.14.03.00	Far-field criticality, sorption on clay/zeolite in Topopah Springs basal vitrophyre	–	–	–	–	–	–	**E/A**	–	**E/A**	–	–	–	–	–	–	74

Table 3.2.1-2. Summary of Features, Events, and Processes Screening Argument Evaluation (continued)

Database Tracking Number	Feature, Event, and Process Name	ENG1	ENG2	ENG3	ENG4	UZ1	UZ2	UZ3	SZ1	SZ2	Direct1	Direct2	Dose1	Dose2	Dose3	Orphan	Technical Exchange
2.2.14.04.00	Far-field criticality, precipitation caused by hydrothermal upwell or redox front in the saturated zone	–	–	–	–	–	–	**E/A**	–	**E/A**	–	–	–	–	–	–	74
2.2.14.05.00	Far-field criticality, precipitation in perched water above Topopah Springs basal vitrophyre	–	–	–	–	–	–	**E/A**	–	**E/A**	–	–	–	–	–	–	74
2.2.14.06.00	Far-field criticality, precipitation in fractures of Topopah Springs welded rock	–	–	–	–	–	–	**E/A**	–	**E/A**	–	–	–	–	–	–	74
2.2.14.07.00	Far-field criticality, dryout produces fissile salt in a perched water basin	–	–	–	–	–	–	**E/A**	–	**E/A**	–	–	–	–	–	–	74
2.2.14.08.00	Far-field criticality associated with colloidal deposits	–	–	–	–	–	–	**E/A**	–	**E/A**	–	–	–	–	–	–	74
2.3.01.00.00	Topography and morphology	–	–	–	–	**I**	–	–	–	–	**U**	–	–	**U**	–	–	75, IA-1
2.3.02.01.00	Soil type	–	–	–	–	–	–	–	–	–	–	–	–	**I**	**I**	–	–
2.3.02.02.00	Radionuclide accumulation in soils	–	–	–	–	–	–	–	**I**	–	**I** **E/U**	–	–	**I** **E/U**	**I** **E/RF**	–	IA-1
2.3.02.03.00	Soil and sediment transport	–	–	–	–	–	–	–	–	–	–	–	–	**I** **E/U**	**I** **E/RF**	–	IA-1
2.3.04.01.00	Surface water transport and mixing	–	–	–	–	–	–	–	–	–	–	–	–	**E/S**	**E/S**	–	–
2.3.06.00.00	Marine features	–	–	–	–	–	–	–	–	–	–	–	–	**E/S**	**E/S**	–	–
2.3.09.01.00	Animal burrowing/intrusion	–	–	–	–	–	–	–	–	–	–	–	–	**E/S**	**E/S**	–	–
2.3.11.01.00	Precipitation	–	–	–	–	**I**	–	–	–	–	–	–	–	**I**	**I**	–	–
2.3.11.02.00	Surface runoff and flooding	–	–	–	–	**I**	–	–	–	–	–	–	–	**U**	**I**	–	IA-1
2.3.11.03.00	Infiltration and recharge (hydrologic and chemical effects)	–	–	**I**	–	**I**	–	–	–	–	–	–	–	**I**	–	–	–
2.3.11.04.00	Groundwater discharge to surface	–	–	–	–	–	–	–	**E/S**	**E/U**	–	–	**E/S**	**E/S**	**U**	–	10, 19
2.3.13.01.00	Biosphere characteristics	–	–	–	–	**I** **E/S**	–	–	–	–	–	–	**I** **E/S**	**I** **E/U**	**I** **E/U**	–	21
2.3.13.02.00	Biosphere transport	–	–	–	–	–	–	–	–	–	–	–	–	**I** **E/U**	**I** **E/U**	–	24, IA-1
2.3.13.03.00	Effects of repository heat on biosphere	–	–	–	–	**E/S**	–	–	–	–	–	–	–	–	–	–	–
2.4.01.00.00	Human characteristics (physiology, metabolism)	–	–	–	–	–	–	–	–	–	–	–	–	–	**I** **E/RF**	–	–
2.4.03.00.00	Diet and fluid intake	–	–	–	–	–	–	–	–	–	–	–	–	–	**I** **E/RF**	–	–
2.4.04.01.00	Human lifestyle	–	–	–	–	–	–	–	–	–	–	–	**I** **E/RF**	**I** **E/RF**	**I** **E/RF**	–	–
2.4.07.00.00	Dwellings	–	–	–	–	–	–	–	–	–	–	–	–	–	**I** **E/U**	–	25
2.4.08.00.00	Wild and natural land and water use	–	–	–	–	–	–	–	–	–	–	–	**E/RF**	**E/RF**	**E/RF**	–	–
2.4.09.01.00	Agricultural land use and irrigation	–	–	–	–	–	–	–	–	–	–	–	**I**	**I**	**I** **E/RF**	–	–
2.4.09.02.00	Animal farms and fisheries	–	–	–	–	–	–	–	–	–	–	–	**I**	–	**I**	–	–
2.4.10.00.00	Urban and industrial land and water use	–	–	–	–	–	–	–	–	–	–	–	**E/RF**	**E/RF**	**E/RF**	–	–
3.1.01.01.00	Radioactive decay and ingrowth	–	–	–	**I**	–	–	**I**	–	**I**	–	–	**U**	**I**	**U**	–	19
3.2.07.01.00	Isotopic dilution	–	–	–	–	–	–	**I** **E/S**	**I**	–	–	–	**I**	**I**	–	–	–

Table 3.2.1-2. Summary of Features, Events, and Processes Screening Argument Evaluation (continued)

Database Tracking Number	Feature, Event, and Process Name	ENG1	ENG2	ENG3	ENG4	UZ1	UZ2	UZ3	SZ1	SZ2	Direct1	Direct2	Dose1	Dose2	Dose3	Orphan	Technical Exchange
3.2.10.00.00	Atmospheric transport of contaminants	–	–	–	–	–	–	–	–	–	–	**I**	**I** **E/S**	**I** **E/S**	**I** **E/S**	–	–
3.3.01.00.00	Drinking water, foodstuffs and drugs, contaminant concentrations in	–	–	–	–	–	–	–	–	–	–	–	–	–	**I**	–	–
3.3.02.01.00	Plant uptake	–	–	–	–	–	–	–	–	–	–	–	–	–	**I** **E/RF**	–	–
3.3.02.02.00	Animal uptake	–	–	–	–	–	–	–	–	–	–	–	–	–	**I** **E/RF**	–	–
3.3.02.03.00	Bioaccumulation	–	–	–	–	–	–	–	–	–	–	–	–	–	**I**	–	–
3.3.03.01.00	Contaminated nonfood products and exposure	–	–	–	–	–	–	–	–	–	–	–	–	–	**E/RF**	–	–
3.3.04.01.00	Ingestion	–	–	–	–	–	–	–	–	–	–	–	–	–	**I**	–	–
3.3.04.02.00	Inhalation	–	–	–	–	–	–	–	–	–	–	–	–	–	**I**	–	–
3.3.04.03.00	External exposure	–	–	–	–	–	–	–	–	–	–	–	–	–	**I** **E/S**	–	–
3.3.05.01.00	Radiation doses	–	–	–	–	–	–	–	–	–	–	–	–	–	**I**	–	–
3.3.06.00.00	Radiological toxicity/effects	–	–	–	–	–	–	–	–	–	–	–	–	–	**E/RF**	–	–
3.3.06.01.00	Toxicity of mined rock	–	–	–	–	–	–	–	–	–	–	–	–	–	–	**E/S**	–
3.3.06.02.00	Sensitization to radiation	–	–	–	–	–	–	–	–	–	–	–	–	–	**E/RF**	–	–
3.3.07.00.00	Nonradiological toxicity/effects	–	–	–	–	–	–	–	–	–	–	–	–	–	**E/RF**	–	–
3.3.08.00.00	Radon and radon daughter exposure	–	–	–	–	–	–	–	–	–	–	–	–	–	**E/U**	–	26

*See Table 1.1-2 for definitions of integrated subissues.
†See Appendix B for path forward to progress from unsatisfactory (U) to satisfactory (S)

Notations That Refer to Integrated Subissues

ENG1 Degradation of Engineered Barriers
ENG2 Mechanical Disruption of Engineered Barriers
ENG3 Quantity and Chemistry of Water Contacting Waste Packages and Waste Form
ENG4 Radionuclide Release Rates and Solubility Limits
UZ1 Climate and Infiltration
UZ2 Flow Paths in the Unsaturated Zone
UZ3 Radionuclide Transport in the Saturated Zone
SZ1 Flow Paths in the Saturated Zone
SZ2 Radionuclide Transport in the Saturated Zone
Direct1 Volcanic Disruption of Waste Packages
Direct2 Airborne Transport of Radionuclides
Dose1 Representative Volume
Dose2 Redistribution of Radionuclides in Soil
Dose3 Biosphere Characteristics

Symbols

S Satisfactory
U Initially evaluated as Unsatisfactory (items already discussed with DOE, and agreements have been produced to address concern)
I Included
E Excluded
A Existing DOE/NRC Technical Exchange Agreements are related to screening argument
RF Screening argument based on 10 CFR Part 63
QA Screening based on not yet implemented quality assurance procedures; acceptance is pending elaboration of such procedures

objectives consistent with those cited in the screening arguments. Finally, the symbol A identifies those entries for which screening arguments related to or dependent on work needed to satisfy agreements reached at DOE and NRC key technical issue technical exchanges. Appendix B contains details on why some screening arguments were initially classified as unsatisfactory. The comments are listed in ascending order according to database tracking numbers with the exception of the first entries, which address general comments applicable to multiple features, events, and processes. All comments in Appendix B have been discussed with DOE at the May 15–17[18] and August 6–10,[19] 2001, DOE and NRC Technical Exchanges and Management Meetings on Total System Performance Assessment and Integration, and at the September 5,[20] 2001, Technical Exchange and Management Meeting on Igneous Activity. Tracking numbers assigned to the NRC comments at these technical exchanges and the agreed-on paths forward are also included in Appendix B.

In general, DOE agreed to clarify screening arguments or provide technical bases supporting screening decisions. For those features, events, and processes related to existing DOE and NRC agreements, DOE agreed to revise the screening arguments in pertinent analysis and model reports after completion of the work needed to satisfy the agreements. DOE also agreed to expand the scope of analyses and model reports addressing features, events, and processes, to contain relevant items not currently in their scope, and clarify the definition of some features, events, and processes. Details of the concerns and agreed-on paths forward are contained in Appendix B. The agreements reached between DOE and NRC are listed in Section 3.2.1.5.

3.2.1.4.3 Formation of Scenario Classes Using the Reduced Set of Events Is Adequate

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.2.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the adequacy of the formation of scenario classes using the reduced set of events.

DOE indicated that included features, events, and processes are combined in two possible scenario classes (disruptive and nominal), and both classes would be represented in the total

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[18]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration—Features, Events, and Processes (May 15–17, 2001)." Letter (May 30) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[19]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[20]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

system performance assessment[21] (CRWMS M&O, 2000a). The nominal scenario class includes all features, events, and processes assumed to occur during 10,000 years, and the disruptive scenario class encompasses features, events, and processes related to igneous activity (CRWMS M&O, 2000a). This approach to scenario class formation is appropriate. Adequate formation of scenario classes depends in part on a complete identification of features, events, and processes, development of appropriate screening rationale, and screening decisions for features, events, and processes (i.e., either to be included or not into the performance assessment). For example, features, events, and processes exist for which a screening decision could impact the identification of scenario classes such as 2.1.07.02.00 (Mechanical Degradation or Collapse of Drift), given potential implications of drift collapse on temperature, chemistry, seepage rates, and drip shield performance. Nonetheless, the information provided by DOE on its current approach to form scenario classes is sufficient for NRC to make a regulatory decision at the time of future license application.

3.2.1.4.4 Screening of Scenario Classes Is Appropriate

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.2.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the appropriateness of the screening of scenario classes.

DOE indicated that both the disruptive and nominal scenario classes are represented in the total system performance assessment[22] (CRWMS M&O, 2000a,b). Thus, none of the scenario classes identified so far will be screened out from the performance assessment.

3.2.1.5 Status and Path Forward

Table 3.2.1-3 provides related DOE and NRC agreements pertaining to the Scenario Analysis, as well as the status of the associated key technical issue subissues. Note that the status as well as the detailed agreements pertaining to all the key technical issue subissues are provided in Table 1.1-3 and Appendix A. Details on the agreed-on paths forward to address NRC questions on the screening of features, events, and processes discussed at the May 15–17[23] and 6–10,[24] 2001, DOE and NRC Technical Exchanges and Management Meetings, are presented in Appendix B.

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[21]Swift, P. "TSPA-SR Features, Events, and Processes Approach: Process and Methodology." *Presentation at the DOE and NRC Technical Exchange on Total System Performance Assessment (TSPA) for Yucca Mountain, San Antonio, TX*. *June 6–7, 2000*. San Antonio, Texas. 2000.

[22]Ibid.

[23]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration—Features, Events, and Processes (May 15–17, 2001)." Letter (May 30) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[24]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

The NRC staff have confidence the DOE proposed approach, together with DOE agreements to provide NRC with additional information (through specified testing, analyses, and the like), acceptably addresses NRC questions so that no information beyond that provided, or agreed to, will likely be required at the time of an initial license application.

Table 3.2.1-3. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreements*
Container Life and Source Term	Subissue 3—Rate at Which Radionuclides in Spent Nuclear Fuel Are Released from the Engineered Barrier Subsystem through the Oxidation and Dissolution of Spent Fuel	Closed-Pending	CLST.3.01 CLST.3.04
	Subissue 4—Rate at Which Radionuclides in High-Level Waste Glass are Leached and Released from the Engineered Barrier Subsystem	Closed-Pending	CLST.4.01 CLST.4.04
	Subissue 5—Effect of In-Package Criticality on Waste Package and Engineered Barrier Subsystem Performance	Closed-Pending	CLST.5.01 CLST.5.02 CLST.5.03 CLST.5.06 CLST.5.07
Evolution of the Near-Field Environment	Subissue 1—Effects of Coupled Thermal-Hydrologic-Chemical Processes on Seepage and Flow	Closed-Pending	ENFE.1.01 ENFE.1.02 ENFE.1.06
	Subissue 2—Effects of Coupled Thermal-Hydrologic-Chemical Processes on Waste Package Chemical Environment	Closed-Pending	ENFE.2.01 ENFE.2.02 ENFE.2.03
	Subissue 4—Effects of Coupled Thermal-Hydrologic-Chemical Processes on Radionuclide Transport through Engineered and Natural Barriers	Closed-Pending	ENFE.4.03 through ENFE.4.08
	Subissue 5—Effects of Coupled Thermal-Hydrologic-Chemical Processes on Potential Nuclear Criticality in the Near Field	Closed-Pending	ENFE.5.01 ENFE.5.02
Igneous Activity	Subissue 1—Probability of Future Igneous Activity	Closed-Pending	IA.1.01 IA.1.02

Table 3.2.1-3. Related Key Technical Issue Subissues and Agreements (continued)			
Key Technical Issue	**Subissue**	**Status**	**Related Agreements***
Repository Design and Thermal-Mechanical Effects	Subissue 3—Thermal-Mechanical Effects on Underground Facility Design and Performance	Closed-Pending	RDTME.3.19
Radionuclide Transport	Subissue 1—Radionuclide Transport through Porous Rock	Closed-Pending	RT.1.03
	Subissue 2—Radionuclide Transport through Alluvium	Closed-Pending	RT.2.02 RT.2.10 RT.2.11
	Subissue 4—Nuclear Criticality in the Far Field	Closed-Pending	RT.4.01 RT.4.02
Structural Deformation and Seismicity	Subissue 1—Faulting	Closed-Pending	SDS.1.01
	Subissue 2—Seismicity	Closed-Pending	SDS.2.02
Thermal Effects on Flow	Subissue 1—Features, Events, and Processes Related to Thermal Effects on Flow	Closed-Pending	TEF.1.01 TEF.1.02
Unsaturated and Saturated Flow Under Isothermal Conditions	Subissue 5—Saturated Zone Ambient Flow Conditions and Dilution Processes	Closed-Pending	USFIC.5.14
Total System Performance Assessment and Integration	Subissue 1—System Description and Demonstration of Multiple Barriers	Close-Pending	TSPAI.1.01 TSPAI.1.02
	Subissue 2—Scenario Analysis and Event Probability	Closed-Pending	TSPAI.2.01 through TSPAI2.07
	Subissue 3—Model Abstraction	Closed-Pending	TSPAI.3.01 through TSPAI 3.42
	Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards	Closed-Pending	TSPAI.4.01 through TSPAI.4.07
*Related DOE and NRC agreements are associated with one or all four generic acceptance criteria.			

3.2.1.6 References

CRWMS M&O. "Total System Performance Assessment for the Site Recommendation." TDR–WIS–PA–000001. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000a.

———. "Features, Events and Processes: Screening for Disruptive Events." ANL–WIS–MD–000005. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000b.

———. "Features, Events, and Processes in SZ Flow and Transport." ANL–NBS–MD–000002. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000c.

———. "Features, Events, and Processes: System-Level and Criticality." ANL–WIS–MD–000019. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000d.

———. "The Development of Information Catalogued in Rev 00 of the YMP FEP Database." TDR–WIS–MD–000003. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2001a.

———. "Yucca Mountain FEP Database." TDR–WIS–MD–000003. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2001b.

———. "FEPs Screening of Processes and Issues in Drip Shield and Waste Package Degradation." ANL–EBS–PA–000002. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2001c.

———. "Features, Events, and Processes in UZ Flow and Transport." ANL–NBS–MD–000001. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2001d.

———. "Features, Events, and Processes in Thermal Hydrology and Coupled Processes." ANL–NBS–MD–000004. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2001e.

———. "EBS FEPs/Degradation Modes Abstraction." ANL–WIS–PA–000002. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2001f.

NRC. "Issue Resolution Status Report. Key Technical Issue: Total System Performance Assessment and Integration." Revision 3. Washington, DC: NRC. 2000.

———. NUREG–1804, "Yucca Mountain Review Plan—Draft Report for Comment." Revision 2. Washington, DC: NRC. March 2002.

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

3.2.2 Identification of Events with Probabilities Greater Than 10^{-8} Per Year

3.2.2.1 Description of Issue

The Identification of Events with Probabilities Greater Than 10^{-8} Per Year is necessary to ensure that all significant events have been included in demonstrating compliance with the postclosure performance objective at 10 CFR 60.113. (See requirements for performance assessment at 10 CFR 60.114.) The identification of events with probabilities greater than 10^{-8} per year includes the following parts: (i) appropriate definition of events and event sequences, (ii) appropriate determination of the annual probability of each event with sufficient technical basis, (iii) appropriate use of conceptual models to determine the probability of events, (iv) use of appropriate parameters to define the probability of events, and (v) appropriate consideration of uncertainty in models and parameters used to calculate the probability of events.

This section provides a review of the methodologies used by DOE to identify the events that have a probability of occurrence at the Yucca Mountain repository greater than 10^{-8} per year in its Total System Performance Assessment. The DOE description and technical basis for the Identification of Events with Probabilities Greater Than 10^{-8} Per Year are documented in CRWMS M&O (2000a), five supporting analysis and model reports, and a calculational package (CRWMS M&O, 2000b). Portions of additional analysis and model reports are reviewed because they contain data or analyses that support the proposed Total System Performance Assessment abstractions.

3.2.2.2 Relationship to Key Technical Issue Subissues

Event classes identified as potentially significant for the proposed repository system at Yucca Mountain include:

- Igneous Activity
- Faulting
- Seismicity
- Nuclear Criticality

According to 10 CFR Part 63, the disruption of the repository because of human intrusion will be analyzed using a stylized scenario, and the probability of this event class does not have to be determined. The technical basis for the assignment of probability values to these event classes has been previously captured within the framework of the following key technical issue subissues:

- Igneous Activity: Subissue 1—Probability of Igneous Activity (NRC, 1999a)

- Structural Deformation and Seismicity: Subissue 1—Faulting (NRC, 1999b)

- Structural Deformation and Seismicity: Subissue 2—Seismicity (NRC, 1999b)

- Container Life and Source Term: Subissue 5—The Effect of In-Package Criticality on Waste Package and Engineered Barrier Subsystem Performance (NRC, 2001)

- Evolution of the Near-Field Environment: Subissue 4—Effects of Coupled Thermal-Hydrologic-Chemical Processes on Radionuclide Transport Through Engineered and Natural Barriers (NRC, 2000a)

- Radionuclide Transport: Subissue 4—Nuclear Criticality in the Far Field (NRC, 2000b)

- Total System Performance Assessment and Integration: Subissue 2—Scenario Analysis and Event Probability (NRC, 2000c)

- Total System Performance Assessment and Integration: Subissue 3—Model Abstraction (NRC, 2000c)

The key technical issue subissues formed the bases for the previous version of the issue resolution status reports and also were the bases for technical exchanges with DOE where agreements were reached about what additional information DOE needed to provide to resolve the subissue. The resolution status of the Scenario Analysis and Event Probability Subissue is based on the resolution status of the contributing key technical issue subissues. The subsequent sections incorporate applicable portions of these key technical issue subissues. No effort was made, however, to explicitly identify each subissue.

3.2.2.3 Importance to Postclosure Performance

One aspect of risk informing the NRC review was to determine how the Identification of Events with Probabilities Greater Than 10^{-8} Per Year is related to the DOE repository safety strategy. The probability of igneous activity must be known to accurately estimate the long-term risk, as recognized in CRWMS M&O (2000c) for the proposed Yucca Mountain site. CRWMS M&O (2000c) identifies the probability of igneous intrusion as one of the eight principal factors for the Yucca Mountain repository system. The occurrence of seismic activity or faulting could result in failure of the waste package or drip shield. Performance of the waste package and performance of the drip shield/drift invert system are also identified as principal factors for the Yucca Mountain repository system (CRWMS M&O, 2000c).

The Identification of Events with Probabilities Greater Than 10^{-8} Per Year is important because this identification determines which events are needed to be considered further in the performance assessment. 10 CFR 63.114(d) requires that the performance assessment for Yucca Mountain must consider all events with at least 1 chance in 10,000 of occurring during the 10,000-year compliance period for the repository, which corresponds to an annual probability of 10^{-8} per year for events that have probabilities of occurrence that are independent of time. Events that are less likely than this do not need to be considered in the performance assessment. Events that are at least this likely must either be modeled within the performance assessment or be shown to not significantly affect the magnitude and time of the resulting radiological exposures to the reasonably maximally exposed individual or radionuclide releases to the accessible environment.

Additionally, Identification of Events with Probabilities Greater Than 10^{-8} Per Year is important for appropriately comparing the consequences of disruptive events against the 0.15-mSv/yr [15-mrem/yr] all-pathways dose standard in 10 CFR Part 63. 10 CFR 63.2 indicates in the definition of performance assessment that estimates of dose from all significant events and processes should be weighted by their probability of occurrence when included in the calculation of dose to the reasonably maximally exposed individual. Therefore, the probability of occurrence of a disruptive event is an important factor in the determination of whether the repository system will meet the limits specified in 10 CFR Part 63.

3.2.2.4 Technical Basis

NRC developed a plan (2002) consistent with the acceptance criteria and review methods found in previous issue resolution status reports. A review of DOE approaches for Identification of Events with Probabilities Greater Than 10^{-8} Per Year is provided in the following subsections. The review will be divided into four subsections: Igneous Activity, Seismicity, Faulting, and Criticality. Each subsection is organized according to the acceptance criteria in the Yucca Mountain Review Plan: (i) Events Are Adequately Defined, (ii) Probability Estimates for Future Events Are Supported by Appropriate Technical Basis , (iii) Probability Model Support Is Adequate, (iv) Probability Model Parameters Have Been Adequately Established, and (v) Uncertainty in Event Probability Is Adequately Evaluated.

3.2.2.4.1 Igneous Activity

The probability of igneous activity affecting the repository system was discussed and reached closed-pending status at a technical exchange held in August 2000.[1] NRC expects to receive all information required to complete the agreements by fiscal year 2003.

3.2.2.4.1.1 Events Are Adequately Defined

Overall, the current information is sufficient to conclude that the necessary information will be available to assess the probability of igneous activity affecting the repository system at the time of a potential license application.

Repository performance considerations require that the probability of volcanic disruption is calculated discretely from the probability of intrusive disruption because the effects on repository performance are significantly different for extrusive and intrusive processes. A volcanic igneous event that penetrates the repository has the potential to entrain, fragment, and transport radioactive material into the accessible environment. In contrast, an intrusive igneous event that penetrates the repository would produce thermal, mechanical, and chemical loads on engineered systems, which could affect waste-package degradation. Radioactive release associated with intrusive igneous events is through hydrologic flow and transport rather than through direct transport by volcanic processes. Therefore, probability calculations need to

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[1]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (August 29–31, 2000)." Letter (October 23) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

distinguish between volcanic and intrusive igneous events to appropriately determine the contribution of each event to the probability weighted dose.

DOE documented the approach and technical basis for the definition of an igneous event in CRWMS M&O (2000a) and supporting analysis and model reports. CRWMS M&O (2000f) summarizes the technical basis for the definition of an igneous event. DOE estimate of the probability of an igneous event affecting the repository is based on the results of an expert elicitation to determine the probability of igneous activity at Yucca Mountain (CRWMS M&O, 1996). DOE defined a volcanic event as a point in space representing a volcano and an associated intrusive dike having length, azimuth, and location extending from the point event (CRWMS M&O, 2000f). Although the probabilistic volcanic hazard assessment assumed volcanic events to have both an extrusive (eruptive volcano) and intrusive component (dike), the output of the probabilistic volcanic hazard assessment was the annual frequency of intersection of the repository by only an intrusive basaltic dike. The probability of a volcanic eruption, conditional on dike intersection through the repository, likely would be lower using the probabilistic volcanic hazard assessment methodology. The DOE probabilistic volcanic hazard assessment did not calculate the conditional probability that a dike intersecting the repository footprint would result in an extrusive volcanic eruption through the repository. Models for the distribution of vents along a dike (based on the DOE probabilistic volcanic hazard assessment expert output and some observed vent spacings in the Yucca Mountain region) indicate that the probabilistic volcanic hazard assessment-derived eruption probability is always less than the dike intersection probability by a factor of approximately two (CRWMS M&O, 2000f).

The distinction between intrusive and extrusive igneous events is sufficiently clear in the DOE documentation to allow NRC to have enough information at the time of licensing to make a regulatory decision in this area.

3.2.2.4.1.2 Probability Estimates for Future Events Are Supported by Appropriate Technical Basis

Overall, the current information, along with agreements reached between DOE and NRC, is sufficient to conclude that the necessary information will be available to assess the probability of igneous activity affecting the repository system at the time of a potential license application.

Previous studies of volcanism in the Yucca Mountain region and elsewhere cumulatively indicate that models describing the recurrence rate or probability of basaltic volcanism should reflect the clustered nature of basaltic volcanism and shifts in the locus of basaltic volcanism through time. Models also should be amenable to comparison with basic geological data, such as fault patterns and neotectonic stress information, that affect vent distributions on a comparatively more detailed scale. The models used to estimate future igneous activity in the Yucca Mountain region should either explicitly account for the following or obtain bounding estimates:

- Shifts in the locus of volcanic activity through time
- Vent clusters
- Vent alignments and correlation of vents and faults

Data from other basaltic volcanic fields may be used to test the models. The nature of these spatial patterns in the Yucca Mountain region and how these compare with spatial patterns in cinder cone volcanism observed in other basaltic volcanic fields are reviewed in this section.

DOE documented the approach and technical basis for calculating the probability of an igneous event affecting the repository system in CRWMS M&O (2000a) and supporting analysis and model reports. The analysis and model report (CRWMS M&O, 2000f) summarizes the technical basis for the estimate of the probability of igneous activity affecting the Yucca Mountain repository. The results of the probabilistic volcanic hazard assessment (CRWMS M&O, 1996) form the basis of the DOE estimate of the probability of igneous activity affecting the repository system. For the probabilistic volcanic hazard assessment, an expert panel was convened in 1995 to review pertinent data relating to volcanism at Yucca Mountain and, based on these data, to quantify both the annual probability and associated uncertainty of an intrusive volcanic event intersecting a potential repository at Yucca Mountain. The experts reviewed two decades of data collected by volcanologists who conducted studies to quantify the probability that a future volcanic eruption would disrupt the potential repository. The mean intersection probability based on the results of the probabilistic volcanic hazard assessment was slightly greater than 10^{-8} per year (CRWMS M&O 2000f).

Agreement exists between the models and observed data on the basic patterns of basaltic volcanism in the Yucca Mountain region. These patterns include changes in the locus of volcanism with time, recurring volcanic activity within vent clusters, formation of vent alignments, and structural controls on the locations of volcanoes. Each of these patterns in vent distribution has an important impact on volcanic probability models and is considered in many probability models.

All current probability estimates for future igneous activity at the proposed repository site are based on past patterns of igneous activity in the Yucca Mountain region. Some parameter values or ranges used in these probability models, however, are dependent on definitions of the spatial or temporal extent of the Yucca Mountain region igneous system. Ongoing work suggests Crater Flat Basin basalts since about 12 million years may have a common petrogenesis, whereas 7–12-million years Yucca Mountain region basalt petrogenesis may be strongly influenced by silicic caldera-forming processes. Thus, Miocene basalt in the Crater Flat basin provides relevant information for risk assessments not included in current DOE models. Additionally, there are concerns about how the probabilistic volcanic hazard assessment was conducted. DOE selected only a limited range of experts for the probabilistic volcanic hazard assessment, using an internal nomination rather than a self-selection process. Potential biases or conflicts of interest among the experts are not documented. Modifications to initial elicitation reports also are not documented. These items do not follow the guidance in NUREG–1563 (NRC, 1996) for conducting an expert elicitation, and, therefore, make it difficult to evaluate the conclusions of the probabilistic volcanic hazard assessment elicitation (CRWMS M&O, 1996). Therefore, there is concern that the DOE probability model could result in an inaccurate estimate of the probability of igneous activity affecting the repository system. NRC staff independent assessments of the probability of igneous activity affecting the Yucca Mountain repository estimate it to be approximately 10^{-7} per year for both extrusive and intrusive volcanism (Hill and Connor, 2000). Therefore, DOE agreed[2] to include, in the Total System Performance Assessment–Site Recommendation and any license application, the results of a single-point sensitivity analysis for extrusive and intrusive igneous processes affecting the repository system at a probability of 10^{-7} per year. The NRC staff will consider this sensitivity analysis in its review.

3.2.2.4.1.3 Probability Model Support Is Adequate

Overall, the current information, along with agreements between DOE and NRC, is sufficient to conclude that the necessary information will be available to assess the probability of igneous activity affecting the repository system at the time of a potential license application.

DOE documented the support for the models predicting the probability of an igneous event affecting the repository system in CRWMS M&O (2000a) and supporting analysis and model reports. The CRWMS M&O (2000f) analysis and model report summarizes the technical basis for the estimate of the probability of igneous activity affecting the Yucca Mountain repository. The results of the probabilistic volcanic hazard assessment (CRWMS M&O, 1996) form the basis of the DOE estimate of the probability of igneous activity affecting the repository system. The conceptual model of volcanism, including how and where magmas form and what processes control the timing and location of magma ascent through the crust to form volcanoes, has a fundamental impact on how probability models are formulated and the consequent results of probability models. The probabilistic volcanic hazard assessment experts distinguished between deep (mantle source) and shallow (upper crustal structure and stress field) processes when considering different scales (regional and local) of spatial control on volcanism. Many probabilistic volcanic hazard assessment models restricted the areas of greatest likelihood for future volcanic activity to the areas where previous volcanism has occurred. DOE also justifies the probabilistic volcanic hazard assessment volcanic source-zone definitions by relating these zones to areas within the crater flat basin that have undergone the greatest amount of shallow crustal extension (e.g., Fridrich, et al., 1999, Figure 5; CRWMS M&O, 2000f, Figures 9a and 9b).

Although some volcanic source zones in CRWMS M&O (1996, 2000f) are supported by tectonic models, many other zones and other tectonic models are not supported. Few tectonic models or data are cited in CRWMS M&O (1996) for zone definitions. Currently available geophysical data (gravity, aeromagnetic, and seismic) do not support zone definitions used in the probabilistic volcanic hazard assessment (CRWMS M&O, 1996, 2000f). DOE does not seem to have established the validity of the probabilistic volcanic hazard assessment source-zone modeling approach. Additionally, there is an inconsistency between the probabilistic volcanic hazard assessment and the current DOE probability models. Probabilistic volcanic hazard assessment volcanic source zones clearly were defined on timing and location of past volcanism within the source zone. A new event center (i.e., volcano) forms only in the source zone, with only a subsurface intrusion potentially extending out of the zone and intersecting the

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[2]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

repository. The model in CRWMS M&O (2000f), however, has new volcanoes forming randomly along the intrusion, often outside the predefined volcanic source zone. By probabilistic volcanic hazard assessment definition, new volcanoes should occur only within the source zone at recurrences defined by past patterns of volcanic activity within that zone. If volcanoes can form outside the source zone as indicated in CRWMS M&O (2000f), the source zones must be expanded to encompass the location of future volcanism. The frequency of dike intersections would then increase using the expanded zones, as shorter, more abundant dikes would intersect the proposed repository location. DOE needs to demonstrate that its preferred approach can reasonably forecast the timing and location of future igneous events (cf., Condit and Connor, 1996). Therefore, there is concern that the probability model used could result in an inaccurate estimate of the probability of igneous activity affecting the repository system. NRC staff independent assessments of the probability of igneous activity affecting the Yucca Mountain repository estimate it to be approximately 10^{-7} per year for both extrusive and intrusive volcanism (Hill and Connor, 2000). Therefore, DOE agreed[3] to include, in the Total System Performance Assessment–Site Recommendation and any license application, the results of a single-point sensitivity analysis for extrusive and intrusive igneous processes affecting the repository system at a probability of 10^{-7} per year. The NRC staff will consider this sensitivity analysis in its review.

3.2.2.4.1.4 Probability Model Parameters Have Been Adequately Established

Overall, the current information, along with agreements reached between DOE and NRC, is sufficient to conclude that the necessary information will be available to assess the probability of igneous activity affecting the repository system at the time of a potential license application.

DOE documented the technical basis for the parameters supporting the models that predict the probability of an igneous event affecting the repository system in CRWMS M&O (2000a) and supporting analysis and model reports. The analysis and model report in CRWMS M&O (2000f) summarizes the technical basis for the probability model parameters. The results of the probabilistic volcanic hazard assessment (CRWMS M&O, 1996) form the basis of the DOE estimate of the probability of igneous activity affecting the repository system.

NRC staff have concerns about the selective use of data from the probabilistic volcanic hazard assessment (CRWMS M&O, 1996) that occurs in CRWMS M&O (2000f). For example, vent spacing (CRWMS M&O, 2000f, Section 6.5.2.2) only uses data from the 1-million years Crater Flat and 0.3-million years Sleeping Butte volcanoes, but ignores relevant information from the 3.7-million years Crater Flat, buried anomalies in Amargosa Desert, Paiute Ridge Intrusive Complex, and other features used by DOE to support igneous process models for the Yucca Mountain region. There also is an assumption that a relationship exists in the probabilistic volcanic hazard assessment (CRWMS M&O, 1996) between the number of events and the number of dikes. The probabilistic volcanic hazard assessment (CRWMS M&O, 1996) considered these as independent parameters. Thus, there is concern that the parameters used

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[3]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

in the probability model could result in an inaccurate estimate of the probability of igneous activity affecting the repository system. NRC staff independent assessments of the probability of igneous activity affecting the Yucca Mountain repository estimate it to be approximately 10^{-7} per year for both extrusive and intrusive volcanism (Hill and Connor, 2000). Therefore, DOE agreed[4] to include, in the Total System Performance Assessment–Site Recommendation and any license application, the results of a single-point sensitivity analysis for extrusive and intrusive igneous processes affecting the repository system at a probability of 10^{-7} per year. The NRC staff will consider this sensitivity analysis in its review.

3.2.2.4.1.5 Uncertainty in Event Probability Is Adequately Evaluated

Overall, the current information, along with agreements reached between DOE and NRC, is sufficient to conclude that the necessary information will be available to assess the probability of igneous activity affecting the repository system at the time of a potential license application.

DOE documented the technical basis for the uncertainty in the probability of an igneous event affecting the repository system in CRWMS M&O (2000a) and supporting analysis and model reports. CRWMS M&O (2000f) summarizes the technical basis for the uncertainty in the estimate of the probability of igneous activity affecting the Yucca Mountain repository. The results of the probabilistic volcanic hazard assessment (CRWMS M&O, 1996) form the basis of the DOE estimate of the probability of igneous activity affecting the repository system. There are no generally accepted methodologies for calculating the probabilities of future igneous activity in distributed volcanic fields for periods of 10,000 years. In addition, more than one conceptual model can be applied to this problem, resulting in a wide range of probability values. DOE is using expert elicitation (CRWMS M&O, 1996) to construct a range of probability models, estimate uncertainties in model results caused by reasonable variations in model parameters, and calculate a probability distribution for use in performance assessment models.

The use of an expert elicitation conducted following NRC guidance in NUREG–1563 (NRC, 1996) is an acceptable methodology to determine the uncertainty in the probability of an igneous event. NRC staff have some concerns about how the DOE expert elicitation was conducted and documented, as discussed in Section 3.2.2.4.1.2. Additionally, NRC has concerns that uncertainty in the probability of igneous activity caused by undetected igneous events in the Yucca Mountain region could significantly affect the DOE calculation of the probability of igneous activity affecting the repository system. Therefore, DOE agreed[5] to evaluate new aeromagnetic data for potential buried igneous features and the effect on the probability estimate.

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[4]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[5]Ibid.

3.2.2.4.2 Faulting

The probability of a faulting event affecting the repository system was discussed at a Technical Exchange held in October 2000.[6] The Structural Deformation and Seismicity Subissue 1, Faulting, reached closed-pending status at this technical exchange. NRC expects to receive all information required to complete the agreements by fiscal year 2003.

3.2.2.4.2.1 Events Are Adequately Defined

Overall, the current information is sufficient to conclude that the necessary information will be available to assess the probability of faulting affecting the repository system at the time of a potential license application.

The approach and technical basis for defining faulting events are contained in CRWMS M&O (2000a). DOE divides faulting events into separate features, events, and processes based on their potential consequence. DOE considers that faulting events could potentially alter groundwater flow around and below the drift or could potentially disrupt engineered barriers in the repository system. When considering the effects of faulting on groundwater flow, DOE defined an event as a fault displacement event that could either change fracture properties throughout the unsaturated zone flow model domain or change the fracture properties specifically within fault zones. These two end-member cases relate to the mechanical strain either distributed throughout the strata bounded by the faults or localized to the individual fault zones. When considering the effects of faulting on engineered barriers, DOE defined an event as the failure of a structure, system, or component to perform its functional goal because of fault displacement loading. DOE analyses consider the reactivation of existing faults and the formation of new faults as separate types of events with different probabilities and consequences.

The definition of events is sufficiently clear in the DOE documentation to allow NRC to have enough information at the time of licensing to make a regulatory decision in this area.

3.2.2.4.2.2 Probability Estimates for Future Events Are Supported by Appropriate Technical Basis

Overall, the current information, along with agreements reached between DOE and NRC, is sufficient to conclude that the necessary information will be available to assess the probability of faulting affecting the repository system at the time of a potential license application.

The approach and technical basis for defining the probability of faulting affecting the repository system are contained in CRWMS M&O (2000a) and the analysis and model reports in CRWMS M&O (2000e,g,h,i). The basis for the estimates of the probability of faulting events

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[6]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Structural Deformation and Seismicity (October 11–12, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

affecting the repository system is the result of an expert elicitation documented in the U.S. Geological Survey (1998). The probabilistic seismic hazard assessment used data collected on faulting characteristics at Yucca Mountain and in the Basin and Range province during past earthquakes to develop a displacement hazard curve. Principal and secondary (or distributed) faulting were considered. Principal faulting refers to displacement along the main fault zone responsible for the release of seismic energy (i.e., an earthquake) (dePolo, et al., 1991). At Yucca Mountain, principal faulting is assumed to occur only along principal faults, mainly block-bounding faults like the Solitario Canyon and Paintbrush Canyon faults. In contrast, secondary or distributed faulting is defined as rupture of smaller faults, such as the Ghost Dance fault, that occurs in response to the rupture in the vicinity of the principal fault (dePolo, et al., 1991). These two subsets of faults are not mutually exclusive. Faults capable of principal rupture can also undergo secondary faulting in response to faulting on another principal fault. Because principal and secondary faults pose a potential risk to repository performance, DOE considered both types. NRC (1999) provides a review of the methodology used by the DOE expert elicitation to develop an appropriate probabilistic fault displacement hazard assessment. This curve plots the frequency of exceeding a fault displacement value. The probabilistic seismic hazard assessment concluded that mean displacements at all locations within the repository system, except for Bow Ridge and Solitario Canyon faults, are 0.1 cm [30.039 in.] or less at the 10^{-5} annual exceedance probability. The mean displacements for the Bow Ridge and Solitario Canyon faults are 8 and 32 cm [3.15 and 12.6 in], respectively, at the 10^{-5} exceedance probability. DOE extrapolated these results and used the median value predicted by the experts to provide estimates of the displacement at the 10^{-8} annual exceedance probability.

DOE concluded faulting affecting groundwater flow is credible because the fault displacement could change the properties of the fractures in the unsaturated zone rock. DOE has developed criteria for fault setback distances for the design of the repository, which will be applied to existing faults with known or suspected Quaternary-age displacements. This setback distance is designed to mitigate the shear stresses induced on the waste packages and drip shields. The probabilistic seismic hazard assessment concluded that the mean displacement at a 10^{-8} annual exceedance probability for small faults and shear fractures in the repository system is less than 1 m [39.4 in.]. This displacement roughly corresponds to the maximum measured Quaternary per-event displacement on the Solitario Canyon fault. Based on the gap between the drip shields and the drift walls, DOE concluded this displacement could not cause the failure of the waste package nor the drip shield. The probabilistic seismic hazard assessment also concluded that the mean annual probability of a shear fracture developing in intact rock is less than 10^{-8}. Therefore, DOE concluded that all aspects of faulting could be screened based on low probability except for the effects of faulting on groundwater flow.

Staff reviewed the data, conceptual models, and assumptions developed by DOE in the probabilistic seismic hazard assessment (U.S. Geological Survey, 1998) and found that DOE adequately evaluated the nature and amount of faulting and the appropriate range of both principal and secondary faulting hazard sources within the repository block. In addition, DOE adequately determined fault geometry applicable to development of the probabilistic fault displacement hazard assessment. Given present knowledge, the DOE interpretations of faulting from surficial and underground mapping, as presented in the DOE probabilistic seismic hazard assessment (U.S. Geological Survey, 1998), are geologically consistent and reasonable. The experts adequately noted faults as primary or secondary, because these classifications pertain to the probabilistic fault displacement hazard assessment. Faulting characteristics identified subsequently or for which new data are developed should be evaluated or reevaluated, respectively. Variation of fault orientation data is within acceptable limits for normal geologic work. Staff disagree, however, with the statistic used to combine the fault displacement hazard curves from the different experts in the probabilistic seismic hazard assessment. DOE uses the median value of the curves of the experts as the statistic of interest, whereas NRC staff believe that the mean is the more appropriate measure. Using the mean value of the curves would lead to a larger displacement being predicted at the 10^{-8} annual probability level. DOE agreed[7] to provide technical justification for use of median values or another statistical measure, such as the mean, or evaluate and implement an alternative approach.

3.2.2.4.2.3 Probability Model Support Is Adequate

Overall, the current information is sufficient to conclude that the necessary information will be available to assess the probability of faulting affecting the repository system at the time of a potential license application.

The support for the probability model is contained in the CRWMS M&O (2000a) and the analysis and model reports (CRWMS M&O, 2000e,h,i). The basis for the probability of faulting affecting the repository system is the result of probabilistic seismic hazard assessment. The experts in the probabilistic seismic hazard assessment appropriately considered primary and secondary faulting when defining fault displacement hazard curves. The level of ground motion predicted by the probabilistic seismic hazard assessment has been compared to tectonically and seismically active sites elsewhere in the Basin and Range Province (Wong and Olig, 1998) and found to be lower than other more seismically active areas in the Basin and Range province, such as along the Wasatch fault in north central Utah.

Staff review indicates that DOE adequately evaluated the nature and amount of faulting and the appropriate range of both principal and secondary faulting hazard sources within the repository block. In addition, DOE adequately determined fault geometry applicable to development of the probabilistic fault displacement hazard assessment. Given present knowledge, the DOE interpretations of faulting from surficial and underground mapping, as presented in U.S. Geological Survey (1998), are geologically consistent and reasonable.

3.2.2.4.2.4 Probability Model Parameters Have Been Adequately Established

Overall, the current information is sufficient to conclude that the necessary information will be available to assess the probability of faulting affecting the repository system at the time of a potential license application.

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[7]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Structural Deformation and Seismicity (October 11–12, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

The technical basis for the parameters used in the probability model is contained in CRWMS M&O (2000a) and the (CRWMS M&O, 2000i) analysis and model report. The basis for the probability model is the result of the probabilistic seismic hazard assessment. The assessment of seismic hazards at Yucca Mountain in the probabilistic seismic hazard assessment relied on the results of scientific studies that characterized the tectonic activity in the region. These studies provided data and information on (i) the presence of faults within approximately 100 km [62 mi] of Yucca Mountain and if these faults had sustained Quaternary activity; (ii) the history and characteristics of past earthquakes, which were obtained from the results of detailed paleoseismic fault-trenching studies of active faults near Yucca Mountain; (iii) contemporary seismicity; (iv) historical and instrumentally recorded earthquakes in the Yucca Mountain region; (v) ground motion attenuation relationships for extensional tectonic regimes; (vi) local site attenuation characteristics; (vii) the tectonic stresses from hydrofracture measurements and earthquake focal mechanisms; (viii) geophysical data to assess tectonic models and identify subsurface faults; and (ix) geodetic data to measure ongoing crustal deformation.

Staff review indicates DOE adequately evaluated the nature and amount of faulting and the appropriate range of both principal and secondary faulting hazard sources within the repository block. In addition, DOE adequately determined fault geometry applicable to development of the probabilistic fault displacement hazard assessment. Given present knowledge, the DOE interpretations of faulting from surficial and underground mapping, as presented in U.S. Geological Survey (1998), are geologically consistent and reasonable. The experts adequately noted faults as primary or secondary for the purpose of the probabilistic fault displacement hazard assessment. The fault displacement hazard assessment must be reevaluated, however, if new faulting characteristics or data are identified. Some fault data taken by DOE from surface outcrops and from the exploratory studies facilities have been confirmed by independent checks by the NRC staff (NRC, 1999b). The variation of fault orientation data is within acceptable limits for normal geologic work. Field checks of fault locations, orientations, displacements, and other selected geometric features are generally in close agreement with the DOE observations and interpretations.

3.2.2.4.2.5 Uncertainty in Event Probability Is Adequately Evaluated

Overall, the current information, along with agreements reached between DOE and NRC, is sufficient to conclude that the necessary information will be available to assess the probability of faulting affecting the repository system at the time of a potential license application.

The technical basis for the estimate of uncertainty in the probability model is contained in CRWMS M&O (2000a) and the CRWMS M&O (2000i) analysis and model report. The uncertainty in the event probability is obtained from the results of the probabilistic seismic hazard assessment. Uncertainty in the estimate of the probability of a faulting event is based on the range of results in the probabilistic fault displacement hazard assessment from the different experts. DOE incorporates the uncertainty in the probability of the event by using the median value from the range of expert predictions for low probability ($<10^{-6}$ per year) fault displacements.

Staff disagree with the statistic used to combine the fault displacement hazard curves from the different experts in the probabilistic seismic hazard assessment. DOE uses the median value of the curves of the experts as the statistic of interest, whereas NRC staff believe that the mean is the more appropriate measure. Using the mean value of the curves would lead to a larger displacement being predicted at the 10^{-8} annual probability level. DOE agreed[8] to provide technical justification for use of median values or another statistical measure, such as the mean, or will evaluate and implement an alternative approach.

3.2.2.4.3 Seismicity

The probability of a seismic event affecting the repository system was discussed and reached closed-pending status at a technical exchange held in October 2000.[9] All information required to complete the agreements is expected to be received by the NRC by fiscal year 2003.

3.2.2.4.3.1 Events Are Adequately Defined

Overall, the current information is sufficient to conclude that the necessary information will be available to assess the probability of seismicity affecting the repository system at the time of a potential license application.

The approach and technical basis for defining seismic events are contained in CRWMS M&O (2000a). DOE indicates that small magnitude seismic events will be common at the Yucca Mountain repository whereas larger, more damaging seismic events will be less likely. Seismic events have the potential to affect performance through any of three effects: (i) rockfall causing direct damage to engineered barriers, (ii) failure of cladding, or (iii) changes to the groundwater flow system. These effects depend on the magnitude of the seismic event, so DOE defined a hazard curve in the probabilistic seismic hazard assessment (U.S. Geological Survey, 1998) that describes the probability of exceeding an earthquake of a given magnitude. A detailed review of the seismic aspects of the probabilistic seismic hazard assessment, including staff concerns and related agreements, is contained in Section 3.3.2 of this issue resolution status report.

The definition of events is sufficiently clear in the DOE documentation to allow NRC to have enough information at the time of licensing to make a regulatory decision in this area.

3.2.2.4.3.2 Probability Estimates for Future Events Are Supported by Appropriate Technical Basis

Overall, the current information, along with agreements reached between DOE and NRC, is sufficient to conclude that the necessary information will be available to assess the probability of seismic activity affecting the repository system at the time of a potential license application.

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[8]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Structural Deformation and Seismicity (October 11–12, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[9]Ibid.

Repository Safety After Permanent Closure

The approach and technical basis for defining the probability of seismicity affecting the repository system are contained in CRWMS M&O (2000a,e,i). DOE concluded that seismicity at Yucca Mountain is likely but that the magnitude of the event is an inverse function of the probability. The basis for the estimate of the probability of seismic events exceeding a given magnitude is the result of an expert elicitation documented in the U.S. Geological Survey (1998). A detailed review of the seismic aspects of the probabilistic seismic hazard assessment related to damage to cladding, including staff concerns and related agreements, is contained in Section 3.3.1 of this issue resolution status report. A detailed review of the seismic aspects of the probabilistic seismic hazard assessment related to rockfall and drift collapse, including staff concerns and related agreements, is contained in Section 3.3.2 of this issue resolution status report.

NRC staff have not identified any additional concerns beyond those identified in Sections 3.3.1 and 3.3.2 of this issue resolution status report.

3.2.2.4.3.3 Probability Model Support Is Adequate

Overall, the current information, along with agreements reached between DOE and NRC, is sufficient to conclude that the necessary information will be available to assess the probability of seismic activity affecting the repository system at the time of a potential license application.

The support for the probability model for seismicity affecting the repository system is contained in the CRWMS M&O (2000a,e,i). DOE concluded that seismicity at Yucca Mountain is likely but that the magnitude of the event is an inverse function of the probability. The basis for the estimate of the probability of seismic events exceeding a given magnitude is the result of an expert elicitation documented in the U.S. Geological Survey (1998). A detailed review of the seismic aspects of the probabilistic seismic hazard assessment related to damage to cladding, including staff concerns and related agreements, is contained in Section 3.3.1 of this issue resolution status report. A detailed review of the seismic aspects of the probabilistic seismic hazard assessment related to rockfall and drift collapse, including staff concerns and related agreements, is contained in Section 3.3.2 of this issue resolution status report.

NRC staff have not identified any additional concerns beyond those identified in Sections 3.3.1 and 3.3.2 of this issue resolution status report.

3.2.2.4.3.4 Probability Model Parameters Have Been Adequately Established

Overall, the current information, along with agreements reached between DOE and NRC, is sufficient to conclude that the necessary information will be available to assess the probability of seismic activity affecting the repository system at the time of a potential license application.

The approach and technical basis for defining the parameters for the probability model for seismicity affecting the repository system are contained in CRWMS M&O (2000a,e,i). DOE concluded that seismicity at Yucca Mountain is likely but that the magnitude of the event is an inverse function of the probability. The basis for the estimate of the probability of seismic events exceeding a given magnitude is the result of an expert elicitation documented in the U.S. Geological Survey (1998). A detailed review of the seismic aspects of the probabilistic seismic hazard assessment related to damage to cladding, including staff concerns and related agreements, is contained in Section 3.3.1 of this issue resolution status report. A detailed review of the seismic aspects of the probabilistic seismic hazard assessment related to rockfall and drift collapse, including staff concerns and related agreements, is contained in Section 3.3.2 of this issue resolution status report.

NRC staff have not identified any additional concerns beyond those identified in Sections 3.3.1 and 3.3.2 of this issue resolution status report.

3.2.2.4.3.5 Uncertainty in Event Probability Is Adequately Evaluated

Overall, the current information, along with agreements reached between DOE and NRC, is sufficient to conclude that the necessary information will be available to assess the probability of seismic activity affecting the repository system at the time of a potential license application.

The approach and technical basis for determining the uncertainty in the probability of seismicity affecting the repository system are contained in the CRWMS M&O (2000a,e,i). DOE concluded that seismicity at Yucca Mountain is likely but that the magnitude of the event is an inverse function of the probability. The basis for the estimate of the probability of seismic events exceeding a given magnitude is the result of an expert elicitation documented in the U.S. Geological Survey (1998). A detailed review of the seismic aspects of the probabilistic seismic hazard assessment related to damage to cladding, including staff concerns and related agreements, is contained in Section 3.3.1 of this issue resolution status report. A detailed review of the seismic aspects of the probabilistic seismic hazard assessment related to rockfall and drift collapse, including staff concerns and related agreements, is contained in Section 3.3.2 of this issue resolution status report.

NRC staff have not identified any additional concerns beyond those identified in Sections 3.3.1 and 3.3.2 of this issue resolution status report.

3.2.2.4.4 Nuclear Criticality

The probability of a criticality event affecting the repository system was discussed and reached closed-pending status at a technical exchange held in October 2000.[10] NRC expects to receive all information required to complete the agreements by fiscal year 2003 or before the submission of any license application for a repository at Yucca Mountain.

3.2.2.4.4.1 Events Are Adequately Defined

Overall, the current information is sufficient to conclude that the necessary information will be available to assess the probability of criticality in the repository system at the time of a potential license application.

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[10]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Criticality (October 23–24, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

The approach and technical basis for defining criticality events are contained in DOE (2000), and the calculation is in CRWMS M&O (2000b). DOE considers three major categories of criticality events: events, near-field events, and far-field events. The fuel can be in either intact or degraded condition for in-package events that occur within the waste package or near-field events that occur within the drift. Far-field events occur in the unsaturated zone or saturated zone below the repository and can only occur after the fuel degrades and releases fissile material.

NRC considers acceptable the division of criticality events based on the location of the event (e.g., in-package, near-field, and far-field).

3.2.2.4.4.2 Probability Estimates for Future Events Are Supported by Appropriate Technical Basis

Overall, the current information, along with agreements reached between DOE and NRC, is sufficient to conclude that the necessary information will be available to assess the probability of criticality in the repository system at the time of a potential license application.

The approach and technical basis for estimating the probability of criticality events are contained in DOE (2000), and the calculation is in CRWMS M&O (2000b). The probability of criticality in 10,000-year calculations does not follow the methodology outlined in the Topical Report on Disposal Criticality. Instead, it attempts to perform a simplified analysis to demonstrate that criticality events can be screened from the Total System Performance Assessment. The screening argument in this document for criticality is based on the low probability of a waste package failing within the first 10,000 years except through igneous events. Criticality in the waste package or the near field after an igneous event can be screened on the basis of low probability of forming a critical configuration after the event (CRWMS M&O, 2000b). The probability of a waste package failing before 10,000 years is stated to be 2.7×10^{-11}/waste package (CRWMS M&O, 2000b) based on results in the analysis and model report (CRWMS M&O, 2000j). This value, however, is based only on the probability of early waste package failure because of welding flaws. Other mechanisms for waste package failure are analyzed in this analysis and model report, including failures caused by flaws in the base metal, use of improper weld material, improper heat treatment of the welds, and damage incurred during handling operations. The occurrence of these failure mechanisms is much more likely than failures caused by flaws in the welds [a total of about 5.5×10^{-5} waste package (CRWMS M&O, 2000j)]. Additionally, this value of 2.7×10^{-11} was based on a value of 11.5 mm [0.45 in.] for the depth at which the stress in the waste package goes from compressive to tensile. However, this value is identified as being used only for an example to demonstrate the models rather than defensible data. Therefore, this value should not be used to screen events from the Total System Performance Assessment. NRC staff review of the analysis and model report (CRWMS M&O, 2000j) also identified several concerns. First, failure rates used in the calculations averaged failure data throughout a long history that allowed for improvements in fabrication techniques. These data may not be appropriate for the waste package, which will be manufactured using a new fabrication process and may not be able to benefit from the identification of improvements in the fabrication process as failures are identified. Second, the welding and heat treatment of the outer lids are remote operations (Bechtel SAIC Company, LLC, 2001), so the sequence of operations may not include a final laboratory check. This laboratory check was relied on when developing the probability of failure because of an improper heat treatment, and the probability of failure of a waste package would increase substantially without it. Third, the probability of handling damage did not include the possibility that an uninspected, damaged disposal container arriving from the fabricator remains undetected during arrival inspections at the repository. Additionally, a screening argument for criticality after igneous-induced waste package failure has only been provided for commercial spent nuclear fuel, not for DOE spent nuclear fuel or defense high-level waste. Therefore, the probability estimates that are used as the basis of the screening argument are not sufficient to support the screening of criticality from the performance assessment.

DOE submitted a topical report (DOE, 2000) that describes the methodology that will be used to determine the probability and consequences of a criticality event at the Yucca Mountain repository. This methodology provides a detailed analysis of possible locations within the repository system where a criticality event may occur. Using a probabilistic methodology, the criticality analysis will perform a detailed tracking of the fissile and neutron poison materials during the degradation of the waste form and waste package structural materials to determine the probability of a critical configuration being generated. NRC reviewed the initial revision of DOE (1998) and issued a safety evaluation report documenting the results of the staff review of the document (NRC, 2000d). This safety evaluation report contained 28 Open Items, which are areas of concern that NRC staff have about the methodology. DOE indicated that Revision 1 of the topical report has addressed 27 of the Open Items, and the resolution of the other Open Item, related to the verification of burnup of the spent nuclear fuel, is the subject of Agreement PRE.07.01. Additionally, a recent document DOE released attempts to screen criticality using a simple fault tree to determine the probability of criticality in the repository system. Both documents are currently being reviewed by NRC staff. Therefore, although DOE has not provided adequate justification for the screening of criticality from the repository system at this point, the information provided, along with the information required to be provided in the agreements,[11] will allow NRC staff to have sufficient information at the time of the license application to evaluate the DOE safety case.

3.2.2.4.4.3 Probability Model Support Is Adequate

Overall, the current information, along with agreements reached between DOE and NRC, is sufficient to conclude that the necessary information will be available to assess the probability of criticality in the repository system at the time of a potential license application.

The description of the support for the probability model is contained in DOE (2000) CRWMS M&O (2000d). The models that will be used to calculate the probability of a criticality event occurring within the repository system will be controlled under the DOE Configuration Management system. The primary codes in DOE (2000) that will need to be validated include geochemistry codes, neutron transport codes, and the configuration generator code. Where possible, DOE will use the same geochemistry codes as those in other areas of the repository

[11]Schlueter, J.R. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Criticality (October 23–24, 2000).” Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

program, within their range of validation. For example, the in-package chemistry code used in the criticality analysis will be validated to support the spent nuclear fuel dissolution model in the repository program, and the validation will not be repeated for the criticality analysis. However, the criticality analysis may need to perform geochemistry calculations for materials and areas of the repository outside the range of validation performed for the repository system. DOE will have to perform additional software validation to support the use of these models in these situations. The validation of the geochemical codes will be performed by comparing the results from the code against analytical solutions and against results obtained from other geochemistry-transport codes.

The neutron transport code will be validated by comparing the results of the code to data obtained from Commercial Reactor Critical experiments, radiochemical analyses, and Laboratory Critical Experiments. Any bias associated with the neutron transport code will be identified using these experiments and will be accounted for before comparing the calculated neutron multiplication factor to the critical limit. The configuration generator code will be validated by comparing the results of the code with appropriate hand calculations to demonstrate that it is implementing the model correctly.

Additionally, natural analog information will be used to gain insight in the behavior of radionuclides in the natural environment. For example, information from the natural reactors at Oklo, Gabon, Equatorial Africa, will provide insight on mechanisms of accumulation of fissile materials and transport of the resulting actinides and fission products away from the fissioning material. Additionally, information from the natural uranium deposit in Peña Blanca, Mexico, provides insights into the processes that lead to the accumulation and mobilization of uranium in unsaturated tuff.

The NRC staff review indicates that the proposed methodology of providing support for the probability calculation is appropriate. DOE agreed[12] to submit validation reports documenting the validation of the computer codes that will be used to calculate the probability of criticality within the repository system before the license application.

3.2.2.4.4.4 Probability Model Parameters Have Been Adequately Established

Overall, the current information, along with agreements reached between DOE and NRC, is sufficient to conclude that the necessary information will be available to assess the probability of criticality in the repository system at the time of a potential license application.

The approach for developing the technical bases for parameters used in the probability models is contained in DOE (1998, 2000), and the calculation is in CRWMS M&O (2000b). The parameters that will be used in calculating the probability of a criticality event occurring in the repository system will be derived from information developed and reviewed from other areas of the repository system. Important parameters in calculating the probability of criticality in the

[12]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Criticality (October 23–24, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

repository system that will be justified in other areas of the repository program include the number of waste packages failed, parameters affecting the quantity of water entering the waste package (including the percolation rate and the seepage flow rate), water chemistry, the degradation rate of the fuel, and transport properties of the fissile materials (DOE, 1998). To support the use of these parameters, DOE will need only to demonstrate the parameters are consistent with the repository program and that there are no assumptions made in the selection of these parameter values that would be conservative with respect to nominal repository performance but nonconservative for the criticality calculation. Other parameters may be important in the calculation of the probability of criticality but not in other areas of the repository program, such as the degradation rate of basket support materials (DOE, 1998). DOE has agreed to providing proper justification for any parameter values for which sufficient justification has not been developed in other areas of the repository program. In general, DOE agreed[13] to provide an updated technical basis for screening criticality from the postclosure performance assessment.

The proposed methodology of using appropriate parameter values from other areas of the repository program in the criticality modeling is acceptable. Review of the justification of parameter values not defended in other areas of the repository program will be conducted when DOE provides the detailed calculations to determine the probability of criticality for all fuel types.

3.2.2.4.4.5 Uncertainty in Event Probability Is Adequately Evaluated

Overall, the current information is sufficient to conclude that the necessary information will be available to assess the probability of criticality in the repository system at the time of a potential license application.

The approach for calculating the uncertainty in the probability of criticality events is contained in DOE (2000), and the calculation is in CRWMS M&O (2000b). Using the topical report methodology, DOE will determine the probability of criticality by performing a Monte Carlo simulation that tracks the failure of the waste package, degradation of internal components of the waste package, and transport of fissile and poison materials through the repository system.

Parameters used in this model will be sampled from an uncertainty distribution to determine whether the system could go critical for a given parameter set. The estimate of the probability of criticality will be controlled by the uncertainty distributions used in the models. In the Monte Carlo process, an additional source of uncertainty is statistical uncertainty based on the number of realizations run. DOE indicated it will conduct sufficient realizations to ensure that this component of uncertainty is very small.

The methodology to estimate the probability of criticality in CRWMS M&O (2000b) is a deterministic calculation. These deterministic calculations rely on conclusions in other documents that the waste package will not fail within 10,000 years because of corrosion

[13]Schlueter, J.R. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Criticality (October 23–24, 2000).” Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

processes (CRWMS M&O, 2000j) or seismic events (CRWMS M&O, 2000a). After an igneous event, these calculations use the mean values of distributions for water transport parameters and the fraction of waste packages capable of supporting a criticality event to demonstrate that the probability of a criticality event is a low-probability event.

The NRC staff review indicates that the proposed methodology in the Topical Report to include uncertainty in the estimate of the probability of a criticality event is appropriate.

3.2.2.5 Status and Path Forward

Table 3.2.2-1 provides related DOE and NRC agreements pertaining to the Identification of Events with Probability Greater Than 10^{-8} Per Year. The status and the detailed agreements (or path forward) pertaining to all the key technical issue subissues are provided in Table 1.1-3 and Appendix A. Additional agreements from the DOE and NRC Technical Exchange on August 6–10, 2001, are summarized in Appendix B.

The Total System Performance Assessment and Integration Key Technical Issue Subissue pertaining to the scenario analysis is considered closed-pending. Following is a summary of issues that DOE needs to resolve before this subissue can be closed.

Table 3.2.2-1. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreements*
Igneous Activity	Subissue 1—Probability of Igneous Activity	Closed-Pending	IA.1.01 IA.1.02
Structural Deformation and Seismicity	Subissue 1—Faulting	Closed-Pending	SDS.1.02
	Subissue 2—Seismicity	Closed-Pending	SDS.2.01 SDS.2.03
Container Life and Source Term	Subissue 5—The Effect of In-Package Criticality on Waste Package and Engineered Barrier Subsystem Performance	Closed-Pending	CLST.5.01 CLST.5.03 CLST.5.04
Evolution of the Near-Field Environment	Subissue 4—Coupled Thermal-Hydrologic-Chemical Processes on Radionuclide Transport Through Engineered and Natural Barriers	Closed-Pending	ENFE.5.01 ENFE.5.03
Radionuclide Transport	Subissue 4—Nuclear Criticality in the Far Field	Closed-Pending	RT.4.01 RT.4.03

Table 3.2.2-1. Related Key Technical Issue Subissues and Agreements (continued)			
Key Technical Issue	**Subissue**	**Status**	**Related Agreements***
Total System Performance Assessment and Integration	Subissue 2—Scenario Analysis and Event Probability	Closed-Pending	TSPAI.2.01 TSPAI.2.02 TSPAI.2.05 TSPAI.2.06 TSPAI.2.07
	Subissue 3—Model Abstraction	Closed-Pending	TSPAI.3.06
*Related DOE and NRC agreements are associated with one or all five generic acceptance criteria.			

3.2.2.6 References

Bechtel SAIC Company, LLC. "FY01 Supplemental Science and Performance Analyses." Vol. 1: Scientific Bases and Analyses. TDR–MGR–MD–000007. Revision 00 ICN 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2001.

Condit, C.D., and C.B. Connor. "Recurrence Rates of Volcanism in Basaltic Volcanic Fields: An Example from the Springerville Volcanic Field, Arizona." *Geological Society of America Bulletin*. Vol. 108, No. 10. pp. 1,225–1,241. 1996.

CRWMS M&O. "Probabilistic Volcanic Hazard Analysis for Yucca Mountain, Nevada." BA0000000–01717–2200–00082. Revision 0. Las Vegas, Nevada: CRWMS M&O. 1996.

———. "Disruptive Events Process Model Report." TDR–NBS–MD–000002. Revision 00 ICN 01. North Las Vegas, Nevada: DOE, Yucca Mountain Site Characterization Office. 2000a.

———. "Probability of Criticality Before 10,000 Years." CAL–EBS–NU–000014. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000b.

———. "Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations." TDR–WIS–RL–000001 Revision 04 ICN 01. North Las Vegas, Nevada: DOE, Yucca Mountain Site Characterization Office. 2000c.

———. "Total System Performance Assessment for the Site Recommendation." TDR–WIS–PA–000001. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000d.

———. "Disruptive Events FEPs." ANL–WIS–MD–000005. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000e.

———. "Characterize Framework for Igneous Activity at Yucca Mountain, Nevada." ANL–MGR–GS–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000f.

———. "Fault Displacement Effects on Transport in the Unsaturated Zone." ANL–NBS–HS–000020. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000g.

———. "Effects of Fault Displacement on Emplacement Drifts, Analysis and Models Report." ANL–EBS–GE–000004. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000h.

———. "Characterize Framework for Seismicity and Structural Deformation at Yucca Mountain." ANL–CRW–GS–000003. Las Vegas, Nevada: CRWMS M&O. 2000i.

———. "Analysis of Mechanisms for Early Waste Package Failure." ANL–EBS–MD–000023. Revision 02. Las Vegas, Nevada: CRWMS M&O. 2000j.

dePolo, C.M., D.G. Clark, D.B. Slemmons, and A.R. Ramelli. "Historical Surface Faulting in the Basin and Range Province, Western North America, Implications for Fault Segmentation." *Journal of Structural Geology*. Vol. 13. pp. 123–126. 1991.

DOE. "Disposal Criticality Analysis Methodology Topical Report." YMP/TR–004Q. Revision 00. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 1998.

———. "Disposal Criticality Analysis Methodology Topical Report." YMP/TR–004Q. Revision 01. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 2000.

Fridrich, C.J., J.W. Whitney, M.R. Hudson, and B.M. Crowe. "Late Cenozoic Extension, Vertical-Axis Rotation, and Volcanism in the Crater Flat Basin, Southwest Nevada, Cenozoic Basins of the Death Valley Region." Geological Society of America Special Paper 333. L. Wright and B. Troxel, eds. Boulder, Colorado: Geological Society of America. pp. 197–212. 1999.

Hill, B. and C. Connor. "Technical Basis for Resolution of the Igneous Activity Key Technical Issue." San Antonio, Texas: CNWRA. 2000.

NRC. NUREG–1563, "Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program." Washington, DC: NRC. November 1996.

———. "Issue Resolution Status Report, Key Technical Issue: Igneous Activity." Revision 2. Washington, DC: NRC. 1999a.

———. "Issue Resolution Status Report, Key Technical Issue: Structural Deformation and Seismicity." Revision 2. Washington, DC: NRC, Division of Waste Management. 1999b.

———. "Issue Resolution Status Report, Key Technical Issue: Evolution of the Near-Field Environment." Revision 3. Washington, DC: NRC. 2000a.

———. "Issue Resolution Status Report, Key Technical Issue: Radionuclide Transport." Revision 2. Washington, DC: NRC. 2000b.

———. "Issue Resolution Status Report, Key Technical Issue: Total System Performance Assessment and Integration." Revision 3. Washington, DC: NRC. 2000c.

———. "Safety Evaluation Report for Disposal Criticality Analysis Methodology Topical Report." Revision 0. Washington, DC: NRC. 2000d.

———. "Issue Resolution Status Report Key Technical Issue: Container Life and Source Term." Revision 3. Washington, DC: NRC. 2001.

———. NUREG–1804, "Yucca Mountain Review Plan—Draft Report for Comment." Revision 2. Washington, DC: NRC. March 2002.

U.S. Geological Survey. "Probabilistic Seismic Hazard Analyses for Fault Displacement and Vibratory Ground Motion at Yucca Mountain, Nevada, Final Report." I.G. Wong and C. Stepp, coords. SP32IM3, WBS Number 1.2.3.2.8.3.6. Oakland, California: DOE. 1998.

Wong, I.G. and Olig, S.S. "Seismic Hazards in the Basin and Range Province: Perspectives from Probabilistic Analyses." In Western States Seismic Policy Council Proceedings Volume. Basin and Range Province Seismic-Hazards Summit. W.R. Lund, ed. *Miscellaneous Publication*. Vol. 98, No. 2. pp. 110–127. West Valley, Utah: Utah Geological Survey. 1998.

3.3 Model Abstraction

3.3.0 Model Abstraction—Generic Discussion

3.3.0.1 Description of Issue

When reviewing the DOE total system performance assessment, the NRC staff will evaluate elements (or model abstractions) of the repository system to determine how effective the overall system is at protecting the public health and safety. As discussed in Chapter 1, Introduction, there are 14 model abstraction sections the staff will use to determine compliance with 10 CFR 63.114 (see Figure 1.1-2 for a description of the model abstractions). These abstractions consider the aspects of the engineered, geosphere, and biosphere subsystems that may be important to performance. Important to performance means important to meeting the postclosure performance objectives specified at 10 CFR 63.113 and 63.311. The staff will use risk insights to focus their review on the important assumptions, models, and data in the total system performance assessment. The staff will also focus their review to ensure the degree of technical support for models and data abstractions is commensurate with its contribution to risk, which means the staff will review in greater detail those model abstractions and their important components on which DOE relies more heavily to prove its safety case.

The staff will also review the DOE total system performance assessment to decide if DOE properly characterized the features, events, and processes and properly incorporated them into the total system performance assessment. This review is necessary to decide if the DOE total system performance assessment is acceptable and complies with 10 CFR 63.114 and 63.115. The review methods and acceptance criteria the staff will use to evaluate compliance with the performance objectives (numerical standards) are in Section 4.2.1.4 of NRC (2002).

3.3.0.2 Relationship to Key Technical Issue Subissues

The following sections (3.3.1–3.3.14) discuss the 14 model abstractions. In each section, staff describes the relationship between the key technical issue subissues and the specific model abstraction being addressed.

The remainder of Section 3.3.0 discusses general issues and concerns associated with multiple model abstractions. These issues were identified as part of the staff review of the DOE site recommendation documents (CRWMS M&O, 2000a,b; DOE, 2001; Bechtel SAIC Company, LLC, 2001a,b) and various analysis and model reports (received through October 2001). The general issues the staff identified include

- Improvement needed in transparency and traceability of the model abstraction documentation

- Appropriately rigorous methodology not used for model abstraction simplifications and selections of parameter distributions, conceptual models, or modeling approaches

- Inadequate basis provided for the amount of information retained by the model abstractions

- Inadequate support for the process model results abstracted in the total system performance assessment and for the total system performance assessment

3.3.0.3 Importance to Postclosure Performance

A full and clear understanding of model abstractions is important to gain reasonable assurance in the estimated postclosure performance of the repository. The generic items discussed in this section (i.e., transparency and traceability of analyses, consistency of assumptions across various abstractions, and the verification of abstracted models through comparison with results from detailed process models) are applicable to all 14 abstractions discussed in Sections 3.3.1–3.3.14.

3.3.0.4 Technical Basis

Overall, the current information, along with the DOE and NRC agreements (Section 3.3.0.5), is sufficient to conclude the necessary information will be available, at the time of a potential license application, to allow NRC to conduct a detailed review.

A number of positive examples in the documentation are related to transparency and traceability. A positive example of transparency and traceability is seen in the DOE consideration and comparison of advective versus diffusive releases from the waste package. There are some areas, however, that need improvement. In particular, numerous examples exist where the discussion in a summary section or an individual abstraction section is inconsistent with other sections, the actual total system performance assessment model, or with the related analysis and model reports.[1] In particular, there are contradictory statements about the role of environmental variables in the corrosion models. In aggregate, the inconsistencies make it difficult for the reviewers to understand clearly some parts of the total system performance assessment model.

DOE agreed that transparency and traceability of documents will be improved and outlined its planned activities to improve the transparency and traceability:

- Update review procedures, with an emphasis on vertical slice reviews (e.g., by chapter and between documents to improve consistency)

- Improve or update the documents mentioned in the specific examples noted by NRC

- Complete a vertical slice review for consistency, which was under way at the time of the technical exchange

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[1]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

- Develop additional transparency tools, such as a flow chart of the total system performance assessment model, to further explain how data are passed between components and subcomponents of the overall Total System Performance Assessment–Site Recommendation model and the sources of these data and new graphics

- Allow time for additional reviews to include international peer review panels, internal review teams, and technical editors

To improve transparency and traceability, DOE also agreed to revisit the abstraction of colloid modeling and the use of the Waste Package Degradation Model in modeling the failure of the engineered barrier subsystem. NRC considered adequate the DOE general response addressing transparency and traceability, during the technical exchange of August 6–10, 2001.[2]

Based on a review of the Total System Performance Assessment–Site Recommendation and the supporting analysis and model reports, NRC staff consider the DOE methodology used for model abstraction simplifications and the selection of conservative parameter distributions, conceptual models, or modeling approaches needs additional rigor. In addition to integrating various abstractions into the total system performance assessment, DOE needs to use a consistent approach for conducting the total system performance assessment and making judgments regarding conservatism (i.e., leading to overestimating radiological consequences) and the treatment of uncertainty. For example, the system model or individual abstractions are sufficiently complex, which means human intuition cannot be relied on to make accurate decisions consistently. Specifically, it may be impossible to determine the effect of a parameter *a priori* for the complex, nonlinear models embedded in the total system performance assessment. Because of the interactions at the system level or among different parts of the system, intermediate parameter values may lead to larger doses to the reasonably maximally exposed individual than either bound of the distribution. For example, if ionic strength affected both colloid stability and cladding corrosion, it is possible that minimizing ionic strength to maximize colloid stability may not result in maximizing dose to the reasonably maximally exposed individual because it would also reduce the rate of cladding corrosion. A reduction in cladding corrosion corresponds to reduced releases of radionuclides and, consequently, a reduction in the transport of radionuclides in colloids and a reduction in the dose.

DOE agreed to improve this area and to develop written guidance in the model abstraction process for model developers so that: (i) the model abstraction process, (ii) the selection of conservatism in components, and (iii) the representation of uncertainty are systematic across the total system performance assessment model. These guidelines will address the evaluation of nonlinear models when conservatism is being used to address uncertainty and decisions are based on technical judgment in a complex system. DOE agreed the guidelines will be developed, implemented, and made available to NRC in fiscal year 2002. These proposed improvements represent an acceptable approach to address the NRC questions. In addition,

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[2]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

the opportunity provided by availability of the guidance in fiscal year 2002 provides additional confidence that DOE will be able to implement these changes systematically in sufficient time to improve the total system performance assessment. Finally, if NRC has questions regarding the specific DOE approach, these questions can be communicated to DOE in a timely manner.

The abstraction process is typically a simplification of process model results into a form that represents an appropriate amount of uncertainty and variability, while allowing a computationally efficient solution. NRC recognizes that it is impossible to represent all of the spatial and temporal uncertainty and variability, as well as conceptual model uncertainty, in the overall total system performance assessment model. Staff have identified several instances, however, where DOE has not provided sufficient justification for the amount of information retained by the abstraction.[3] Specifically, DOE needs to justify the simplifications used with consideration of all affected subsystems or models. Two examples of inadequate technical bases for the simplification used in a model abstraction include (i) the DOE decision not to represent uncertainty in the infiltration map at each climate state and (ii) the DOE assumption that three seepage threshold levels adequately capture the contribution from the tails of the distribution.

DOE agreed to document the simplifications used for abstractions for all future total system performance assessments (TSPAI.3.39). DOE agreed to provide justification to show that the simplifications appropriately represent the necessary processes and appropriately propagate process model uncertainties. DOE also agreed to provide comparisons of output from process models to total system performance assessment abstractions. DOE indicated that the level of detail in the comparisons will be commensurate with any reduction in propagated uncertainty and the risk significance of the model. DOE stated that the documentation of the information will be provided in abstraction analysis and model reports in fiscal year 2003.

As part of the model development process, it is necessary to verify that the model is calculating properly, validate that an appropriate model has been developed for the problem being examined, and explain the detailed functioning of the model through complete analyses. DOE provided information on all three topics in CRWMS M&O (2000b). Several concerns were identified during the NRC staff review of the DOE Total System Performance Assessment–Site Recommendation model documentation. The following are examples of these concerns:

- Various errors were found in the DOE hand calculations.
- Abstracted models were used outside the ranges for which they were developed.
- It is not clear that DOE evaluated the significance of warnings and errors in the GoldSim (Golder Associates, 2000) error log file: neither the significance nor the evaluation of the warnings and errors were documented.

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[3]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

- DOE identified the elements of verification in CRWMS M&O (2000b) and supporting documents but has not rigorously verified the Total System Performance Assessment–Site Recommendation computer program.

- The limited set of random hand calculations did not represent a systematic approach to verification.

DOE issued Corrective Action Report No. BSC–01–C–001 dated May 3, 2001, that found " ... the area of model validation is considered to be a significant condition adverse to quality." The corrective action report indicates that 18 of 24 analysis and model reports were inadequately validated, including 8 that were not validated at all. As the corrective action report indicates, the other methods deemed acceptable to develop support for process models were not satisfied.

DOE indicated that a root-cause analysis was being performed for Corrective Action Report No. BSC–01–C–001. DOE agreed to document the process used to develop confidence in the total system performance assessment models [e.g., steps similar to those described in NUREG–1636 (NRC, 1999)]. The detailed process is currently documented in the model development procedures being evaluated for process improvement in response to the model validation Corrective Action Report No. BSC–01–C–001. The upgraded model validation procedures will be available for NRC to review in fiscal year 2002. Additionally, DOE will document the implementation of the process for model confidence building and will demonstrate compliance with model confidence criteria in accordance with applicable procedures. This compliance will be documented in the respective analysis and model report revisions and made available to NRC in fiscal year 2003.

3.3.0.5 Status and Path Forward

Table 3.3.0-1 provides the DOE and NRC agreements pertaining to general issues and concerns associated with multiple model abstractions. Note that the status, and also the detailed agreements (or path forward) pertaining to all the key technical issue subissues, are provided in Table 1.1-3 and Appendix A. The DOE approach, together with the DOE agreements to provide NRC with additional information (through specified testing, analyses, and the like), acceptably addresses the NRC questions so that no information beyond that provided or agreed to will be required at the time of a potential license application. Sections 3.3.1 through 3.3.14 identify specific issues and concerns associated with each individual model abstraction.

Table 3.3.0-1. Related Key Technical Issue Subissues and Agreements			
Key Technical Issue	**Subissue**	**Status**	**Related Agreements***
Total System Performance Assessment and Integration	Subissue 3—Model Abstraction	Closed-Pending	TSPAI.3.38 TSPAI.3.39
	Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards	Closed-Pending	TSPAI.4.05 TSPAI.4.06
*Related DOE and NRC agreements are associated with one or all five generic acceptance criteria.			

3.3.0.6 References

Bechtel SAIC Company, LLC. "FY01 Supplemental Science and Performance Analyses." Vol. 1: Scientific Bases and Analyses. TDR–MGR–MD–000007. Revision 00 ICN 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2001a.

Bechtel SAIC Company, LLC. "FY01 Supplemental Science and Performance Analyses." Vol. 2: Performance Analyses. TDR–MGR–PA–000001. Revision 00 ICN 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2001b.

CRWMS M&O. "Total System Performance Assessment for the Site Recommendation." TRD–WIS–PA–000001. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000a.

———. "Total System Performance Assessment (TSPA) Model for Site Recommendation." MDL–WIS–PA–000002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000b.

DOE. "Yucca Mountain Science and Engineering Report—Technical Information Site Recommendation Consideration." DOE/RW–0539. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 2001.

Golder Associates. "Software Code: GoldSim." 6.04.007. Redmond, Washington: Golder Associates. 2000.

NRC. NUREG–1636, "Regulatory Perspectives on Model Validation in High-Level Radioactive Waste Management Programs: A Joint NRC/SKI White Paper." Washington, DC: NRC. March 1999.

———. NUREG–1804, "Yucca Mountain Review Plan—Draft Report for Comments." Revision 2. Washington, DC: NRC. March 2002.

3.3.1 Degradation of Engineered Barriers

3.3.1.1 Description of Issue

The Degradation of Engineered Barriers Integrated Subissue addresses the assessment of engineered barrier performance and waste package lifetimes. Engineered barriers include, in addition to the waste package, other components of the engineered barrier subsystem such as drip shield, drift invert, and backfill if any. In the proposed DOE site recommendation reference design for the various types of spent nuclear fuel and high-level waste glass, the waste package is composed (in addition to the various waste forms) of two concentric containers of different metallic materials emplaced horizontally in a drift. The outer container or barrier will be of a highly corrosion-resistant nickel-chromium-molybdenum alloy, Alloy 22, surrounding an inner container made of Type 316 nuclear grade stainless steel. Additionally, an inverted U-shaped drip shield, fabricated with a titanium-palladium alloy (Titanium Grade 7), will be extended over the length of the emplacement drifts, resting on the drift invert, to enclose the top and sides of the waste packages. Each waste package will rest on an emplacement pallet made of two Alloy 22 V-shaped supports connected by square stainless steel tubes, and emplaced on top of the drift invert. The current repository reference design does not include backfill. For undisturbed repository conditions, corrosion is expected to be the primary degradation process limiting the life of the principal engineered barriers, which are the waste package and the drip shield. Through-wall penetration of the drip shield by corrosion will facilitate contact of the water entering into the emplacement drifts with the waste package outer surface. The quantity and chemistry of water contacting the waste package, the relative humidity, the waste package temperature, and the metallurgical condition of the waste package materials will determine the mode and rate of corrosion of the waste package outer container. Loss of containment as a result of corrosion will allow release of radionuclides to the environment surrounding the waste package and their subsequent transport through the engineered barrier subsystem. The relationship between this integrated subissue and other integrated subissues is depicted in Figure 3.3.1-1 (NRC, 2000a). The overall organization and identification of all the integrated subissues are depicted in Figure 1.1-2.

This section provides a review of the abstractions of the engineered barrier degradation processes incorporated by DOE in its Total System Performance Assessment–Site Recommendation (CRWMS M&O, 2000a). Only degradation processes under undisturbed repository conditions are discussed. Mechanical disruption of the engineered barriers and volcanic disruption of waste packages (depicted in the left portion of Figure 3.3.1-1) are discussed in Sections 3.3.2. and 3.3.10. The DOE description and technical bases for the engineered barriers degradation abstractions focused on the waste package and drip shield are documented in the process model report CRWMS M&O (2000b) and in several related analysis and model reports. These analysis and model reports are reviewed to the extent that they contain models, data, and analyses that support the proposed Total System Performance Assessment–Site Recommendation abstractions. As appropriate, several system description documents are also reviewed to complete the evaluation of models and abstractions used by DOE in the performance assessment of the engineered barriers.

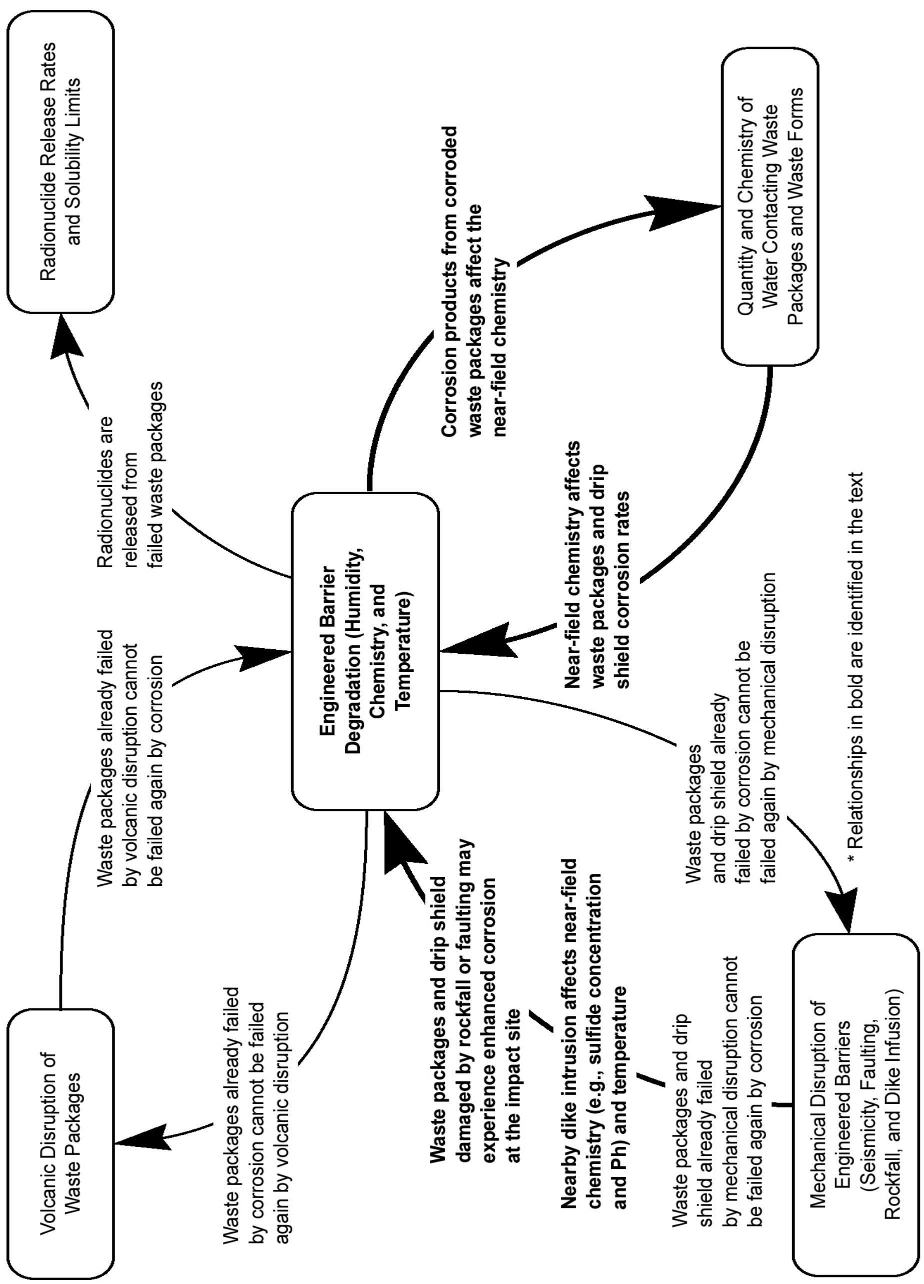

Figure 3.3.1-1. Diagram Illustrating the Relationship Between Engineered Barrier Degradation and Other Integrated Subissues

3.3.1.2 Relationship to Key Technical Issue Subissues

The Degradation of Engineered Barriers Integrated Subissue incorporates subject matter previously included in the following key technical issue subissues:

- Container Life and Source Term: Subissue 1—The Effects of Corrosion Processes on the Lifetime of the Containers (NRC, 2001)
- Container Life Source Term: Subissue 2—The Effects of Phase Instability of Materials and Initial Defects on the Mechanical Failure and Lifetime of the Containers (NRC, 2001)
- Container Life Source Term: Subissue 5—The Effect of In-package Criticality on Waste Package and Engineered Barrier Subsystem Performance (NRC, 2001)
- Container Life and Source Term: Subissue 6—The Effects of Alternate Engineered Barrier Subsystem Design Features on Container Lifetime and Radionuclide Release from the Engineered Barrier Subsystem (NRC, 2001)
- Total System Performance Assessment and Integration: Subissue 1—System Description and Demonstration of Multiple Barriers (NRC, 2000a)
- Total System Performance Assessment and Integration: Subissue 2—Scenario Analysis and Event Probability (NRC, 2000a)
- Total System Performance Assessment and Integration: Subissue 3—Model Abstraction (NRC, 2000a)
- Total System Performance Assessment and Integration: Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards (NRC, 2000a)
- Thermal Effects on Flow: Subissue 2—Thermal Effects on Temperature, Humidity, Saturation, and Flux (NRC, 2000b)
- Evolution of the Near-Field Environment: Subissue 2—The Effects of Coupled Thermal-hydrological-Chemical Processes on the Waste Package Chemical Environment (NRC, 2000c)
- Evolution of the Near-Field Environment: Subissue 3—The Effects of Coupled Thermal-hydrological-Chemical Processes on Chemical Environment for Radionuclide Release (NRC, 2000c)
- Evolution of the Near-Field Environment: Subissue 5—The Effects of Coupled Thermal-hydrological-Chemical Processes on Potential Nuclear Criticality in the Near Field (NRC, 2000c)

- Repository Design and Thermal-Mechanical Effects: Subissue 3—Thermal-Mechanical Effects on Underground Facility Design and Performance (NRC, 2000d)

The key technical issue subissues formed the bases for the previous versions of the issue resolution status reports and also were the bases for technical exchanges with DOE where agreements were reached on what additional information DOE needed to provide to resolve the subissue. The resolution status of this integrated subissue is based on the resolution status of each of the contributing key technical issue subissues. The subsequent sections incorporate applicable portions of these key technical issue subissues, however, no effort was made to explicitly identify each subissue.

3.3.1.3 Importance to Postclosure Performance

One aspect of risk-informing the NRC review was to determine how this integrated subissue is related to the DOE repository safety strategy. The performance of the engineered barriers after waste emplacement is extremely important to protect the public from any unreasonable long-term risk, as recognized in the DOE repository safety strategy for the proposed Yucca Mountain site (CRWMS M&O, 2000c). Both the performance of the waste package and that of the drip shield/drift invert system are listed among the eight principal factors for the postclosure safety case (CRWMS M&O, 2000c).

The waste package, composed of the containers and the waste forms, is the primary engineered barrier controlling the release of radionuclides from spent nuclear fuel and high-level waste glass. It should be noted, that contrary to the definitions of 10 CFR Part 63, DOE defines the waste package with the exclusion of the waste forms. Because corrosion processes, promoted by the presence of an aqueous environment contacting the surface of the containers, are the primary cause of container failure under undisturbed conditions, both the mode and rate of corrosion need to be evaluated to determine container lifetime. Corrosion processes potentially important in the degradation of the engineered barriers include humid-air and uniform aqueous corrosion, localized (pitting, crevice, and intergranular) corrosion, microbially influenced corrosion, stress corrosion cracking, and hydrogen embrittlement. In addition, dry-air oxidation occurs during the initial period after waste emplacement when the radioactive decay heat keeps moisture away from the gaseous environment surrounding the waste package. The ability of the waste package to contain radionuclides, and to limit their release after any initial penetration, is, therefore, determined by its long-term resistance to any of the modes of corrosion listed previously.

Performance of the drip shield needs to be considered as an important factor regarding safety because DOE incorporated it in the design of the engineered barrier subsystem to provide defense in depth by limiting the amount of water contacting the waste package as a result of dripping (CRWMS M&O, 2000c). Hence, the initiation of aqueous corrosion of waste packages can be delayed, resulting in a significantly longer container lifetime. In addition, once the containers are breached, the amount of water available for dissolution of both spent nuclear fuel and high-level waste glass and advective transport of the released radionuclides could be limited, even by the presence of a partially damaged drip shield.

The possibility of in-package criticality needs to be considered because steady-state criticality events could lead to increased radionuclide inventories. Depending on the power level and duration of critical conditions, significant amounts of radionuclides, including Tc-99, Np-237, and I-129, would be produced. The impact on repository performance would be an increase in radionuclide inventory available for release from the waste package and a potential increase in dose to the reasonably maximally exposed individual. Additionally, heat production from the additional fission reactions taking place during criticality conditions could indirectly impact repository performance by affecting the near-field environment and potentially increasing the waste package corrosion rate of the waste package and the dissolution rate of the waste form. Finally, a transient criticality event could result in mechanical failure of the already corroded waste package rupture of the spent nuclear fuel cladding, or both, increasing the exposed surface area and degradation rate of the spent nuclear fuel matrix.

3.3.1.4 Technical Basis

NRC developed a plan (2002) consistent with the acceptance criteria and review methods found in previous issue resolution status reports. A review of DOE approaches for including the degradation of engineered barriers in total system performance assessment abstractions is provided in the following subsections. The review of the technical basis for the degradation of engineered barriers abstraction is divided into three subsections: waste package, drip shield, and criticality within the waste package. Each subsection is organized according to the five acceptance criteria identified in Section 1.5: (i) System Description and Model Integration Are Adequate, (ii) Data Are Sufficient for Model Justification, (iii) Data Uncertainty Is Characterized and Propagated Through the Model Abstraction, (iv) Model Uncertainty Is Characterized and Propagated Through the Model Abstraction, and (v) Model Abstraction Output Is Supported by Objective Comparisons.

3.3.1.4.1 Degradation of the Waste Package

For undisturbed repository conditions, corrosion is considered the primary degradation process of the engineered barriers. In recent performance assessment studies, regardless of the specific waste package design, waste package degradation has been shown to be important to waste isolation at the proposed Yucca Mountain repository (Wilson, et al., 1994; CRWMS M&O, 1995, 1998a, 2000a; NRC, 1995, 1999; Kessler and McGuire, 1996; Shoesmith and Kolar,1998; DOE, 1998a; Mohanty and McCartin, 1998; Mohanty, et al., 1999). In addition, the NRC sensitivity studies have shown that the estimated average system performance during the 10,000-year period of regulatory interest is strongly influenced by the waste package lifetime (Mohanty, et al., 1999).

3.3.1.4.1.1 System Description and Model Integration Are Adequate

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the degradation of engineered barriers (degradation of the waste package) with respect to system description and model integration.

DOE documented the approach and technical basis for the abstraction of the degradation of the waste package in total system performance assessment in the process model report (CRWMS M&O, 2000b) and supporting analysis and model reports. The reference waste package design recommended for the proposed site recommendation (CRWMS M&O, 1999a) consists of an outer container of Alloy 22 surrounding an inner 5-cm [1.97-in] thick container made of Type 316 nuclear grade stainless steel. The main purpose of the inner container is to provide structural strength to the waste package. There are several design concepts for spent nuclear fuel and high-level waste glass containers (CRWMS M&O, 2000d), including five different designs for the commercial spent nuclear fuel with the same wall thickness {2 cm [0.79 in]} (CRWMS M&O, 2000e). The length, diameter, and interior of these five designs vary to accommodate fuel assembly variations. The commercial spent nuclear fuel disposal containers will be fabricated in two sizes (21 and 12 pressurized water reactor fuel assemblies) in which neutron absorber plates will be used. An additional waste package design for 21 pressurized water reactor fuel assemblies will contain control rods. The disposal containers for boiling water reactor spent nuclear fuel will be fabricated in two sizes for 44 and 24 fuel assemblies, both using neutron absorber plates. There are two designs that differ in length to hold the U.S. Navy spent nuclear fuel, both consisting of a single canister inside a disposal container with a wall thickness of 2.5 cm [0.98 in]. There are two designs of the codisposal container for DOE-owned spent nuclear fuel and high-level waste glass canisters, that only differ in length, having an outer container wall thickness of 2.5 cm [0.98 in]. These codisposal containers will hold five high-level waste glass canisters surrounding a DOE-owned spent nuclear fuel disposal canister inserted in the center of the container. The third waste package design for the DOE-owned spent nuclear fuel will accommodate two high-level waste glass canisters and two multicanister overpacks containing DOE-owned spent nuclear fuel canisters. A dual closure-lid design has been adopted for the waste package to mitigate against premature failure of the outer container as a result of stress corrosion cracking in the closure weld area. The closure end of the outer container, instead of one lid, has two lids. The inner lid is 1-cm [0.39-in] thick, and the outer lid is 2.5-cm [0.98-in] thick, with a physical gap between the two lids.

The corrosion processes potentially important in the degradation of the waste package outer container such as dry-air oxidation, humid-air and uniform aqueous corrosion, localized (pitting and crevice) corrosion, microbially influenced corrosion, stress corrosion cracking, and hydrogen embrittlement are considered in the process model report (CRWMS M&O, 2000b). The evaluation of features, events, and processes concerning waste package degradation that DOE has included or excluded (CRWMS M&O, 2000f) is described in Section 3.2.1 and incorporated into a features, events, and processes table. In general, there is agreement with DOE regarding the included features, events, and processes. The screening however, arguments and technical basis for several excluded features, events, and processes were not adequate, particularly those related to electrochemical processes and fabrication effects, including initial defects, welding processes, and postweld treatments. As described in Section 3.2.1, features, events, and processes were discussed during two Total System

Performance Assessment and Integration Technical Exchanges in May[1] and August 2001.[2] As a result of the meetings, DOE and NRC agreed on a path forward for each feature, event, and process (see Appendix B for specific details).

Dry-air oxidation is assumed to occur when the relative humidity of the repository environment is less then the critical relative humidity for the initiation of humid-air corrosion (CRWMS M&O, 2000b,g). The rate of dry-air oxidation is modeled assuming mass transport of reacting species limited by diffusion through the tightly adhering passive oxide film that results in a parabolic growth law where the film thickness is proportional to the square root of time. It is concluded that the oxidation rate is low at the waste package temperatures predicted after waste emplacement, and dry-air oxidation does not appear to limit waste package lifetime. For humid-air corrosion, DOE assumes that no water dripping occurs when relative humidity is greater than critical relative humidity. The corrosion rate and the distribution of corrosion rates are the same as for aqueous corrosion and are independent of time (CRWMS M&O, 2000b). The critical relative humidity is based on the deliquescence point (lowest relative humidity at which a saturated solution of the salt can be maintained at a given temperature) for sodium nitrate, which is conservatively assumed to be the salt that prevails on the container surface because it is the most hygroscopic salt that can be precipitated.

Aqueous corrosion is classified into two corrosion modes: general corrosion and localized corrosion. For corrosion-resistant nickel-chromium-molybdenum alloys such as Alloy 22, general corrosion in the expected waste package environments occurs in the form of passive corrosion, whereas localized corrosion is limited to pitting and crevice corrosion. Two conditions are considered to be simultaneously present for stabilization of an aqueous film on the waste package surface leading to aqueous corrosion—relative humidity in the emplacement drift greater than the deliquescence point of any salts deposited on the waste package surface and water dripping on the waste package. Below 100 °C [212 °F] the composition of water that contacts the waste package surface is assumed to be simulated J–13 concentrated water, whereas simulated saturated water is assumed to be present above 100 °C [212 °F]. Basic saturated water also has been identified as another plausible water chemistry that may develop on the waste package surface as a result of dripping and evaporation. The chemical composition of these waters is given in Table 3.3.1-1 (CRWMS M&O, 2000h). Two types of distinctive water chemistries were identified as produced by evaporation in laboratory experiments (CRWMS M&O, 2000i,j,k). Bicarbonate-type waters were generated by evaporation of synthetic J–13 water, whereas chloride-sulfate-type waters were formed by evaporation of pore water. In the bicarbonate-type waters, the ratio of the fluoride to chloride concentration is similar to that of the original J–13 water, however, the chloride concentration reaches values around 0.2 M. In the chloride-sulfate types, the concentration of fluoride is low

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[1]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration–Features, Events, and Processes (May 15–17, 2001)." Letter (May 30) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[2]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

Table 3.3.1-1. Molar Concentration of Key Species in Simulated Concentrated Water, Simulated Saturated Water, and Basic Saturated Water*

Species	Simulated J–13 Concentrated Water (Molar)	Simulated Saturated Water (Molar)	Basic Saturated Water (Molar)
K^+	0.09	3.62	1.77
Na^+	1.78	2.12	4.74
F^-	0.07	0.00	0.07
Cl^-	0.19	3.62	3.82
NO_3^-	0.10	21.1	2.32
SO_4^{2-}	0.17	0.00	0.15
HCO_3^-	1.15	0.00	0.00
pH	—	—	11–13

*CRWMS M&O. "Environment on the Surfaces of the Drip Shield and Waste Package Outer Barrier." ANL–EBS–MD–000001. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000.

but the chloride concentration is in the molarity range. High chloride concentrations are also obtained in the modeling of the in-drift environment, taking into account seepage and thermal-hydrological-chemical coupled processes. These two types of water chemistries can lead to significant differences in the mode and rate of corrosion of waste package materials.

General corrosion is assumed to occur within the range of potentials leading to passive corrosion when the corrosion potential (E_{corr}) is less than the critical potential for the initiation of localized corrosion ($E_{critical}$). No mechanistic model is used to calculate corrosion rates within this regime. General corrosion rates are derived from weight-loss data obtained from the long-term corrosion test facility where numerous test specimens have been exposed to aqueous solutions based on modifications of J–13 water (CRWMS M&O, 2000g; McCright, 1998). Enhancement factors were used to consider the increases in corrosion rate associated with the effect of microstructural changes resulting from thermal treatments or modifications of the environment as a result of microbial activities. An enhancement factor is used to model the corrosion rate of thermally aged Alloy 22. Acceleration of the corrosion rates as a result of microbial activity is also treated using an enhancement factor, G_{MIC}. The condition for the occurrence of microbially influenced corrosion is a threshold relative humidity of 90 percent.

Localized corrosion of Alloy 22 is assumed to occur when the E_{corr} is greater than the $E_{critical}$. Mechanistic modeling of crevice corrosion to calculate spatial distributions of potential and current density, as well as transient calculations of dissolved species, was conducted. However, this deterministic modeling was not used in the model abstraction. Instead, initiation and repassivation potentials, as well as a potential defined by the occurrence of an anodic peak, defined as $E_{critical}$, were obtained in cyclic potentiodynamic polarization tests in a variety of electrolytes based on modifications of J–13 water. The potential for the anodic peak was conservatively selected to define the conditions for localized corrosion. The difference between $E_{critical}$ and E_{corr} for each solution tested was fitted to a function of the absolute temperature, the logarithm of the chloride concentration and the pH. It was found that the difference between $E_{critical}$ and E_{corr} depends on pH, but it does not exhibit any dependence on both absolute temperature and chloride concentration, over the range of conditions tested. Because of the lack of DOE experimental data, the rate of localized penetration of Alloy 22 was estimated from data available in the open literature using corrosion rates obtained in highly corrosive environments such as 10 percent $FeCl_3$ at 75 °C [167 °F]; dilute boiling HCl; and a solution containing 7 vol%, H_2SO_4, 3 vol% HCl, 1 wt% $FeCl_3$ and 1 wt% $CuCl_2$ at 102 °C [216 °F].

Stress corrosion cracking is one of the potential failure modes of the Alloy 22 outer container. DOE proposed two models for the evaluation of stress corrosion cracking susceptibility—the stress corrosion cracking stress intensity threshold model and the slip dissolution/film rupture model (CRWMS M&O, 2000b). The stress corrosion cracking threshold model is based on fracture mechanics concepts that suggest for stress corrosion cracking to occur, the stress intensity (K_I) at a flaw or defect must be equal to or greater than the threshold stress intensity factor for stress corrosion cracking (K_{Iscc}) in the presence of a corrosive environment. The slip dissolution/film rupture model relates crack advance to the metal anodic oxidation that occurs when the protective film at the crack tip is ruptured as a result of a tensile stress. In this model, a simple expression relates the crack propagation rate (V_t) with the crack tip strain rate ($\dot{\varepsilon}_{ct}$) and the crack tip strain rate with K_I, according to a power law relationship (CRWMS M&O, 2000l). For both the slip dissolution/film rupture model and the stress corrosion cracking threshold model, through-wall radial cracking is predicted as a result of the high values of the calculated stress intensity factor. Stress corrosion cracking, however, is limited to the surface area defined by the closure-lid welds. Therefore, the approach adopted by the DOE to mitigate or eliminate the possibility of crack growth is to reduce the residual stresses associated with welding. One method proposed involves the use of laser peening to introduce compressive stresses on the surface using multiple passes of a laser beam (CRWMS M&O, 2000b). This method will be used in the inner closure lid. The other method consists of localized annealing of the weld region using induction heating. This method will be applied to the weld in the outer closure lid.

All the corrosion process models discussed previously are abstracted and integrated in WAPDEG, the waste package degradation code, Version 4.0 (CRWMS M&O, 2000m). WAPDEG is a probabilistic code, incorporated in the Total System Performance Assessment–Site Recommendation, designed to run stochastic simulations in which random values are sampled to represent parameters in the corrosion models for calculating waste package lifetimes.

The description of the waste package, in terms of materials and fabrication processes that influence the consideration of corrosion processes affecting performance is adequate to the current level of design; however, a detailed description of the fabrication sequence and additional information on the effects of fabrication processes (e.g., welding and postweld thermal treatments) on the degradation of the containers will be needed as part of issue resolution. DOE studied the phase stability of Alloy 22, considering the precipitation of secondary topologically close-packed phases, such as μ, σ, and P-phase, which depend on time and temperature, (CRWMS M&O, 2000n). Alloy 22 specimens, exposed to temperatures in the range 427–800 °C [800–1,472 °F] for periods up to 40,000 hours, were analyzed for precipitation of topologically close-packed phases and long-range order. An activation energy for the precipitation of topologically close-packed phases has been determined to be near 280 kJ mol^{-1} [66.9 kcal mol^{-1}]. Based on the results of specimens analyzed thus far, bulk precipitation of topologically close-packed phases is not predicted in 10,000 years at 300 °C [572 °F] (CRWMS M&O, 2000b). The formation of grain boundary precipitates is deemed a worst-case scenario that would be equivalent to a 100-hour exposure at 700 °C [1,262 °F]. Using a similar Arrhenius-type relationship, it is predicted that the long-range order may occur after 1,000 years at 300 °C [572 °F]. No long-range order is predicted if the temperature remains below 260 °C [500 °F], however. Additional data and evaluations are necessary to properly model the effects of welding and thermal aging on the intergranular and crevice corrosion susceptibility of Alloy 22. The additional evaluations should include the effects of variations in base alloy composition, cold work, and water chemistry. In addition, the effects of welding parameters such as welding method, heat input, joint geometry, number of passes, and weld filler metal composition must be considered. DOE agreed[3] to provide updated information on aging, fabrication process, and welding. Detailed clarifications stated here need to be included in the agreed-on information.

In summary, the description of likely corrosion processes is sufficient for NRC to make regulatory decisions at the time of any future license application. Several aspects of modeling and model integration have limitations, however, because they are based on an empirical approach without sufficient mechanistic support. There is no clear integration between modeling of the environment in contact with the waste package, as discussed in detail in Section 3.3.3, and certain corrosion processes (e.g., localized corrosion), taking into account uncertainties in the calculated values of environmental variables such as chloride concentration and pH, among other factors. Additional information will be necessary to complete the evaluation of stress corrosion cracking modeling taking into account the proposed stress mitigation techniques resulting from postweld treatments and the detrimental effect of specific chemical species that may be present in the waste package environment. Most of these comments have been presented in more detail in NRC (2001). The technical bases for these comments are supported by the experimental work conducted at CNWRA, together with an extensive review of the open literature referenced in NRC (2001). Agreements reached with DOE regarding these comments are also documented in that report and summarized in Section 3.3.1.5. With the DOE agreement to provide the additional information, sufficient information should be available at the time of a potential license application for NRC to make a regulatory decision.

[3]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000)." Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

3.3.1.4.1.2 Data Are Sufficient for Model Justification

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the degradation of engineered barriers (degradation of the waste package) with respect to data being sufficient for model justification.

There are not enough data available for an accurate evaluation of dry-air oxidation and humid-air corrosion, but the data DOE used seem to be sufficient to bound the expected behavior. The assumption of parabolic growth of oxides on stainless steel and nickel-chromium-molybdenum alloys is not supported by either DOE data or independent tests performed outside the high-level waste disposal program (NRC, 2001). Parabolic oxidation kinetics, however, result in greater oxide penetration compared with either logarithmic or inverse logarithmic kinetics (Fehlner, 1986). At the temperatures expected for the proposed repository, complete oxide penetration of the Alloy 22 outer container by uniform oxidation is not expected. Physical processes that lead to accelerated oxidation rates, such as spalling or mechanical abrasion of the oxide layer, are not expected either. The DOE assumption of parabolic oxidation of Alloy 22 is bounding but should be supported by empirical evaluations of Alloy 22 and similar nickel-chromium-molybdenum alloys. An evaluation of the possibility of preferential oxidation at grain boundaries would be desirable based on the apparent susceptibility of nickel-base alloys to enhanced intergranular oxidation, which has been shown to be a factor in stress corrosion cracking of steam generator tubing (Bruemmer, et al., 2000). To address this issue, DOE agreed[4] to provide information on oxide film growth in air. Detailed clarifications stated here need to be included in the agreed-on information.

The approach used by DOE, assuming that the corrosion rates of Alloy 22 under humid-air conditions are the same as those for aqueous conditions, appears to be conservative. A comparison of aqueous and humid-air corrosion rates for Type 316L stainless steel (CRWMS M&O, 2000b) reveals that the humid-air corrosion rates are almost one order of magnitude less than the aqueous corrosion rates and thus supports the DOE approach.

General corrosion rates of Alloy 22 specimens exposed in the long-term corrosion test facility were calculated by measuring the weight loss of the specimens (American Society for Testing and Materials, 1997) after exposures of 6, 12, and 24 months. Weight gain was observed on 25 percent of the Alloy 22 specimens as a result of the deposition of silica (assumed to be amorphous SiO_2) on specimen surfaces. Data from specimens with weight gains were excluded from the distribution of corrosion rates that is equal to 0 nm/yr at the 0^{th} percentile, 27 nm/yr [1.06×10^{-3} mpy] at the 50^{th} percentile, 98 nm/yr [3.86×10^{-3} mpy] at the 90^{th} percentile, and 730 nm/yr [2.87×10^{-2} mpy] at the 100^{th} percentile. The distribution includes data from tests in a variety of solutions derived from J–13 water and included tests at 60 and 90 °C [140 and 194 °F], but is restricted to the 2-year exposure data. The abstracted general corrosion rate for the Alloy 22 outer container was found to be distributed between

[4]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000)." Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

10^{-6} and 7.3×10^{-5} mm/yr [3.9×10^{-5} and 2.9×10^{-3} mpy]. It was suggested, based on atomic force microscopy measurements, that the entire corrosion rate distribution can be corrected to take into account the weight gain caused by the deposited silicate by adding a value of 63 nm/yr [2.5×10^{-3} mpy] to the measured rates (CRWMS M&O, 2000b,g). The resulting distribution, that DOE defined as an alternative conservative model for waste package general corrosion, ranged from 4.0×10^{-6} to 1.8×10^{-4} mm/yr [1.6×10^{-4} to 7.1×10^{-3} mpy].

An enhancement factor, uniformly distributed between 1 and 2.5, is used to account for the corrosion rate of thermally aged Alloy 22. The value of the factor is based on the passive current density of the thermally aged specimen {700 °C [1,292 °F] for 173 hours} compared with that of an annealed specimen, both measured in potentiodynamic polarization tests (CRWMS M&O, 2000b).

The enhancement factor G_{MIC} is used to account for the acceleration of the corrosion rates as a result of microbial activity. For Type 316 nuclear grade stainless steel, a value of 10 is used for G_{MIC}, based on results obtained with Type 304 stainless steel. For Alloy 22, experimental results indicate a G_{MIC} of 2, based on the corrosion rate measured in short-term exposure tests (CRWMS M&O, 2000b,g). A value of G_{MIC} uniformly distributed between 1 and 2.0 is used in WAPDEG.

The distribution of localized corrosion rates is centered around the highest passive current density of 10 μA/cm^2 [9.2×10^{-4} A/ft^2] that corresponds to a corrosion rate of 100 μm/yr [3.94 mpy]. The cumulative distribution of penetration rates for localized corrosion is equal to 12.7 μm/yr [0.5 mpy] for the 0th percentile, 127 μm/yr [5 mpy] for the 50th percentile, and 1,270 μm/yr [50 mpy] for the 100th percentile (CRWMS M&O, 2000b,g).

For the stress corrosion cracking of Alloy 22, crack propagation rates ranging from 2.1×10^{-11} to 7.6×10^{-12} m/s [8.27×10^{-10} to 3.0×10^{-10} in/s] were measured using a compact tension specimen at K_I = 30 MPa·m$^{1/2}$ [27.3 ksi·in$^{1/2}$] in an air-saturated alkaline solution (pH 13.4) with a composition similar to basic saturated water (Table 3.3.1-1) at 110 °C [230 °F] after a 3,585-hour exposure. These crack growth rates were used to determine the value of the repassivation parameter *n* (CRWMS M&O, 2000l). The parameter *n* is the exponent in the expression relating crack velocity with K_I in the slip dissolution/film rupture model. Because of the lack of sufficient data, the preexponential parameter *A* was considered to be equal to that reported for austenitic stainless steels in boiling water reactor environments. Assuming such a value for *A*, values of *n* ranging from 0.843 to 0.92 were then calculated from the measured crack growth rates listed previously. DOE recognizes that the variation of *n* as a function of environmental factors, which is one of the most important parameters in the model, is not available because of lack of experimental data. It should be noted that the range of values measured for *n* is the result of a single test conducted for 3,585 hours. Considering the uncertainty associated with the determination of *n*, values of 0.843 and 0.92 were selected to represent the lower and upper bounds of *n* using a uniform distribution (CRWMS M&O, 2000l). In the case of the stress intensity threshold model, a value of K_{Iscc} equal to 33 MPa·m$^{1/2}$ [30.3 ksi·in$^{1/2}$] was measured in N_2^- deaerated 5-percent sodium chloride acidified to pH 2.7 at 90 °C [194 °F] (CRWMS M&O, 2000l). The value of 33 MPa·m$^{1/2}$ [30.3 ksi·in$^{1/2}$] with a standard deviation of 1.77 MPa·m$^{1/2}$ [1.61 ksi·in$^{1/2}$] was calculated from the results of duplicate tests using double cantilever beam specimens at 4 different initial K_I values ranging from 22 to 43 Mpa·m$^{1/2}$ [20 to 39 ksi·in$^{1/2}$].

In summary, the available data are not sufficient to justify the model abstractions for aqueous corrosion, in particular, for localized corrosion. The corrosion rates for general and localized corrosion, as well as the effect of changes in material conditions from fabrication processes (e.g., cold-working, welding, shop annealing, laser peening, and induction annealing) or environmental modifications as a result of microbial activity, do not include consideration of the complete range of environmental conditions that can be expected in the emplacement drifts. The solutions used in the tests, based on variations of J–13 Well water at 60 and 90 °C [140 and 194 °F], are not consistent with the environments predicted to result from the evolution of near-field processes (see Section 3.3.3). Lack of sufficient data weakens the justification of model abstractions (e.g., range of values assigned to enhancement factors). The enhancement factor for thermally aged specimens, based on limited short-term tests, implies that thermal aging will result only in an increased passive corrosion rate rather than in an increased susceptibility to localized or intergranular corrosion, as noted in other studies (NRC, 2001). The enhancement factor for microbially influenced corrosion, G_{MIC}, was calculated from the results of exposures to sterile and inoculated solutions (CRWMS M&O, 2000b,g). No information is provided on the possible preferential dissolution of alloying elements or on localized corrosion susceptibility as a result of microbial activity. In addition, the effects of temperature and environmental variations (e.g., pH, redox conditions, and ionic species) on the value of G_{MIC} are not available. To address all these concerns, DOE agreed[5] to provide sufficient information on aqueous corrosion, in particular, for localized corrosion. The agreed-on information will also include the effects of fabrication process and microbial activity and all credible environmental conditions.

3.3.1.4.1.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the degradation of engineered barriers (degradation of the waste package) with respect to data uncertainty being characterized and propagated through the model abstraction.

The most important implication of data uncertainty is related to the estimation of the distribution of waste package failure times. The importance of data uncertainty is also related to the contribution of specific corrosion processes to the overall performance and the propagation of data uncertainty in related and interdependent corrosion processes, as discussed next.

As noted in Section 3.3.1.4.1.1, humid-air corrosion is assumed to occur when relative humidity is greater than critical relative humidity. To define the characteristics of the environment in

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[5]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000)." Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

contact with the surface of the waste package and the drip shield (dry versus humid air) and the corresponding corrosion process, the deliquescence point for $NaNO_3$ is used as the criterion for critical relative humidity (CRWMS M&O, 2000h). This choice is not justified, even though the deliquescence point for $NaNO_3$ seems to be the lowest among the salts that may be deposited on the surfaces of the waste package or the drip shield (CRWMS M&O, 2000h) because a mixture of salts usually has a lower deliquescence point than any of the individual salts that form the mixture. A relevant example is the NaCl-$NaNO_3$-KNO_3 system as shown in Table 3.3.1-2. For this system, the deliquescence point of the three salt mixture[6] is significantly lower than that of any of the individual salts. The lower deliquescence point or critical relative humidity implies that the waste package or drip shield may be subject to aqueous corrosion for a longer period of time when the temperature and the concentration of salts are both higher than those predicted. Additional details on this issue are provided in Section 3.3.3. To address this concern, DOE agreed[7] to provide information on the credible environmental conditions. More detailed clarifications stated here need to be included in the agreed-on information.

The DOE assumption of humid-air corrosion rates of Alloy 22 bounded by aqueous corrosion rates is acceptable. It would be useful to have additional data obtained outside the Yucca Mountain Project using information for Alloy 22 and similar nickel-chromium-molybdenum alloys. It appears that the uncertainty in the data will not lead to an erroneous evaluation of the effect of humid-air corrosion on waste package degradation. As the rates of aqueous corrosion are likely to encompass the humid air corrosion, NRC has no additional questions on this issue at this time.

Table 3.3.1-2. Deliquescence Point for Single Salts and Salt Mixtures at 16.5 °C [54.5 °F]

Salt(s)	Deliquescence Point
Pure NaCl	76 percent*
Pure $NaNO_3$	78 percent*
Pure KNO_3	95 percent*
Mixture of the listed salts (with a composition corresponding to a saturated solution of the three salts)	30.5 percent†

*CRWMS M&O. "Environment on the Surfaces of the Drip Shield and Waste Package Outer Barrier." ANL–EBS–MD–000001. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000.
†Weast, R.C., ed. *Handbook of Chemistry and Physics.* 54th Edition. Cleveland, Ohio: CRC Press. 1973.

[6]During the evaporation process, the composition of the dry salt formed as a result of losing the last amount of water would always have a composition corresponding to the saturated solution, no matter what the starting solution composition is, as long as it is within the chemical divide.

[7]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000)." Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

For aqueous corrosion, the DOE approach relies on passive dissolution rates of Alloy 22 determined via weight loss measurements. Because the passive corrosion rate of Alloy 22 is quite low, the change in mass is also small. For a typical 50- × 50- × 3.175-mm [1.97- × 1.97- × 0.125-in] test specimen with an area of 56.35 cm^2 [8.74 in^2] and a weight of 68.97 g [0.152 lb], a corrosion rate of 26.6 nm/yr [1.05×10^{-3} mpy] (DOE 50th percentile) is equivalent to a passive current density of 2.6×10^{-9} A/cm^2 [2.42×10^{-6} A/ft^2] or a mass loss rate of 0.00125 g/yr [0.000049 oz/yr]. For a 1-year exposure, the change in weight is less than 2×10^{-3} percent. Such small changes in weight can be determined provided there is not substantial interference from a competing process. In the case of the long-term corrosion test facility data, the deposition of silica was shown to interfere with the weight-loss data. The suggested correction (CRWMS M&O, 2000b,g) to the corrosion rate distribution {e.g., addition of 63 nm/yr [2.5×10^{-3} mpy]} may lead to a nonconservative estimation of the actual corrosion rates by overcorrecting the measured rates because the estimation does not account for the time-dependent changes in corrosion rate that must have occurred after the silica deposition. In addition, the value of the correction factor is more than twice the value of the median corrosion rate. An additional factor to consider is the use of a distribution in the corrosion rates that tends to give excessive weight in the computations of waste package life to the lowest corrosion rates within the distribution. On the contrary, the highest corrosion rates measured, if not accounted for in the distribution, would lead to container failure times much shorter that those currently predicted in the Total System Performance Assessment–Site Recommendation. To address this concern, DOE agreed[8] to provide justifications on the accurate measurements of corrosion rates, and their extrapolation and abstraction.

Higher corrosion rates have been observed for nickel-chromium-molybdenum alloys similar to Alloy 22 in a variety of environmental conditions relevant to the Yucca Mountain Project. Smailos (1993) reported corrosion rates of Alloy C–4 in brine environments containing 25.9 percent sodium chloride at 150 °C [302 °F] calculated, from weight loss measurements after 18-month exposures, to be in the range from 6×10^{-5} to 7×10^{-5} mm/yr [2.4×10^{-3} to 2.8×10^{-3} mpy]. In brines with 26.8 and 33 percent $MgCl_2$, the welded Alloy C-4 had a corrosion rate of 0.005 to 0.006 [0.2 to 2.4 mpy] . Bickford and Corbett (1985) measured corrosion rates of Alloy 22 in environments containing 20,000-p/m Cl^-; 2,300-p/m F^-; and 1,400-p/m SO_4^{2-}. In solutions with a pH of 1.6, the corrosion rates were 5 mm/yr [2 mpy] at 40 °C [104 °F] and 5 mm/yr [2 mpy] at 90 °C [194 °F], whereas, in solutions with pH 6, the corrosion rates were 5 mm/yr [2 mpy] at 40 °C [104 °F] and 0.012 mm/yr [0.47 mpy] at 90 °C [194 °F]. Harrar, et al. (1977, 1978) reported the corrosion rates of Alloys C-276 and 625 exposed to chloride containing groundwater at the Salton Sea geothermal field {100 °C [212°F] brine containing 12-percent chloride at a pH of 3.4}. General corrosion rates calculated using linear polarization were 0.0015 mm/yr [5.9×10^{-5} mpy] for Alloy C-276 and 0.007 mm/yr [2.8×10^{-4} mpy] for Alloy 625. In summary, the distribution of corrosion rates used for DOE in the WAPDEG calculations is lower than data reported in the literature, in some cases by more than one order of magnitude, for environments that appear to be relevant to the repository

[8]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000)." Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

conditions. To address this concern, DOE agreed[9] to provide justifications on the accurate measurements of corrosion rates, and their extrapolation and abstraction.

The relative corrosion rates of welded and base metal Alloy 22 were also determined using weight-loss specimens. Although the welded specimens are exposed along with the base alloy, the area of the welded region is quite small {approximately 10–15 cm^2 [1.6–2.35 in^2] and accounts for less than 25 percent of the total specimen-surface area. As a result, any accelerated corrosion rate of the welded region would be masked by the much larger area of the base alloy. To address this concern, DOE agreed[10] to use a larger surface area in corrosion testing, including welded samples cut from mockups.

The enhancement factor for the thermally aged specimens is based solely on short-term data and does not consider the effects of preferential corrosion that may occur at the grain boundary regions as indicated in previous investigations (Heubner, et al., 1989). Reductions in the $E_{critical}$ value are a strong indication that thermal aging increases the susceptibility of the alloy to localized corrosion, and more appropriate values of $E_{critical}$, such as crevice corrosion initiation and repassivation potentials, are necessary for a proper evaluation of thermal aging effects on localized corrosion. The increased current density, measured during an anodic polarization scan of an Alloy 22 specimen thermally aged for 173 hours at 700 °C [1,292 °F], was averaged over the entire exposed surface area. In light of the increased susceptibility of thermally aged nickel-chromium-molybdenum alloys to intergranular corrosion, the increased current density observed in the DOE test may be the result of preferential dissolution at grain boundaries rather than an overall increase in the corrosion rate. Such preferential attack, mainly confined to the grain boundary regions, would result in a true enhancement factor much greater than 2.5. To address this concern, DOE agreed[11] to provide updated information on the effects of thermal aging on corrosion.

Uncertainty in the data for the general corrosion rate of Alloy 22 also applies to the effects of long-term changes on the chemical composition and stability of oxide films. Previous investigations indicated that the composition of the passive oxide film becomes enriched in chromium and depleted in molybdenum and nickel (NRC, 2001). The long-term effects of preferential dissolution of alloying elements may include changes in the oxide film composition that, in turn, may alter the passive corrosion rate or promote susceptibility to localized corrosion. Information on the preferential dissolution of alloying elements has not been obtained from the specimens tested in the long-term corrosion test facility. To address this concern, DOE agreed[12] to provide information on the long-term behavior of passive films.

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[9]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000)." Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[10]Ibid.

[11]Ibid.

[12]Ibid.

Localized corrosion rates assumed by DOE, obtained from literature data using acidic chloride and acidic oxidizing chloride solutions, appear to correspond to measured corrosion penetration rates obtained in certain service environments, as reviewed by Cragnolino, et al. (1999). Smailos (1993) reported a maximum pit depth of 0.90 mm [0.035 in] in Alloy 625 after 18 months in 33 percent $MgCl_2$ at 150 °C [272 °F], corresponding to a localized corrosion penetration rate of 0.6 mm/yr [24 mpy]. Carter and Cramer (1974) reported that pit penetration rates for Alloy 625 were 0.22 mm/yr [8.7 mpy] after 45 days in 105 °C [221 °F] brine containing 155,000 p/m chloride with 30-p/m sulfur. Oldfield (1995) observed crevice corrosion of Alloys 625 and C-276 in both natural and chlorinated seawater at ambient temperature. The average penetration rate for Alloy 625 following a 2-year exposure was 0.049 mm/yr [1.9 mpy]. These observations clearly indicate the importance of defining conditions for the initiation and arrest of localized corrosion because these rates of penetration are several orders of magnitude greater than those corresponding to passive general corrosion and also are greater than those selected by DOE for localized corrosion. To address this concern, DOE agreed[13] to provide information on the environmental and electrochemical conditions for localized corrosion.

The DOE modeling of stress corrosion cracking of the Alloy 22 outer container considers a narrow range of expected waste package environments and is limited to the closure lid weld stresses. As noted, two stress corrosion cracking models, the threshold stress intensity model and the slip dissolution/film rupture model, are being used (CRWMS M&O, 2000l). In the first model, stress corrosion cracking susceptibility of Alloy 22 is evaluated using model parameters obtained from Lawrence Livermore National Laboratory data; whereas, in the second case, experimental data obtained at General Electric Corporation for Alloy 22 are combined with data reported for stainless steel in boiling water reactor environments. Evaluation of these two alternative models reveals that while a K_{Iscc} value of 33 MPa·m$^{1/2}$ [30 ksi·in$^{1/2}$], determined by Roy, et al. (1998), is adopted in the threshold model, the slip dissolution/film rupture model predicts crack propagation at K_I values less than the experimentally determined value of K_{Iscc}. It is claimed, however, that the General Electric Corporation data were obtained during cyclic loading conditions rather than constant load. Crack propagation rates for Alloy 22 are found to be extremely low, and the absence of crack growth under constant load conditions was confirmed experimentally.

The residual stress analyses performed by DOE, using a finite element method, indicate that given the calculated maximum stress intensity factors from weld residual stress and a K_{Iscc} determined by Lawrence Livermore National Laboratory, a radially oriented flaw perpendicular to weld may initiate stress corrosion cracking of the Alloy 22 outer container. In contrast, no stress corrosion cracking initiation at a circumferentially oriented flaw parallel to weld is expected based on the threshold value. These arguments are based on the threshold or minimum stress intensity criterion. K_{Iscc}, however, could be lower in a different environment than that tested (Speidel, 1981). The validity of K_{Iscc} as a bounding parameter for performance should be assessed through an appropriate combination of experimental and modeling work. K_{Iscc} values ranging from approximately 8 to 20 MPa·m$^{1/2}$ [7.3 to 18.2 ksi·in$^{1/2}$] have been

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[13]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000)." Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

observed for Types 304, 304L, 316, and other similar austenitic stainless steels in chloride-containing solutions at temperatures ranging from 80 to 130 °C [176 to 266 °F] (Cragnolino and Sridhar, 1992). As expected, the values in the lower end of that range are observed with both increasing temperatures and chloride concentration. It is also recognized that K_{Iscc} values are affected by the electrode potential. On the basis of these observations, it is apparent that the composition of the environment is another constraint that must be considered when using K_{Iscc} as a bounding parameter for the initiation of stress corrosion cracking. To address this concern, DOE agreed[14] to provide stress corrosion cracking data for credible environmental conditions.

The effects of waste package fabrication processes (e.g., welding and heat treatments) on stress corrosion cracking of candidate container materials still remain major concerns. Residual stresses from waste package fabrication or applied stresses resulting from seismic events combined with the necessary environmental conditions may be sufficient to cause stress corrosion cracking of the outer container. If high residual stresses result from fabrication processes, the mechanical component necessary for stress corrosion cracking may be present in every waste package placed in the repository. As noted, DOE proposed postweld treatments to mitigate the effect of residual stresses. The effects of welding and postweld heat treatments on the stress corrosion cracking susceptibility of Alloy 22, as well as the respective K_{Iscc} values in the expected waste package environment, have not been evaluated. Additionally, the DOE stress corrosion cracking models consider weld residual stress the only source of stresses significant to stress corrosion cracking (CRWMS M&O, 2000b,l). Other sources of stress are assumed to be either insignificant such as dead load stress or temporary like seismic stress. Accordingly, the effects of other possible types of applied stresses in the repository have not been assessed. In particular, stresses generated at the line of contact of the waste package with the emplacement pallet should be evaluated. To address this concern, DOE agreed[15] to provide updated information on metallurgical conditions for stress corrosion cracking and its mitigation processes. More detailed clarifications stated here need to be included in the agreed-on information.

Data used to analyze the effects of initial defects on the performance of the waste package outer barrier (CRWMS M&O, 2000o) have uncertainties that have not been characterized nor propagated through the model abstraction. DOE estimates of the probabilities for initial defects in the waste package from various sources range from 10^{-8} to 10^{-3} per waste package. In the specific case of weld flaw, the probability of initial through-wall defect {e.g., defect size larger than 20 mm [0.79 in]} is estimated to be less than 10^{-11} per waste package for the top lid closure weld of Alloy 22. The consequence of this initial flaw is calculated as stress corrosion cracking growth. The effects of initial defects on other corrosion and mechanical failure processes were also ignored. Although surface intersecting flaws are more important for stress corrosion cracking than completely enclosed flaws, the stress and strain localization from the latter may adversely affect stress corrosion cracking, depending on the size and location of the

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[14]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000)." Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[15]Ibid.

flaw. Additionally, if one of the sources of defect is mis-heat treatment, the potential lowering of fracture toughness parameters because of precipitation of embrittling phases (μ-phase in Alloy 22), in combination with internal flaws and residual stresses, can cause mechanical fracture of the container as discussed in Section 3.3.2. To address this concern, DOE agreed[16] to information on stress corrosion cracking covering a full range of metallurgical conditions. Detailed clarifications stated here need to be included in the agreed-on information.

In the application of the slip dissolution/film rupture model to Alloy 22, DOE adopted values ranging from 0.843 to 0.92 for the repassivation slope, *n* (CRWMS M&O, 2000l). This range of values for *n* was calculated from a single experiment conducted for 3,385 hours during cyclic loading conditions R = 0.5–0.7, with frequency 0.001–0.003 Hz, at a maximum K_I = 30 MPa·m$^{1/2}$ [27.3 ksi·in$^{1/2}$]. Input for the model includes average crack growth rates ranging from 2.1×10^{-11} to 7.6×10^{-12} m/s [8.3×10^{-10} to 3.0×10^{-10} in/s] and the empirical relationship adopted from the work of Ford and Andresen (1988) on the stress corrosion cracking of austenitic stainless steels in boiling water reactor environments as previously reviewed by Sridhar, et al. (1993), in the empirical relationships developed by Ford and Andresen (1988), the two interdependent model parameters (*n* and *A*) used to define the crack propagation rate/crack tip strain rate relationship in the slip dissolution/film rupture model are dependent on material properties and the environment at the crack tip. From analysis of the extensive work conducted by Ford and Andresen (1988), it can be concluded that most of the final expressions for calculating crack propagation rates and crack tip strain rates requires the input of field data to adjust several of the parameters included in the model. This is particularly true in the case of the parameter *n*, but also applies to the preexponential coefficient *A*. The model parameters in the slip dissolution/film rupture model are largely empirical correlations on the basis of a combination of laboratory experimental results and field observations. Therefore, application of these empirical relationships to Alloy 22 requires a more complete database to limit propagation of the uncertainty characterizing currently available data into the modeling of stress corrosion cracking of Alloy 22. To address this concern, DOE agreed[17] to provide sufficient data on relevant parameters for stress corrosion cracking models.

Recently, Barkatt and Gorman (2000) reported stress corrosion cracking of Alloy 22 in concentrated J–13 Well water of pH 0.5 (acidified with hydrochloric acid) containing lead at relatively high concentrations (~1,000 p/m). Tests were conducted at 250 °C [452 °F] using U-bend specimens. These test conditions were extremely severe in terms of lead concentrations, temperature, and stress, and the results are preliminary; nevertheless, the possible detrimental effects of impurities such as lead, mercury, or arsenic require further evaluation. If the results are valid for the repository conditions, the current model abstraction for stress corrosion cracking will need reevaluation to account for the effects of these species.

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[16]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000)." Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[17]Ibid.

To address this concern, DOE agreed[18] to provide credible environmental conditions. Detailed clarifications stated here need to be included in the agreed-on information.

This section summarized characterization and propagation of data uncertainties. Various sources of the uncertainties were identified from the involved corrosion processes. They include credible environmental conditions, accurate measurements of corrosion rates, acceptable extrapolation and abstraction of laboratory data, and acceptable conditions for localized corrosion and stress corrosion cracking. The effects of thermal aging, fabrication processes (including welding), and microbial activity on corrosion were also evaluated.

As noted previously, DOE agreed to provide the needed information before any future license application being submitted.

3.3.1.4.1.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the degradation of engineered barriers (degradation of the waste package) with respect to model uncertainty being characterized and propagated through the model abstraction.

The distribution of passive corrosion rates used by DOE is not supported by the electrochemical measurements conducted within the Yucca Mountain project and is lower than corrosion rates measured in a variety of service environments. Combining electrochemical techniques with chemical analysis of alloying elements is a well-established method for measuring passive dissolution rates. The low passive corrosion rate of Alloy 22 is the result of formation of a protective chromium oxide passive film. Kirchheim, et al. (1989) reported a passive current density of 0.014 $\mu A/cm^2$ [1.3×10^{-5} A/ft^2] {corrosion rate of 9.68×10^{-5} mm y^{-1} [3.8×10^{-3} mpy]} for pure chromium in 1 N H_2SO_4. The rates for Ni-Cr-Mo alloys are expected to be higher, even in neutral chloride solutions simulating the aqueous environments contacting waste packages. To address this concern, DOE agreed[19] to conduct appropriate electrochemical tests or provide justification for the approach adopted in the measurements currently being conducted.

In addition, the corrosion rate data used by DOE do not consider the effects of long-term changes to the composition of the oxide films. Previous investigations (Lorang, et al., 1990) indicated that the composition of the oxide film, which acts as a barrier for mass transport, becomes enriched in chromium and depleted in molybdenum and nickel. The long-term effects of preferential dissolution of alloying elements may include changes to the oxide film composition that could, in turn, alter the passive corrosion rate or promote an increase in the susceptibility of the alloy to localized corrosion. Information on the preferential dissolution of

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[18]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000)." Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[19]Ibid.

alloying elements has not been obtained from long-term corrosion test facility specimens. To address this concern, DOE agreed[20] to provide information on the long-term behavior of passive films.

Determination of the Alloy 22 localized corrosion susceptibility by comparing the corrosion potentials and critical potentials measured in short-term tests may not be acceptable. Selection of the $E_{critical}$ should be based on the most likely corrosion mode for the alloy and must consider the environmental effects of temperature, solution chemistry, and the presence of microbes, as well as the effects of material property variations caused by fabrication, welding, thermal aging, and long-term evolution of the oxide film composition and characteristics. In addition, the range of environmental effects such as radiolysis and water chemistry, material factors such as formation of thermal oxide films, and the long-term evolution of the oxide film composition should be included in the bounding analyses of the E_{corr}. The present set of data used as criteria to evaluate the localized corrosion susceptibility of the outer container, as referenced in CRWMS M&O (2000b), is limited to $E_{critical}$ obtained in short-term tests. Confirmatory tests designed to determine the validity of the $E_{critical}$ approach seem to be necessary. To address this concern, DOE agreed[21] to provide information on the electrochemical and environmental conditions for localized corrosion.

Determination that the localized corrosion susceptibility of Alloy 22 is not affected by thermal aging based on the difference between the E_{corr} and the $E_{critical}$ may be nonconservative. The selected value of the $E_{critical}$, which may be a combination of pit initiation, transpassive dissolution and oxygen evolution, is misleading because it does not compare other possible values of $E_{critical}$ such as the initiation and repassivation potentials for crevice corrosion with E_{corr}. Reduction of the pit initiation potential observed for the thermally aged specimen is a strong indication that thermal aging reduces the localized corrosion susceptibility of Alloy 22. Previous investigations identified the formation of topologically closed-packed phases in both thermally aged (Heubner, et al., 1989) and welded (Cieslak, et al., 1986) Alloy 22. Observations of preferential initiation of localized corrosion in weldments and grain boundary attack of the thermally aged material (Heubner, et al., 1989), as well as a lower critical pitting temperature for welded Alloy 22 (Sridhar, 1990), do not support the DOE conclusion of no reduced susceptibility to localized corrosion after thermal aging. Reduction of the E_{corr} after thermal aging suggests an increase in the passive current density. As previously indicated, this increase may be a result of significantly enhanced dissolution at grain boundaries. To address this concern, DOE agreed[22] to provide evaluation of metallurgical conditions affecting localized corrosion, especially for thermally aged samples.

In addition to environmental effects, the DOE evaluation of the stress corrosion cracking susceptibility of Alloy 22 should consider the effects of variations in material properties,

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[20]Schlueter, J.R. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000).” Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[21]Ibid.

[22]Ibid.

fabrication and welding, and long-term exposure to elevated temperatures. These variations are not easily correlated with compositional variations or differences in mechanical properties. Segregation of alloying elements and the formation of topologically close-packed phases in the welded regions has been shown to occur for Alloy 22 (Cieslak, et al., 1986), and thermal aging has been shown to increase localized corrosion susceptibility (Heubner, et al., 1989). Long-term exposure of the waste package to elevated temperatures expected in the proposed repository may result in microstructural alterations that may be equivalent to aging for 100 hours at 700 °C [1,292 °F] (CRWMS M&O, 2000b). To address this concern, DOE agreed[23] to provide acceptable evaluation of metallurgical conditions for stress corrosion cracking, especially for thermally aged samples.

This section summarized characterization and propagation of model uncertainty. The sources of the uncertainties included the long-term behavior of passive film, and conditions for localized corrosion and stress corrosion cracking.

As noted previously, DOE agreed to provide the needed information before any future license application is submitted.

3.3.1.4.1.5 Model Abstraction Output Is Supported by Objective Comparisons

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the degradation of engineered barriers (degradation of the waste package) with respect to model abstraction output being supported by objective comparisons.

DOE data for the corrosion rates of Alloy 22, obtained in the long-term corrosion test facility, are not reliable because of the deposition of silica and the limitations of the weight loss measurements to evaluate the effects of welding. Additional tests, where interference from deposition processes do not occur, should be performed to confirm or correct the results obtained using long-term corrosion test facility specimens. Determination of passive corrosion rates from weight loss may be possible in solutions that do not contain dissolved silica, divalent cations such as calcium, or other species that can precipitate from solution and deposit on the test specimens. As an alternative to weight loss, steady-state anodic current density measurements obtained under potentiostatic conditions can be used to determine corrosion rates according to American Society for Testing and Materials G102 (American Society for Testing and Materials, 1999). A more substantiated discussion about the long-term validity of low passive corrosion rates of Alloy 22 needs to be justified using an appropriate combination of testing and calculations. The use of source data in the models appears to be inconsistent. To address this concern, DOE agreed[24] to provide accurate corrosion data and its acceptable extrapolation.

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[23]Schlueter, J.R. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000).” Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[24]Ibid.

Although the K_{Iscc} value determined by Roy, et al. (1998) is adopted in the threshold model, different source data for the crack growth rate are used in the slip dissolution/film rupture model. Not only the data have been obtained using different types of fracture mechanics specimens and test methods, but the environments are widely different in chemical composition, pH, and redox potential. In addition, the environments used to evaluate the stress corrosion cracking susceptibility of Alloy 22 using the stress corrosion cracking threshold model are not consistent with the environments expected on the drip shield and waste package (CRWMS M&O, 2000h). K_{Iscc} values used to determine stress corrosion cracking susceptibility should be based on measurements conducted in environments that may be expected in the proposed repository because K_{Iscc} values are strongly dependent on both the material and the environment (Speidel, 1981). At present, the slip dissolution/film rupture model for Alloy 22 uses a combination of parameters derived from stainless steel in boiling water reactor environments (Ford and Andresen, 1988; Ford, 1990) and limited amount of data obtained from laboratory tests (CRWMS M&O, 2000l). Although the model is theoretically based on fundamental parameters such as the repassivation rate, in practice, the critical parameters are empirically derived using a substantial volume of data obtained in boiling water reactor environments (Ford and Andresen, 1988; Ford, 1990) that are not available for Alloy 22 in the expected waste package environments. To address this concern, DOE agreed to provide supporting data bases for stress corrosion cracking models. Detailed clarifications stated here need to be included in the agreed-on information.

The effects of the postweld annealing treatment proposed for the dual lid waste package outer container on the stress corrosion cracking susceptibility of Alloy 22 should also be evaluated. The proposed annealing treatment relies on rapid heating and cooling cycles (CRWMS M&O, 2000b). Because only the end of the waste package is elevated to temperatures beyond 1,000 °C [1,802 °F], significant thermal gradients will exist that may result in the exposure of some portions of the waste package outer barrier to temperatures that favor the formation of detrimental topologically close-packed phases. Variations in the annealing parameters may exacerbate microstructural alterations and further reduce the stress corrosion cracking resistance of the alloy. There is no specific experience on laser peening of Alloy 22. To address this concern, DOE agreed[25] to provide additional information on postwelding processes for mitigating stress corrosion cracking.

Section 3.3.1.4.1.5 addresses uncertainties associated with accurate determinations of uniform corrosion rates, insufficient data base and rationales in using existing stress corrosion cracking models, and insufficient evaluation of welding effects on stress corrosion cracking.

As noted previously, DOE agreed to provide the needed information before any future license application is submitted.

[25]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000)." Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

3.3.1.4.2 Degradation of the Drip Shield

For undisturbed repository conditions, corrosion is also considered the primary degradation process of the drip shield. Because the drip shield was not included as an engineered barrier subsystem design feature in the viability assessment, there is only a single calculation showing the beneficial effect of the drip shield on waste package life and dose in DOE (1998a). In recent performance assessment sensitivity analysis calculations for the site recommendation, the beneficial effect of the drip shield on the predicted annual dose rate is only apparent after 50,000 years (CRWMS M&O, 2000a).

3.3.1.4.2.1 System Description and Model Integration Are Adequate

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.6.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the degradation of engineered barriers (degradation of the drip shield) with respect to system description and model integration.

DOE has documented the approach and technical basis for the abstraction of the degradation of the drip shield in total system performance assessment in a process model report (CRWMS M&O, 2000b) and supporting analysis and model reports. Use of a drip shield as a design option is intended to minimize the possibility of water dripping on containers. Corrosion of the containers can be enhanced by the presence of flowing liquid water that may facilitate localized penetration if the chemical composition of the water is sufficiently aggressive. In addition, liquid water can mobilize and advectively transport most radionuclides. Although moisture condensation between the waste package and the drip shield cannot be prevented, the purpose of the drip shield is to reduce water contact arising from fracture flow. Where active flowing fractures in the repository are coupled with sharp drift wall edges, seeps (drips) into the drift can occur. The principal function of the drip shield is to divert these drips from the waste package surface. The site recommendation design calls for an inverted U-shaped drip shield to be constructed with 1.5-cm [0.59-in]-thick Titanium Grade 7 (Ti-0.15Pd) or Grade 16 (Ti-0.05Pd) plates and structural members made of Titanium Grade 24 (Ti-6Al-4V-0.15Pd) for long-term structural support (CRWMS M&O, 2000p). The drip shield will be extended throughout the length of the emplacement drifts to enclose the top and sides of the waste package and will rest on top of the drift invert made of steel beams and filled up with ballast. The emplacements drifts will have steel sets and lagging (or, in some cases, rock bolts and mesh) for ground support instead of the concrete liner proposed in the viability assessment design.

The DOE approach consists of examining the possible environments to which the drip shield may be exposed (e.g., temperature and chemistry of incoming water) and evaluating the effects of these conditions on the possible degradation modes and rates for palladium-bearing titanium alloys. Degradation modes considered (CRWMS M&O, 2000b) include thermal embrittlement, dry-air oxidation, humid-air corrosion, uniform aqueous corrosion, localized (pitting and crevice) aqueous corrosion, and environmentally assisted cracking (consisting of stress corrosion cracking and hydrogen embrittlement or hydride-induced cracking).

The possibility for thermal embrittlement of titanium used in drip shield construction was excluded for further analysis because thermal embrittlement was considered to have a low probability of occurrence in the features, events, and processes analysis (CRWMS M&O, 2000f), discussed in Section 3.2.1. Mechanical degradation and collapse of the emplacement drifts, with potential effects on the temperature of the drip shield and moisture flow into the engineered barrier subsystems was also screened out (CRWMS M&O, 2000q). This type of drift degradation event may have an important effect on the integrity of the drip shield and should be considered, as discussed in detail in Section 3.3.2. To address this concern, DOE agreed[26] to provide information on the embrittlement of drip shield materials. Detailed clarifications stated here need to be included in the agreed-on information, especially related to the drip shield fabrication. DOE also agreed to provide sufficient information on the mechanical degradation of drip shields and the effects of drift collapse.

Environmentally assisted cracking was examined considering two main processes: stress corrosion cracking and hydride-induced cracking. The process model report, corresponding analysis and model reports, and other technical documents (CRWMS M&O, 2000b,l,r,s) made a clear distinction between stress corrosion cracking and hydride-induced cracking. Within this framework, the only viable source of stress needed for stress corrosion cracking results from rockfall because it is stated that the drip shield will be fully annealed after welding to minimize residual stresses. Two different models for evaluating stress corrosion crack propagation were considered—the stress intensity threshold model and the slip dissolution-film rupture model. The approach taken by DOE to evaluate hydride-induced cracking is based on the assumption that the dominant cathodic reaction occurring on the metal surface during passive (uniform) dissolution is hydrogen evolution, and it is assigned a reaction rate equal to the passive dissolution rate calculated from weight-loss coupon testing. Of the hydrogen gas produced from this cathodic reaction, a fraction (between 0.02 and 0.10) is postulated to enter into the metal as hydrogen atoms and precipitate as hydrides, which may then lead to a loss in ductility (e.g., hydride embrittlement). Hydride-induced cracking is said to be possible once a critical hydrogen concentration has been exceeded. Based on the uniform corrosion rates calculated from weight-loss coupon testing and assumptions regarding the fraction of hydrogen eventually absorbed into the metal lattice, it was concluded that hydride-induced cracking does not have a significant effect on the drip shield life expectancy during the 10,000-year performance period.

Additional examination of possible galvanic interactions with iron-based components in the repository (e.g., rock bolts, steel supports, and gantry rail) led DOE to suggest that only localized areas of galvanic interaction were possible. Given that the cathode (drip shield) to anode (steel component) area ratios would be large, it is assumed that any hydrogen produced would be mostly absorbed in a large volume of titanium such that the concentration would be low. In any event, the consequence for both stress corrosion cracking and hydride-induced cracking was considered to be low because any cracks that developed would be plugged by corrosion products and, therefore, would not be available for the transport of water and subsequent dripping onto the waste package.

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[26]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000)." Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

The dry-air oxidation and the humid-air and aqueous corrosion processes of the drip shield are integrated in WAPDEG (CRWMS M&O, 2000m) and the model abstraction used for these processes is identical to that used for calculating the lifetime of the waste packages. However, a logarithmic growth law was considered as appropriate for the dry-air oxidation of titanium alloys instead of the parabolic law used for the outer waste package container (CRWMS M&O, 2000b,t). A similar criterion to that used in the case of the waste package was applied for the initiation of humid-air and general aqueous corrosion. The general corrosion rates used for these two processes were derived from weight-loss data obtained from the long-term corrosion test facility using Titanium Grade 16 instead of Titanium Grade 7.

As for Alloy 22, localized corrosion of titanium alloys is assumed to occur when the E_{corr} is greater than the $E_{critical}$. Only crevice corrosion is considered because pitting corrosion is disregarded as a plausible degradation process because it was not observed in the long-term corrosion test facility tests. Initiation and threshold potentials were obtained in cyclic potentiodynamic polarization tests in a variety of electrolytes based on modifications of J–13 Well water. The difference between $E_{critical}$ and E_{corr} for each solution tested was plotted as a function of temperature. The difference between $E_{critical}$ and E_{corr} was sufficiently large to preclude the occurrence of crevice corrosion for the range of conditions tested.

In summary, the description of the drip shield materials is adequate for consideration of the corrosion processes affecting performance; however, many details regarding fabrication (e.g., welding, postweld treatments) will be needed for performance assessment at the time of license application. Whereas the description of likely corrosion processes is sufficient, many aspects of model abstraction and integration have limitations. Uncertainties in the composition of the water contacting the drip shield (e.g., fluoride content) may have a significant effect on performance of the drip shield and its expected function. To address these concerns, DOE agreed[27] to provide sufficient information on detailed fabrication processes, model abstraction and integration of corrosion processes, and credible environmental conditions including the composition of the contacting water (e.g., fluoride content).

3.3.1.4.2.2 Data Are Sufficient for Model Justification

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the degradation of engineered barriers (degradation of the drip shield) with respect to data being sufficient for model justification.

There are not enough data available to accurately evaluate dry-air oxidation and humid-air corrosion of the drip shield, but the data DOE used seem sufficient for bounding the expected behavior. According to the waste package degradation process model report and the general and localized corrosions of the drip shield analysis and model report (CRWMS M&O, 2000b,t), Titanium Grade 16 coupons were exposed for 1 year to several aqueous solutions that were

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[27]Schlueter, J.R. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000).” Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

variants of J–13 Well water. Tests showed there was little influence of temperature from 60 to 90 °C [140 to 164 °F] nor was there a significant influence of the testing environment. A wide variation in the measured weight loss, resulting in corrosion rates of ~ −1,700 to 150 nm/yr [6.7×10^{-2} to 5.9×10^{-3} mpy], was reported, however. It is apparent from the negative values that the data include specimens exhibiting significant weight gain. The variability was explained as resulting from differences in the postexposure cleaning procedures used to remove corrosion product buildup. Similar tests were conducted using creviced specimens with no significant attack observed under the crevice former. In this case, rates ranging from −350 to 350 nm/yr [-1.4×10^{-2} to 1.4×10^{-2} mpy] were calculated. Because it was noted that the corrosion rates were similar for the uniform corrosion coupons and the crevice corrosion coupons, it was assumed that the main corrosion mode for the creviced specimens was also uniform passive corrosion of the exposed surfaces. Data from specimens exhibiting weight gain were excluded from the cumulative distribution function of corrosion rates. Based on the maximum corrosion rates observed {350 nm/yr [1.4×10^{-2} mpy] for creviced specimens}, it was concluded that failure of titanium alloy drip shields would be unlikely within the 10,000-year performance period.

A limited set of cyclic potentiodynamic polarization experiments was also performed to examine localized corrosion susceptibility. Based on experiments conducted in simulated saturated water at 120 °C [218 °F] and in simulated J–13 concentrated water at 90 °C [164 °F] (the nominal compositions for these solutions are shown in Table 3.3.1-1), no localized corrosion was noted even when polarization was conducted to 2.5 $V_{Ag/AgCl}$. A critical threshold potential was observed in the polarization scans near 1 $V_{Ag/AgCl}$ and was believed to be associated with oxygen evolution (CRWMS M&O, 2000t).

In summary, the available data, although sufficient for justification of the uniform corrosion model abstraction, do not incorporate the effects of fabrication processes nor the complete range of environmental conditions that can be expected in the emplacement drifts. In particular, the potential detrimental effect of fluoride anions in accelerating the dissolution rate of titanium alloys above a certain threshold concentration (Brossia and Cragnolino, 2001a,b) is not considered. Furthermore, the possible increase of hydrogen uptake by Titanium Grade 7 in the presence of fluoride leading to enhanced susceptibility to hydride-induced cracking has not been evaluated. To address these concerns, DOE agreed[28] to provide sufficient information on detailed fabrication processes and credible environmental conditions, including the composition of the contacting water. In particular, DOE agreed to address the potential detrimental effect of fluoride anions leading to accelerating the drip shield dissolution and hydrogen uptake/hydride cracking of drip shields from the accelerated dissolution.

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[28]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000)." Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

3.3.1.4.2.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the degradation of engineered barriers (degradation of the drip shield) with respect to data uncertainty being characterized and propagated through the model abstraction.

As is the case of the waste package outer container, the most important implication of data uncertainty is related to the estimation of the distribution of drip shield failure times. It should be noted that the maximum error in the determination of corrosion rates from weight-loss measurements in the case of titanium alloy is more than two times that of Alloy 22. The difference can be attributed mostly to differences in density. The main source of uncertainties, however, is related to variation in environmental conditions promoting accelerated corrosion rates.

Though considerable data have been obtained examining the possibility and rates associated with uniform and localized corrosion, several areas of uncertainty still exist. The low corrosion rates measured from weight-loss experiments need to be confirmed with other tests designed to sensitively measure the passive corrosion rate. This confirmation is particularly important because it appears there is an inconsistency between the analysis and model report (CRWMS M&O, 2000t) and the process model report (CRWMS M&O, 2000b). This analysis and model report claims that the weight-loss measurements are at or below the reliable detection limit, yet these values are used for life prediction purposes in the process model report. To address this concern, DOE agreed[29] to provide sufficient data on the uniform corrosion from alternative test methods.

In addition, uncertainties related to the presence of fluoride in the waters contacting the drip shield can lead to much higher rates of uniform corrosion that, in turn, can result in higher absorption rates of hydrogen by the titanium alloys. In this case, the propagation of data uncertainty can affect the evaluation of the potential occurrence of delayed hydrogen cracking as a coupled failure mode. To address this concern, DOE agreed[30] to provide sufficient information on the fluoride concentration of the groundwater in contact with drip shields and its effects on accelerated drip shield corrosion and hydrogen uptake/hydride cracking.

Error propagation from data uncertainties was considered to originate mainly from variations in environmental conditions, low sensitivity in the measurement of uniform corrosion, and possible acceleration of uniform corrosion and hydride embrittlement in the presence of fluoride ions.

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[29]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000)." Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[30]Ibid.

In summary, as noted previously, DOE agreed to provide the needed information before any future license application is submitted.

3.3.1.4.2.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the degradation of engineered barriers (degradation of the drip shield) with respect to model uncertainty being characterized and propagated through the model abstraction.

The corrosion rates measured (approximately 10 to a few hundreds of nanometers per year) using weight-loss methods, especially given the uncertainties concerning cleaning procedures, may be unreliable and nonconservative. Furthermore, in the analysis and model report (CRWMS M&O, 2000t) it was concluded that the majority of the weight-loss measurements during coupon exposure tests were at or below the level of detection. Based on electrochemical corrosion tests, much higher passive dissolution rates were observed (at least a factor of 30 times greater and, in some cases, more than 400 times greater), which could lead to a more conservative estimate of the drip shield life. DOE has not considered alternative models for general passive corrosion. The model used is empirical and based only on the experimental determination of corrosion rates (CRWMS M&O, 2000t). As a result, data uncertainty (note the elimination of data exhibiting weight gain) may render model validation unreliable affecting the confidence to predict life for thousands of years. To address this concern, DOE agreed[31] to provide sufficient data on uniform corrosion from more sensitive and alternative test methods.

This issue is also important in relation to the mechanical disruption of the engineered barriers integrated subissue as described in Section 3.3.2. The effect of rockfall calculations on mechanical failure of the drip shield will be affected by consideration of the drip shield wall thinning because of uniform corrosion and simultaneous hydrogen absorption leading to hydride precipitation and embrittlement of titanium alloys. To address this concern, DOE agreed[32] to provide sufficient information on the effect of wall thinning from corrosion and hydride embrittlement on the mechanical failure induced by rockfall.

The rates of DOE drip shield uniform corrosion are neither consistent among different test methods nor consider alternative models. The inaccurate assessment of uniform corrosion rate will lead to the inaccurate prediction of the drip shield mechanical failure by the thinning of the drip shield wall with the impact of rockfalls.

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[31]Schlueter, J.R. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000).” Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[32]Ibid.

In summary, as noted previously, DOE agreed to provide the needed information before any future license application is submitted.

3.3.1.4.2.5 Model Abstraction Output Is Supported by Objective Comparisons

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the degradation of engineered barriers (degradation of the drip shield) with respect to model abstraction output being supported by objective comparisons.

Though not considered important by DOE, thermal embrittlement of titanium alloys has been reported based on thermally driven redistribution of nearly insoluble impurities from grain interiors to grain boundaries (Nesterova, et al., 1980). This redistribution results in embrittlement of the material with negligible change in strength (though wide variations in ductility are observed) and increased intergranular fracture. Such segregation tends to result in precipitation of finely dispersed particles at the grain boundaries. For commercial purity titanium and α-titanium alloys that contain nickel and iron as impurities, these precipitates have been identified as $Ti_2(Fe,Ni)$. Embrittlement has been noted at temperatures as low as 350 °C [662 °F] after 500 hours. The possibility of embrittlement at lower temperatures when exposed for longer periods has not been examined, however. DOE abstraction analyses of hydrogen embrittlement of titanium alloys could be used to capture any possible effects of thermal embrittlement on predicted drip shield life expectancy. DOE agreed[33] to address this concern and needs to include detailed clarifications stated here in the agreed-on information on hydride embrittlement.

Of possibly greater importance is the lack of experimental work examining the possible detrimental effects of fluoride on the corrosion behavior of titanium. Though fluoride was present in some test environments at low levels, the presence of other species, such as calcium and silicon, may have limited the concentration of free fluoride available for complexation with titanium (Schutz and Grauman, 1985) and masked the evaluation of any accelerating effect of fluoride. To address this concern, DOE agreed[34] to provide sufficient information on the fluoride concentration of the groundwater in contact with drip shields and its potential effect on corrosion.

From the perspective of localized corrosion, though little or no localized corrosion has been observed thus far, the localized corrosion behavior of titanium-palladium alloys has not been extensively studied. It has been observed that, under relatively aggressive conditions, these materials still exhibit high crevice corrosion resistance (Brossia and Cragnolino, 2001a,b). In the presence of fluoride, however, significant attack has been reported, and, in fact, some crevice corrosion in chloride-fluoride environments has been observed (Brossia and Cragnolino,

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[33]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000)." Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[34]Ibid.

2001a). In addition, the possible detrimental effects of fabrication methods, such as weldments, have not been evaluated and further evaluation should be provided once the design has been finalized. To address this concern, DOE agreed[35] to provide sufficient data and rationales in assessing the susceptibility of drip shields to localized corrosion.

Environmentally assisted cracking of titanium-palladium alloys has not been extensively examined. As noted, it is generally accepted that environmentally assisted cracking of titanium alloys occurs through a hydrogen embrittlement-type mechanism probably related to hydride precipitation and cracking. DOE, however, considers stress corrosion cracking and hydride-induced cracking to be separate mechanisms. In fact, DOE even is considering two possible models for stress corrosion cracking (threshold stress intensity and slip-film dissolution). It is unclear how these stress corrosion cracking models fit into the generally accepted mechanistic understanding of hydrogen-embrittlement-based environmentally assisted cracking of titanium alloys. DOE should clarify if it plans to use these models to predict environmentally assisted cracking of the Titanium Grade 7 drip shield. With regard to hydride-induced cracking of the drip shield, DOE’s recent change to use the minimum hydrogen concentration necessary for hydride-induced cracking based on limited experimental work using Titanium Grade16 (CRWMS M&O, 2000r) may be more realistic but less conservative than the previous efforts using the values for commercial purity titanium. Given the relative lack of data in this area on titanium-palladium alloys and the uncertainty surrounding the calculations, a more conservative approach may be more adequate. To address this concern, DOE agreed[36] to provide sufficient data and rationales for the possibility of drip shield stress corrosion cracking.

Additional technical bases for the fraction of hydrogen absorbed by titanium during corrosion processes have been provided (CRWMS M&O, 2000s). The effects that palladium may have on this value should be evaluated further, especially given the catalytic effects of palladium on hydrogen generation and the reported increases in absorbed hydrogen at constant corrosion rates for palladium-bearing alloys compared with nonpalladium-titanium alloys (Fukuzuka, et al., 1980). The technical basis for the fraction of hydrogen absorbed, especially considering the well-known catalytic properties of palladium for hydrogen generation, however, needs to be strengthened. In addition, reliance on the passive corrosion rates measured from weight loss coupons may lead to a nonconservative estimate of the quantity of hydrogen absorbed. This estimate suggests that hydride-induced cracking of titanium may occur during anticipated repository conditions. It is suggested DOE examine the possibility of enhanced hydrogen uptake and absorption in the palladium-bearing titanium alloys, especially Grade 7 rather than Grade 16, because the differences in the palladium content of these materials could make a difference in the measured hydrogen uptake rates. The possibility of enhanced hydrogen update in the presence of fluoride through destabilization of the TiO_2 oxide should be evaluated also. It is recommended that DOE confirms the low corrosion rates measured from weight-loss experiments and from polarization data with long-term electrochemical tests or other techniques

[35]Schlueter, J.R. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000).” Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[36]Ibid.

designed to sensitively measure the passive corrosion rate. To address these concerns, DOE agreed[37] to provide sufficient data and rationales for the efficiency of hydrogen uptake along with the sensitive measurement of associated corrosion rates.

The belief that stress corrosion cracking and hydride-induced cracking of the drip shield have low consequences because of presumed crack plugging by corrosion or calciferous deposits should be reevaluated further. Though it may be possible that any cracks forming on the drip shield eventually will be plugged such that no water transport through the crack is possible, the consequence of the crack presence on subsequent rockfall events should be examined. In such cases, it might be envisioned that an existing crack acts as the nucleation point for a substantial opening in the drip shield. To address this concern, DOE agreed[38] to provide sufficient information on the potential effects of crack plugging by corrosion or by calciferous deposits on the further development of stress corrosion cracking.

Sufficient data and rationales are required for the verification of the model abstraction in the drip shield performance. Thermal embrittlement may occur by the formation of secondary phases. The accurate assessment of fluoride ion concentration on the drip shield surface may exclude fluoride-induced fast drip shield corrosion or hydride embrittlement. The likelihood of drip shield susceptibility to localized corrosion needs to be better assessed, especially with respect to drip shield fabrication. The DOE assessment of the environmentally assisted cracking of drip shields is unclear regarding critical hydrogen concentration and the hydrogen uptake process, and the proposed mechanism for crack plugging by corrosion or calciferous deposits as means for crack arrest.

In summary, as noted previously, DOE agreed to provide the needed information before any future license application is submitted.

3.3.1.4.3 Criticality Within the Waste Package

DOE screened the occurrence of nuclear criticality for commercial spent fuel, normal conditions, and seismic events from the Total System Performance Assessment–Site Recommendation based on the lack of waste package breach or failure at any time during the first 10,000 years of postclosure (CRWMS M&O, 2000u). For igneous events, DOE screened the occurrence of criticality based on a low probability of formation of a critical configuration. The basis for this screening has been documented in CRWMS M&O (2000v). NRC concerns regarding the DOE screening argument for nuclear criticality are discussed in Section 3.2.2. Identification of Events with Probabilities Greater Than 10^{-8} Per Year. Per an agreement made

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[37]Schlueter, J.R. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000).” Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[38]Ibid.

during the DOE and NRC Technical Exchange on Criticality,[39] DOE committed to performing a what-if analysis, using the topical report approach, which would simulate the consequence of a criticality event. Discussion of criticality in the following sections relates to the topical report DOE developed to describe the methodology that will be used to assess the probability and consequences of an in-package criticality event within the repository system (DOE, 1998b). NRC reviewed this topical report and documented the results in a safety evaluation report (NRC, 2000d). The safety evaluation report contains 28 open items on the methodology, which, when closed, will document NRC acceptance of the proposed methodology to address criticality in the repository system. Per an agreement made during the DOE and NRC Technical Exchange on Criticality, DOE provided NRC with Revision 1 of this topical report, intended to address 27 of these open items (DOE, 2000). In an NRC letter dated December 10, 2001, NRC stated it accepted Revision 01 of the topical report for detailed technical review. It is expected that the NRC review will be completed by the end of 2002. If this new revision of the topical report is found acceptable, it will provide confidence that DOE will be able to address the effects of criticality on the performance of the repository system in any potential license application even if DOE is unable to support arguments for screening criticality from the total system performance assessment. The remaining open item on burnup measurements was discussed at the DOE and NRC Technical Exchange on Preclosure Safety.[40]

3.3.1.4.3.1 System Description and Model Integration Are Adequate

Overall, the current information, along with the agreements reached between the DOE and NRC (Section 3.3.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess degradation of engineered barriers (criticality within the waste package) with respect to system description and model integration.

The open items associated with the DOE topical report on criticality include many issues related to the in-package criticality model: (i) development of a modeling approach for igneous-activity-induced criticality, (ii) losses of radionuclides from intact assemblies through pinholes and cracks in the cladding and establishment of the uncertainty associated with this loss, (iii) inclusion of a criticality margin, (iv) cross-dependency of configuration parameters for k_{eff} regression equations, (v) provision of a multi-parameter approach in bias-trending analyses, (vi) defense of method used for extending trends, (vii) development of a methodology to determine steady-state criticality consequences for nonaqueous moderators, (viii) addition of consequences other than radionuclide inventory increase to the steady-state criticality consequence model, (ix) description of the interface between the criticality topical report analyses and the total system performance assessment criticality risk analysis, and (x) physical verification of burnup levels of spent nuclear fuel (NRC, 2000d).

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[39]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Key Technical Issue: Criticality (October 23–24, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[40]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Key Technical Issue: Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

As noted above, if the new revision of the topical report is found acceptable, it will provide confidence that DOE will be able to address the effects of criticality on the performance of the repository system in any potential license application even if DOE is unable to support arguments for screening criticality from the total system performance assessment.

3.3.1.4.3.2 Data Are Sufficient for Model Justification

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess degradation of engineered barriers (criticality within the waste package) with respect to data being sufficient for model justification.

The open items associated with the DOE topical report on criticality include several issues related to the sufficiency of data supporting the in-package criticality model, including the DOE needs to use cross-sectional data at the temperature of the waste package or critical benchmarks and DOE must provide justification for the correction factors developed for boron remaining in solution (NRC, 2000d). In the DOE and NRC Technical Exchange on Criticality, DOE indicated that examples of data that would be used in the criticality analyses for the quantity and alternative forms of corrosion products in the waste package and radionuclide release from small cracks in cladding could be found in several reports (CRWMS M&O, 1998b, 1999b,c, 2000w,x,y; Wilson, 1990). Additionally, DOE indicated that additional data would be located in the validation reports for the inventory, neutronics, and geochemistry computer codes that will be used in the criticality modeling. DOE agreed [41] to provide these validation reports to NRC before submitting the license application for the proposed Yucca Mountain repository.

As noted previously, if the new revision of the topical report is found acceptable, it will provide confidence that DOE will be able to address the effects of criticality on the performance of the repository system in any potential license application even if DOE is unable to support arguments for screening criticality from the total system performance assessment.

3.3.1.4.3.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess degradation of engineered barriers (criticality within the waste package) with respect to data uncertainty being characterized and propagated through the model abstraction.

The open items associated with the DOE topical report on criticality include several issues related to the assessment of data uncertainty in the in-package criticality model: (i) DOE needs

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[41]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Key Technical Issue: Criticality (October 23–24, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

to account for bias and uncertainty in the isotopic depletion model, (ii) DOE must account for all types of uncertainty and bias in the criticality analysis, (iii) DOE must include the isotopic bias and uncertainty in developing the critical limit, and (iv) DOE must include uncertainty introduced by the use of a regression equation and look-up tables (NRC, 2000d). In the DOE and NRC Technical Exchange on Criticality, DOE indicated that examples of the consideration of data uncertainty that would be used in the criticality analyses for the quantity and alternative forms of corrosion products in the waste package and radionuclide release from small cracks in cladding could be found in several reports (CRWMS M&O, 1998b, 1999b,c, 2000w,x,y; Wilson, 1990). Additionally, DOE indicated that quantification of data uncertainty would be located in the validation reports for the inventory, neutronics, and geochemistry computer codes that will be used in the criticality modeling. DOE agreed [42] to provide these validation reports to NRC before submitting the license application for the proposed Yucca Mountain repository.

As noted previously, if the new revision of the topical report is found acceptable, it will provide confidence that DOE will be able to address the effects of criticality on the performance of the repository system in any potential license application even if DOE is unable to support arguments for screening criticality from the total system performance assessment.

3.3.1.4.3.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess degradation of engineered barriers (criticality within the waste package) with respect to model uncertainty being characterized and propagated through the model abstraction.

The open items associated with the DOE topical report on criticality include one issue related to the assessment of model uncertainty in the in-package criticality model, demonstrating the adequacy of using a one-dimensional point-depletion calculation in the depletion analyses instead of two- or three-dimensional models (NRC, 2000d). In the DOE and NRC Technical Exchange on Criticality, DOE indicated that the validation reports will support the inventory computer code in the criticality modeling. DOE agreed [43] to provide these validation reports to NRC before submitting the license application for the proposed Yucca Mountain repository.

As noted previously, if the new revision of the topical report is found acceptable, it will provide confidence that DOE will be able to address the effects of criticality on the performance of the repository system in any potential license application even if DOE is unable to support arguments for screening criticality from the total system performance assessment.

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[42]Schlueter, J.R. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Key Technical Issue: Criticality (October 23–24, 2000).” Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[43]Ibid.

3.3.1.4.3.5 Model Abstraction Output Is Supported by Objective Comparisons

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess degradation of engineered barriers (criticality within the waste package) with respect to model abstraction output being supported by objective comparisons.

Open items associated with the DOE topical report on criticality include issues related to the support of models used in the in-package criticality model, including DOE must validate the regression equation or look-up table for all ranges of configurations and waste form parameters affecting k_{eff} and that DOE needs to develop a validation approach for the power model for steady-state criticality consequences (NRC, 2000d). In the DOE and NRC Technical Exchange on Criticality, DOE indicated that the justification of the models used in the criticality analyses would be located in the validation reports for the inventory, neutronics, and geochemistry computer codes. DOE agreed[44] to provide these validation reports to NRC before submitting the license application for the proposed Yucca Mountain repository.

As noted previously, if the new revision of the topical report is found acceptable, it will provide confidence that DOE will be able to address the effects of criticality on the performance of the repository system in any potential license application even if DOE is unable to support arguments for screening criticality from the total system performance assessment.

3.3.1.5 Status and Path Forward

Table 3.3.1-1 provides the status of all key technical issue subissues, referenced in Section 3.3.1.2, for the Degradation of Engineered Barriers Integrated Subissue. The table also provides the related DOE and NRC agreements pertaining to the Degradation of Engineered Barriers Integrated Subissue. The agreements listed in the table are associated with one or all five generic acceptance criteria discussed in Section 3.3.1.4. Note that the status and the detailed agreements (or path forward) pertaining to all the key technical issue subissues are provided in Table 1.1-3 and Appendix A.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (through specified testing, analyses, and the like), acceptably addresses the NRC questions so that no information beyond that provided, or agreed to, will likely be required at the time of a potential license application.

[44]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Key Technical Issue: Criticality (October 23–24, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

Table 3.3.1-3. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreements*
Container Life and Source Term	Subissue 1—The Effects of Corrosion Processes on the Lifetime of Containers	Closed-Pending	CLST.1.01 through CLST.1.17
	Subissue 2—The Effects of Phase Instability of Materials and Initial Defects on Mechanical Failure and Lifetime of Containers	Closed-Pending	CLST.2.04 through CLST.2.08
	Subissue 5—The Effect of In-Package Criticality on Waste Package and Engineered Barrier Subsystem Performance	Closed-Pending	CLST.5.01 CLST.5.03 through CLST.5.07
	Subissue 6—The Effects of Alternate Engineered Barrier Subsystem Design Features on Container Lifetime and Radionuclide Release from the Engineered Barrier Subsystem	Closed-Pending	CLST.6.01 through CLST.6.04
Thermal Effects on Flow	Subissue 2—Thermal Effects on Temperature, Humidity, Saturation, and Flux	Closed-Pending	TEF.2.03 TEF.2.04 TEF.2.09
Evolution of the Near-Field Environment	Subissue 2—Effects of Coupled Thermal-Hydrologic-Chemical Processes on Waste Package Chemical Environment	Closed-Pending	ENFE.2.04 ENFE.2.14
	Subissue 3—Effects of Coupled Thermal-Hydrologic-Chemical Processes on Chemical Environment for Radionuclide Release	Closed-Pending	ENFE.3.01
	Subissue 5—Effects of Coupled Thermal-Hydrologic-Chemical Processes on Potential Nuclear Criticality in the Near-Field	Closed-Pending	ENFE.5.03
Repository Design and Thermal-Mechanical Effects	Subissue 3—Thermal-mechanical Effects on Underground Facility Design and Performance	Closed-Pending	RDTME.3.18
Total System Performance Assessment and Integration	Subissue 1—System Description and Demonstration of Multiple Barriers	Closed-Pending	None
	Subissue 2—Scenario Analysis and Event Probability	Closed-Pending	TSPAI.2.01 TSPAI.2.02 TSPAI.2.04

Table 3.3.1-3. Related Key Technical Issue Subissues and Agreements (continued)			
Key Technical Issue	**Subissue**	**Status**	**Related Agreements***
Total System Performance Assessment and Integration	Subissue 3—Model Abstraction	Closed-Pending	TSPAI.3.01 through TSPAI.3.05
	Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards	Closed-Pending	None

*Related DOE and NRC agreements are associated with one or all five generic acceptance criteria.

Note: Key Technical Issue Agreement GEN.1.01 pertains to multiple integrated subissues, as well as some specific issues related to this integrated subissue.

3.3.1.6 References

American Society for Testing and Materials. “Standard Practice for Preparing, Cleaning, and Evaluating Corrosion Test Specimens: G1–90.” Annual Book of Standards. Vol. 03.02. West Conshohocken, Pennsylvania: American Society for Testing and Materials. 1997.

———. “Standard Practice for Calculation of Corrosion Rates and Related Information from Electrochemical Measurements: G102–89.” Annual Book of Standards. Vol. 03.02. West Conshohocken, Pennsylvania: American Society for Testing and Materials. pp. 416–422. 1999.

Barkatt, A. and J.A. Gorman. “Tests to Explore Specific Aspects of the Corrosion Resistance of C-22.” Nuclear Waste Technical Review Board Meeting, Carson City, Nevada, 1 August 2000. Carson City, Nevada: Nuclear Waste Technical Review Board. 2000.

Bickford, D.F. and R.A. Corbett. “Material Selection for the Defense Waste Processing Facility.” Proceedings of the International Conference on Corrosion of Nickel-Base Alloys, Cincinnati, Ohio, 23–25 October 1984. Metals Park, Ohio: American Society for Metals. pp. 59–67. 1985.

Brossia, C.S. and G.A. Cragnolino. “Effects of Environment and Metallurgical Condition on the Passive and Localized Dissolution of Ti-0.15%Pd.” *Corrosion Journal*. Vol. 57. pp. 768–776. 2001a.

Brossia, C.S. and G.A. Cragnolino. “Effect of Palladium on the Localized and Passive Dissolution of Titanium.” Proceedings of the Corrosion 2001 Conference. Paper No. 01127. Houston, Texas: NACE International. 2001b.

Bruemmer, S.M., V.Y. Gertsman, and L.E. Thomas. "High-Resolution Comparison of Primary- and Secondary-Side Intergranular Degradation in Alloy 600 Steam Generator Tubing." Proceedings of the Corrosion 2000 Conference. Paper No. 196. Houston, Texas: NACE International. 2000.

Carter, J.P. and S.D. Cramer. "Corrosion Resistance of Some Commercially Available Metals and Alloys to Geothermal Brines." *Corrosion Problems in Energy Conversion and Generation*. C.S. Tedmon, ed. Princeton, New Jersey: The Electrochemical Society. pp. 240–250. 1974.

Cieslak, M.J., T.J. Headley, and A.D. Romig, Jr. "The Welding Metallurgy of Hastelloy Alloys C-4, C-22, and C-276." *Metallurgical Transactions*. Vol. 17A. pp. 2,035–2,047. 1986.

Cragnolino, G.A. and N. Sridhar. "A Review of Stress Corrosion Cracking of High-Level Nuclear Waste Container Materials–I." CNWRA 92-021. San Antonio, Texas: CNWRA. 1992.

Cragnolino, G.A., D.S. Dunn, C.S. Brossia, V. Jain, and K. Chan. "Assessment of Performance Issues Related to Alternate EBS Materials and Design Options." CNWRA 99-003. San Antonio, Texas: CNWRA. 1999.

CRWMS M&O. "Total System Performance Assessment–1995." B00000000–01717–2200–00136. Revision 01. Las Vegas, Nevada: CRWMS M&O. 1995.

———. "Chapter 5: Waste Package Degradation Modeling and Abstraction. Total System Performance Assessment–Viability Assessment (TSPA-VA) Analyses Technical Basis Document." B00000000–01717–4301–00005. Revision 01. Las Vegas, Nevada: TRW Environmental Safety Systems, Inc. 1998a.

———. "Probability of a PWR Uncanistered Fuel Waste Package Postclosure Criticality." BBA000000–01717–0210–00010–00. Las Vegas, Nevada: CRWMS M&O. 1998b.

———. "Total System Performance Assessment–Site Recommendation Methods and Assumptions." TDR–MGR–MD–000001. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 1999a.

———. "Updated Calculation of Probability of Criticality for EDA II Waste Package Design." CAL–UDC–MD–0000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 1999b.

———. "Probability of Criticality for MOX SNF." CAL–EBS–NU–0000007. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 1999c.

———. "Total System Performance Assessment for the Site Recommendation." TDR–WIS–PA–000001. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000a.

———. "Waste Package Degradation Process Model Report." TDR–WIS–MD–000002. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000b.

———. “Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations.” TDR–WIS–RL–000001. Revision 04 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000c.

———. “Update to the EIS Engineering File for the Waste Package in Support of the Final EIS.” TDR–EBS–MD–000010. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000d.

———. “Design Analysis for UCF Waste Packages.” ANL–UDC–MD–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000e.

———. “FEPs Screening of Processes and Issues in Drip Shield and Waste Package Degradation.” ANL–EBS–PA–000002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000f.

———. “General Corrosion and Localized Corrosion of Waste Package Outer Barrier.” ANL–EBS–MD–000003. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000g.

———. “Environment on the Surfaces of the Drip Shield and Waste Package Outer Barrier.” ANL–EBS–MD–000001. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000h.

———. “Engineered Barrier System Degradation, Flow, and Transport Process Model Report.” TDR–EBS–MD–000006. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000i.

———. “In-Drift Precipitates/Salts Analysis.” ANL–EBS–MD–000045. Revision 00 ICN 02. Las Vegas, Nevada: CRWMS M&O. 2000j.

———. “Engineered Barrier System: Physical and Chemical Environment Model.” ANL–EBS–MD–000003. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000k.

———. “Stress Corrosion Cracking of the Drip Shield, the Waste Package Outer Barrier, and the Stainless Steel Structural Material.” ANL–EBS–MD–000005. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000l.

———. “WAPDEG Analyses of Waste Package and Drip Shield Degradation.” ANL–EBS–PA–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000m.

———. “Aging and Phase Stability of Waste Package Outer Barrier.” ANL–EBS–MD–000002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000n.

———. “Analysis of Mechanisms for Early Waste Package Failure.” ANL–EBS–MD–000023. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000o.

———. “Design Analysis for the Ex-Container Components.” ANL–XCS–ME–00001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000p.

———. "Features, Events, and Processes: Disruptive Events." ANL–WIS–MD–000005. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000q.

———. "Hydrogen Induced Cracking of Drip Shield." ANL–EBS–MD–000006. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000r.

———. "Review of the Expected Behavior of Alpha Titanium Alloys Under Yucca Mountain Conditions." TDR–EBS–MD–000015. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000s.

———. "General Corrosion and Localized Corrosion of Drip Shield." ANL–EBS–MD–000004. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000t.

———. "Features, Events, and Processes—System Level and Criticality." ANL–WIS–MD–000019. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000u.

———. "Probability of Criticality Before 10,000 Years." CAL–EBS–NU–000014. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000v.

———. "Waste Package Related Impacts of Plutonium Disposition Waste Forms in a Geologic Repository." TDR–EBS–MD–000003. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000w.

———. "EQ6 Calculation for Chemical Degradation of Shippingport LWBR (Th/U Oxide) Spent Nuclear Fuel Waste Packages." CAL–EDC–MD000008. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000x.

———. "EQ6 Calculation for Chemical Degradation of Pu-Ceramic Waste Packages: Effects of Updated Materials Composition and Rates." CAL–EDC–MD–000003. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000y.

DOE. "Viability Assessment of a Repository at Yucca Mountain. Vol. 3: Total System Performance Assessment." DOE/RW–0508/V3. Las Vegas, Nevada: DOE, CRWMS M&O. 1998a.

———. "Disposal Criticality Analysis Methodology Topical Report." YMP/TR–004Q. Revision 00. Las Vegas, Nevada: DOE. 1998b.

———. "Disposal Criticality Analysis Methodology Topical Report." YMP/TR–004Q. Revision 01. Las Vegas, Nevada: DOE. 2000.

Fehlner, F.P. *Low-Temperature Oxidation: The Role of Vitreous Oxides*. New York City, New York: John Wiley and Sons. 1986.

Ford, F.P. "The Crack-Tip System and Its Relevance to the Prediction of Cracking in Aqueous Environments." *Environment-Induced Cracking of Metals.* R.P. Gangloff and M.B. Ives, eds. Houston, Texas: NACE International. pp. 139–165. 1990.

Ford, F.P. and P.L. Andresen. “Development and Use of a Predictive Model of Crack Propagation in 304/316l, A533B/A508 and Inconel 600/182 Alloys in 288 °C Water.” Proceedings of the Third International Symposium on Environmental Degradation of Materials in Nuclear Power Systems—Water Reactors. G.J. Theus and J.R. Weeks, eds. Warrendale, Pennsylvania: The Metallurgical Society. pp. 789–800. 1988.

Fukuzuka, T., K. Shimogori, and H. Satoh. “Role of Palladium in Hydrogen Absorption of Ti-Pd Alloy.” Proceedings of Titanium ‘80: Science and Technology. Warrendale, Pennsylvania: The Metallurgical Society of the American Institute of Mining, Metallurgical and Petroleum Engineers. pp. 2,695–2,703. 1980.

Harrar, J.E., R.D. McCright, and A. Goldberg. “Field Electrochemical Measurements of Corrosion Characteristics of Materials in Hypersaline Geothermal Brine.” UCRL–52376. Livermore, California: Lawrence Livermore National Laboratory. 1977.

———. “Corrosion Characteristics of Materials in Hypersaline Geothermal Brine.” *SAMPE Quarterly*. Vol. 10, No. 1. pp. 1–15. 1978.

Heubner, U.L., E. Altpeter, M.B. Rockel, and E. Wallis. “Electrochemical Behavior and its Relation to Composition and Sensitization of Ni-Cr-Mo Alloys in ASTM G28 Solution.” *Corrosion*. Vol. 45, No 3. pp. 249–259. 1989.

Kessler, J. and R. McGuire. “Yucca Mountain Total System Performance Assessment, Phase 3.” EPRI TR–107191, 3055-02. Palo Alto, California: Electric Power Research Institute. 1996.

Kirchheim, R., B. Heine, H. Fischmeister, S. Hofmann, H. Knote, and U. Stoltz. “The Passivity of Iron-Chromium Alloys.” *Corrosion Science*. Vol. 30, No. 7. pp. 899–917. 1989.

Lorang, G., N. Jallerat, K. Vu Quang, and J.-P. Langeron. “AES Depth Profiling of Passive Overlayers Formed on Nickel Alloys.” *Surface and Interface Analysis*. Vol. 16. pp. 325–330. 1990.

McCright, R.D. “Engineered Materials Characterization, Corrosion Data and Modeling—Update for the Viability Assessment.” UCRL–ID–119564. Vol. 3. Revision 1.1. Livermore, California: Lawrence Livermore National Laboratory. 1998.

Mohanty, S. and T.J. McCartin. “Total-System Performance Assessment (TPA) Version 3.2 Code, Module Descriptions and User’s Guide.” San Antonio, Texas: CNWRA. 1998.

Mohanty, S., R. Codell, R.W. Rice, J. Weldy, Y. Lu, R.M. Byrne, T.J. McCartin, M.S. Jarzemba, and G.W. Wittmeyer. “System Level Repository Sensitivity Analyses Using TPA Version 3.2 Code.” CNWRA 99-002. San Antonio, Texas: CNWRA. 1999.

Nesterova, Ye.V., V.V. Rybin, S.S. Ushkov, and G.I. Kolodkina. “Investigation of the Nature of the Thermal Embrittlement of Low A Alloys of Titanium.” *The Physics of Metals and Metallography*. Vol. 50. pp. 83–89. 1980.

NRC. NUREG–1464, "NRC Iterative Performance Assessment Phase 2: Development of Capabilities for Review of a Performance Assessment for a High-Level Waste Repository." R.G. Wescott, M.P. Lee, and N.A. Eisenberg, eds. Washington, DC: NRC. October 1995.

———. NUREG–1668, "NRC Sensitivity and Uncertainty Analysis for a Proposed HLW Repository at Yucca Mountain, Nevada, Using TPA 3.1." Washington, DC: NRC. March 1999.

———. "Issue Resolution Status Report, Key Technical Issue: Total System Performance Assessment and Integration." Revision 3. Washington, DC: NRC. 2000a.

———. "Safety Evaluation Report for Disposal Criticality Analysis Methodology Topical Report." Revision 0. Washington, DC: NRC. 2000b.

———. "Issue Resolution Status Report, Key Technical Issue: Thermal Effects on Flow." Revision 3. Washington, DC: NRC. 2000c.

———. "Issue Resolution Status Report, Key Technical Issue: Evolution of the Near-Field Environment." Revision 3. Washington, DC: NRC. 2000d.

———. "Issue Resolution Status Report, Key Technical Issue: Repository Design and Thermal-Mechanical Effects." Revision 3. Washington, DC: NRC. 2000e.

———. "Issue Resolution Status Report, Key Technical Issue: Container Life and Source Term." Revision 3. Washington, DC: NRC. 2001.

———. NUREG–1804, "Yucca Mountain Review Plan—Draft Report for Comment." Revision 2. Washington, DC: NRC. March 2002.

Oldfield, J.W. "Corrosion Initiation and Propagation of Nickel Base Alloys in Severe Sea Water Applications." Proceedings of the Corrosion '95 Conference. Paper No. 266. Houston, Texas: NACE International. 1995.

Roy, A.K., D.L. Fleming, and B.Y. Lum. "Stress Corrosion Cracking of Fe-Ni-Cr-Mo, Ni-Cr-Mo, and Ti Alloys in 90 °C Acidic Brine." Proceedings of the Corrosion '98 Conference. Paper No. 157. Houston, Texas: NACE International. 1998.

Schutz, R.W. and J.S. Grauman. "Laboratory Corrosion Behavior of Titanium and Other High Performance Alloys in Representative FGD Scrubber Environments." Proceedings of the Corrosion '85 Conference. Paper No. 52. Houston, Texas: NACE International. 1985.

Shoesmith, D.W. and M. Kolar. "Waste Package Performance, Alternative Approaches to Assessing the Performance and Suitability of Yucca Mountain for Spent Fuel Disposal." EPRI TR–108732. Palo Alto, California: Electric Power Research Institute. 1998.

Smailos, E. "Corrosion of High-level Waste Packaging Materials in Disposal Relevant Brines." *Nuclear Technology*. Vol. 104. pp. 343–350. 1993.

Speidel, M.O. “Stress Corrosion Cracking on Stainless Steels in NaCl Solutions.” *Metallurgical Transactions*. Vol. 12A. pp. 779–789. 1981.

Sridhar, N. “Effect of Alloying Elements on Localized Corrosion Resistance of Nickel-Base Alloys.” *Advances in Localized Corrosion*. H. Isaacs, U. Bertocci, J. Kruger, and S. Smialowska, eds. Houston, Texas: NACE International. pp. 263–269. 1990.

Sridhar, N., J.C. Walton, G.A. Cragnolino, and P.K. Nair. “Engineered Barrier System Performance Assessment Codes (EBSPAC) Progress Report.” CNWRA 93-021. San Antonio, Texas: CNWRA. 1993.

Wilson, C.N. “Results from NNWSI Series 3 Spent Fuel Dissolution Tests.” PNL–7170. Richland, Washington: Pacific Northwest Laboratory. 1990.

Wilson, M.L., J.H. Gauthier, R.W. Barnard, G.E. Barr, H.A. Dockery, E. Dunn, R.R. Eaton, D.C. Guerin, N. Lu, M.J. Martinez, R. Nilson, C.A. Rautman, T.H. Robey, B. Ross, E.E. Ryder, A.R. Schenker, S.A. Shannon, L.H. Skinner, W.G. Halsey, J.D. Gansemer, L.C. Lewis, A.D. Lamont, I.R. Triay, A. Meijer, and D.E. Morris. “Total-System Performance Assessment for Yucca Mountain—SNL Second Iteration (TSPA–1993).” Volume 2. SAND93–675. Albuquerque, New Mexico: Sandia National Laboratories. 1994.

3.3.2 Mechanical Disruption of Engineered Barriers

3.3.2.1 Description of Issue

The Mechanical Disruption of Engineered Barriers Integrated Subissue addresses the DOE total system performance assessment of engineered barriers subjected to mechanically disruptive events. Engineered barriers include the emplacement drift, waste package, waste package pallet, and drip shield and drift invert system. Although engineered backfill is not presently included in the engineered barrier subsystem design, it may be placed within the emplacement drifts of the proposed geologic repository for commercial spent nuclear fuel and high-level waste. If used, engineered backfill would also be assessed to determine how its performance characteristics and interactions with other engineered barrier subsystem components would be affected by mechanically disruptive events. The potential disruptive events to be addressed by the Mechanical Disruption of Engineered Barriers Integrated Subissue review are igneous intrusion, faulting, seismicity, rockfall and drift collapse, and criticality. The relationship between this integrated subissue to other integrated subissues is depicted in Figure 3.3.2-1. The overall organization and identification of all the integrated subissues is depicted in Figure 1.1.2.

The DOE description and technical bases for the analyses of mechanical disruption of engineered barriers model abstraction are documented in various process model reports, analysis and model reports, system description documents, and calculation reports. These documents, which are identified in the appropriate subsections that follow, are reviewed to the extent that they contain (i) process-level models, data, and analyses that support the abstracted models used by DOE in the total system performance assessment of the engineered barrier subsystem when subjected to mechanically disruptive events and (ii) screening arguments used to justify the exclusion of mechanical disruption of engineered barriers processes from consideration.

With the exception of igneous activity, DOE screened out all potential disruptive events from consideration of the repository total system performance assessment based on low-probability and low-consequence arguments. Igneous effects accounted for in the mechanical disruption of engineered barriers model abstraction are presently limited by DOE to interactions between basaltic magma and waste packages not located along a magma flow path to the surface. Waste package response to magma flowing to the surface (i.e., in the subvolcanic conduit) is evaluated as part of the Volcanic Disruption of Waste Packages Integrated Subissue. Key processes associated with the mechanical disruption of engineered barriers by igneous intrusion are (i) basaltic magma flows into proposed repository drifts, (ii) engineered barrier component response to basaltic magma exposure, and (iii) cooling of the basalt and engineered barrier subsystem, allowing reestablishment of long-term hydrologic transport processes.

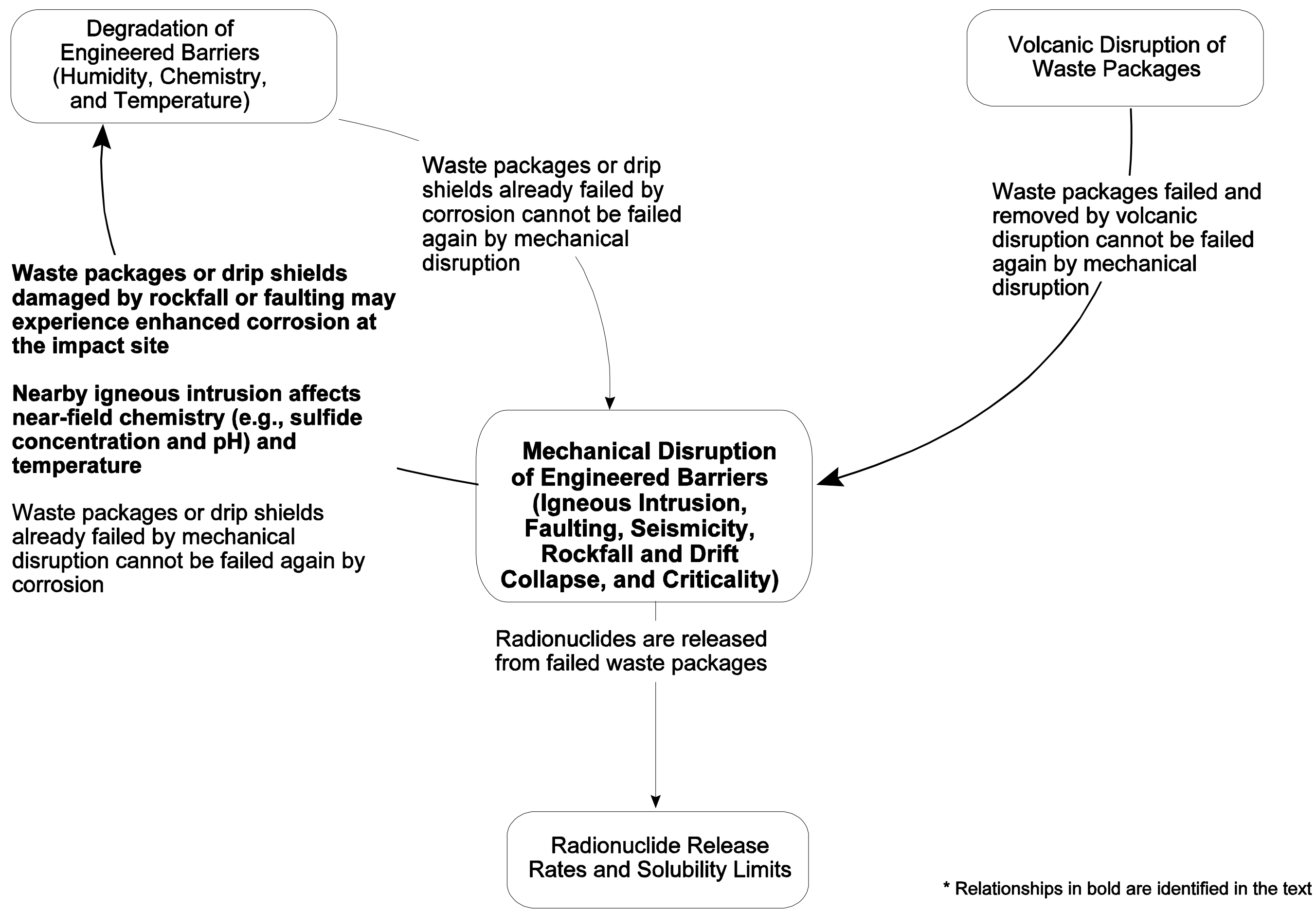

Figure 3.3.2-1. Diagram Illustrating the Relationship Between Mechanical Disruption of Engineered Barriers and Other Integrated Subissues

3.3.2.2 Relationship to Key Technical Issue Subissues

The Mechanical Disruption of Engineered Barriers Integrated Subissue incorporates subject matter previously captured in the following key technical issue subissues.

- Container Life and Source Term: Subissue 1—Effects of Corrosion Processes on the Lifetime of the Containers (NRC, 2001)
- Container Life and Source Term: Subissue 2—Effects of Phase Instability of Materials and Initial Defects on the Mechanical Failure and Lifetime of the Containers (NRC, 2001)
- Container Life and Source Term: Subissue 5—Effect of In-Package Criticality on Waste Package and Engineered Barrier Subsystem Performance (NRC, 2001)
- Container Life and Source Term: Subissue 6—Effect of Alternate Engineered Barrier Subsystem Design Features on Container Lifetime and Radionuclide Release from the Engineered Barrier Subsystem (NRC, 2001)
- Igneous Activity: Subissue 1—Probability of Future Activity (NRC, 1999a)
- Igneous Activity: Subissue 2—Consequences of Igneous Activity (NRC, 1999a)
- Repository Design and Thermal Mechanical Effects: Subissue 1—Implementation of an Effective Design Control Process within the Overall Quality Assurance Program (NRC, 2000a)
- Repository Design and Thermal Mechanical Effects: Subissue 2—Design of the Geologic Repository Operations Area for the Effects of Seismic Events and Direct Fault Disruption (NRC, 2000a)
- Repository Design and Thermal Mechanical Effects: Subissue 3—Thermal-Mechanical Effects on Underground Facility Design and Performance (NRC, 2000a)
- Structural Deformation and Seismicity: Subissue 1—Faulting (NRC, 1999b)
- Structural Deformation and Seismicity: Subissue 2—Seismicity (NRC, 1999b)
- Structural Deformation and Seismicity: Subissue 3—Fracturing and Structural Framework of the Geologic Setting (NRC, 1999b)
- Structural Deformation and Seismicity: Subissue 4—Tectonic Framework of the Geologic Setting (NRC, 1999b)
- Total System Performance Assessment and Integration: Subissue 1—System Description and Documentation of Multiple Barriers (NRC, 2000b)

- Total System Performance Assessment and Integration: Subissue 2—Scenario Analysis and Event Probability (NRC, 2000b)
- Total System Performance Assessment and Integration: Subissue 3—Model Abstraction (NRC, 2000b)
- Total System Performance Assessment and Integration: Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards (NRC, 2000b)

The key technical issue subissues formed the bases for the previous version of the issue resolution status reports and also were the bases for technical exchanges with DOE where agreements were reached on what additional information DOE needed to provide to resolve the subissues. The resolution status of this integrated subissue is based on the resolution status of each of the contributing key technical issue subissues. The subsequent sections incorporate applicable portions of these key technical issue subissues, however, no effort was made to explicitly identify each subissue.

3.3.2.3 Importance to Postclosure Performance

One aspect of risk informing the NRC review was to determine how this integrated subissue is related to the DOE repository safety strategy. Specifically, the DOE Repository Safety Strategy (CRWMS M&O, 2000a) acknowledges that mechanical disruption of engineered barriers will affect the long-term risks of the proposed repository to the public health and safety. Both the performance of the waste package and that of the drip shield and drift invert system are listed among the eight principal factors for the postclosure safety case (CRWMS M&O, 2000a).

The Yucca Mountain area, which lies within the Basin and Range tectonic province of the western Cordillera, has been seismically, tectonically, and volcanically active on the timescale of a geologic repository. Future seismotectonic and volcanic activities could affect the stability of both the natural and engineered barrier subsystems of the repository.

The Total System Performance Assessment–Site Recommendation (CRWMS M&O, 2000b) reports no radiological risk in 10,000 years from the basecase repository. Based on the DOE analyses, intrusive igneous activity has a probability weighted risk of approximately 1 μSv/yr [0.1 mrem/yr] and is classified by DOE as a principle factor (CRWMS M&O, 2000b). This risk value increases by approximately one order of magnitude when a probability value of 1×10^{-7} (NRC, 1999a) is used. DOE agreed[1] to include, for its licensing case, the results of a single-point sensitivity analysis for extrusive and intrusive igneous processes at 1×10^{-7}. In a later DOE analysis, the risk from intrusive igneous activity decreased by approximately one order of magnitude to 0.1 μSv/yr [0.01 mrem/yr] in Bechtel SAIC Company, LLC (2001a), based primarily on changes to radionuclide solubility and transport models. With the exception of

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[1]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (August 29–31, 2000)." Letter (October 23) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

volcanism, this level of intrusive risk clearly exceeds calculated risks from other postclosure features, events, and processes in Bechtel SAIC Company, LLC (2001a,b).

Staff raised concerns with the technical bases used by DOE to evaluate both extrusive and intrusive igneous activities in the Total System Performance Assessment–Site Recommendation[2] (Hill and Connor, 2000). Analyses presented in, for example, NRC (1999a) also demonstrate that probability-weighted risk from postclosure volcanism may be on the order of 10 μSv/yr [1 mrem/yr], with significant uncertainties associated with this value. Further, processes of magma-repository-waste package interactions affect the amount of radionuclide potentially released by groundwater pathways. This interaction directly controls the amount and character of high-level waste potentially available for subsequent hydrologic transport (see Section 3.3.2.4.1 for detailed discussion).

Because postclosure performance requirements rely on continued functionality of the waste package and drip shield and drift invert system, DOE committed to designing these engineered barrier subsystem components to withstand the effects of vibratory ground motion caused by earthquakes and the potential loads arising from drift degradation (i.e., rock block impacts and drift collapse). Although the engineered barrier subsystem design has yet to be finalized, DOE screened out nearly all the primary and secondary features, events, and processes pertaining to vibratory ground motion and drift degradation from consideration in the Total System Performance Assessment Code based on the aforementioned design commitment. The only features, events, and processes pertaining to seismic and drift degradation loads accounted for in the DOE Total System Performance Assessment Code is the potential failure of the commercial spent nuclear fuel cladding caused by vibratory ground motion (CRWMS M&O, 2000c). This scenario is included in the total system performance assessment basecase, and DOE concluded that seismically induced cladding failure does not contribute to dose (CRWMS M&O, 2000c) because the waste packages will remain intact for the entire regulatory period regardless of any cladding failures. The staff reviewed the DOE cladding failure analyses and identified several deficiencies (see Section 3.3.4.4.3).

Criticality is also included in the Mechanical Disruption of Engineered Barriers Integrated Subissue discussion for two reasons. The first reason is an in-package criticality event may cause significant mechanical degradation or outright failure of the waste form and waste package. The second reason is a criticality event could be initiated as a result of another, unrelated mechanically disruptive event (e.g., rockfall). For the second case, the extent of the damage caused by the original disruptive event could be significantly magnified if criticality were to occur as a related consequence.

3.3.2.4 Technical Basis

NRC developed a plan (2002) consistent with acceptable criteria and review methods found in previous issue resolution status reports. A review of the DOE approach for including

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[2]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (August 29–31, 2000)." Letter (October 23) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

mechanical disruption of engineered barriers in the total system performance assessment abstraction is provided in the following subsections. For the sake of clarity, the technical basis for the staff comments will be presented within individual subsections for each mechanically disruptive event being reviewed (i.e., igneous intrusion, faulting, seismicity, rockfall and drift collapse, and criticality). Each of these subsections, in turn, have been subdivided and organized according to the five acceptance criteria identified in Section 1.5: (i) System Description and Model Integration Are Adequate, (ii) Data Are Sufficient for Model Justification, (iii) Data Uncertainty Is Characterized and Propagated Through the Model Abstraction, (iv) Model Uncertainty Is Characterized and Propagated Through the Model Abstraction, and (v) Model Abstraction Output Is Supported by Objective Comparisons.

3.3.2.4.1 Igneous Intrusion

3.3.2.4.1.1 System Description and Model Integration Are Adequate

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.2.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess mechanical disruption of engineered barriers (i.e., igneous intrusion) with respect to system description and model integration.

This subsection provides a review of the system description and model integration of the DOE igneous intrusion abstraction for the Total System Performance Assessment–Site Recommendation (CRWMS M&O, 2000b). The DOE description and technical basis for the igneous intrusion abstraction are documented in CRWMS M&O (2000d) and three supporting analysis and model reports (CRWMS M&O, 2000e–g). Calculation report documents (CRWMS M&O, 2000h,i) also provide information relevant to this review.

The DOE approach to evaluating igneous disruption of waste packages involves several conceptual models. Models for magma ascent and initial interaction with proposed repository drifts are discussed in Section 3.3.10 of this report. For the mechanical disruption of engineered barriers, the DOE models begin with the assumption that basaltic magma has flowed into all drifts directly intersected by an ascending dike [e.g., CRWMS M&O (2000e)].

DOE currently assumes only three waste packages on either side of an igneous intrusion (i.e., Zone 1) are damaged to the extent that the waste package provides no impediment for subsequent hydrologic flow and transport (CRWMS M&O, 2000e,f). Staff agree these models consider a sufficient range of interrelated processes to support this conclusion. The remaining waste packages in an intersected drift (i.e., Zone 2), however, have only limited damage resulting from end-cap failure caused by internal pressurization effects (CRWMS M&O, 2000f,h). Although the spent nuclear fuel cladding degraded completely in the Zone 2 waste packages because thermal effects, waste can be mobilized only by water circulation through a limited number of relatively small openings along the waste package lid. Consideration of the full range of physical conditions associated with igneous events would result in much more extensive damage to Zone 2 waste packages than currently modeled by

DOE. To address this concern, DOE agreed[3] to evaluate waste package performance for the duration of the igneous event if the model abstraction takes credit for engineered barriers providing delay in radionuclide release.

Simple calculations show affected waste packages will likely remain exposed to hot {temperatures approximately 1,100 °C [2,012 °F]} basaltic magma for at least 480 hours (NRC, 1999a; CRWMS M&O, 2000e). The yield stress of Alloy 22 decreases from 370 Mpa [54 ksi] at room temperature to 213 MPa [31 ksi] at 760 °C [1,400 °F] (Haynes International, 1988). Similarly, the ultimate tensile strength of the alloy decreases from 786 MPa [114 ksi] at room temperature to 524 MPa [76 ksi] at 760 °C [1,400 °F]. Although the mechanical property data at higher temperatures are not available from the alloy manufacturer literature, the yield stress and ultimate tensile strength at temperatures above 760 °C [1,400 °F] can be estimated (CRWMS M&O, 2000h). At 1,100 °C [2,012 °F], the ultimate tensile strength is estimated to be 226 MPa [33 ksi] (CRWMS M&O, 2000h), and the yield stress is estimated to be 91 MPa [13 ksi]. The ductility of the alloy is not a function of temperature in the range 25–760 °C [77–1,400 °F]. A marked decrease in ductility above 760 °C [1,400 °F] is not expected for this material. After exposure to temperatures in the range 600–900 °C [1,112–1,652 °F], Alloy 22 undergoes microstructural changes that can result in a significant reduction in ductility at subsequently lower temperatures (Summers, et al., 1999; Rebak, et al., 2000). The loss of ductility may increase the susceptibility of the material to mechanical failure as a result of rockfall or seismic events after the intrusive event. The mechanical properties used by DOE when assessing the potential damage that a waste package might incur as a result of interactions with magma (CRWMS M&O, 2000e,h) do not account for these rapidly induced aging effects, which will produce nonlinear trends in mechanical properties. In addition, Alloy 316 nuclear grade stainless steel, which is used to construct the waste package inner barrier, has approximately 30 percent greater thermal expansivity than materials analogous to Alloy 22 (American Society of Mechanical Engineers, 2001), which is used to construct the waste package outer barrier. For the current waste package design, which uses a narrow gap between the inner and outer barriers, these differences in thermal expansion will create tensile stresses in the waste package outer barrier when subjected to magmatic temperatures. Exposure to magmatic temperatures also causes significant gas pressures within the confines of the waste package (CRWMS M&O, 2000e,h). The combined effects of differential thermal expansion and internal gas pressurization should be considered when assessing waste package response to magmatic temperature exposure. In addition, CRWMS M&O (2000e) concludes that drifts intersected by a dike will be blocked at the ends and will fill with magma until fluid pressures are high enough to fracture the drift roof and allow ascent of basaltic magma to continue. Analyses in CRWMS M&O (2000e,h) have not evaluated waste package response to dynamic external pressures in the 3–7 MPa [435–1,015 psi] range as discussed in CRWMS M&O (2000e). To address these

[3]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (August 29–31, 2000)." Letter (October 23) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

concerns, DOE agreed[4] to evaluate waste package response to stresses from thermal and mechanical effects associated with exposure to basaltic magma. This evaluation will include (i) appropriate at-conditions waste package material strength properties and magma flow paths for the likely duration of an igneous event and (ii) aging effects on waste package material strength properties when exposed to basaltic magmatic conditions for the likely duration of an igneous event.

Analyses in CRWMS M&O (2000e,h) also do not consider potentially adverse high-temperature corrosion processes in response to magmatic degassing or contact with basaltic magma. Cooling or depressurized basaltic magma exsolves significant amounts of gas, which is dominantly water with subordinate amounts of carbon, sulfur, and fluorine species (CRWMS M&O, 2000g). Some fraction of the exsolved gas will likely flow into drift-wall fractures not sealed by magma because the air in these fractures are at pressures close to atmospheric pressure (e.g., Rousseau, et al., 1999). The remainder of the exsolved gasses will flow into available openings in nonintersected drifts, including potential voids in backfilled materials. By analogy with basaltic lava flows, degassing may occur for years, potentially decades, after the eruption has ceased. Although the model in CRWMS M&O (2000e) appears to overestimate gas flow, the report concludes that "… the volume of gas arriving at a container is not directly a limiting factor in corrosion." Corrosion of the waste packages and drip shields by magmatic gas, however, is not considered in subsequent models (CRWMS M&O, 2000b,f). This process may be potentially important because magmatic gasses could extend well beyond the boundaries of magma flow in the drifts if drift ends are not completely sealed (i.e., into Zone 3). The potential exists for accelerated degradation of waste packages and drip shields exposed to magmatically derived gases, even if the waste packages and drip shields are not in direct contact with basaltic magma, as in Zones 1 and 2. To address this concern, DOE agreed[5] to evaluate the response of Zone 3 waste packages, or waste packages covered by backfill or rockfall, if exposed to magmatic gasses at conditions appropriate for an igneous event.

Although CRWMS M&O (2000b) concludes no significant natural backfill should occur within 10,000 years, staff recognize the presence of natural or engineered backfill will affect the extent of magma flow into drifts. Limited intrusion into backfilled drifts, however, will still result in the rapid emplacement of some volume of basaltic magma. Some waste packages may be separated from direct contact with this emplaced magma by backfill or rubble. Nevertheless, during the igneous event, basaltic magma will cool against this material and degas. These processes will likely result in coupled thermal and chemical effects on some waste packages in backfill extending beyond Zone 1 of CRWMS M&O (2000e). An appropriate range of temperatures, pressures, and gas geochemical effects has not been evaluated for waste packages in backfilled drifts outlined in CRWMS M&O (2000e,i). The potential exists for accelerated degradation of waste packages and drip shields exposed to high temperatures and magmatically derived gases, even if the waste packages and drip shields are not in direct

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[4]Reamer C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[5]Ibid.

contact with basaltic magma in Zones 1 and 2. To address this concern, DOE agreed[6] to evaluate the response of Zone 3 waste packages, or waste packages covered by backfill or rockfall, if exposed to magmatic gasses at thermal conditions appropriate for an igneous event.

In summary, internal gas pressurization and differential thermal expansion at elevated temperatures, coupled with the large dynamic loads of the overlying magma, aging effects on mechanical strength, and adverse geochemical effects appear sufficient to breach currently proposed waste packages located in DOE Zone 2 during basaltic igneous events. There is insufficient technical bases to conclude that any barrier to subsequent hydrologic transport processes remains for waste packages in Zone 2. Models for basalt degassing also show that corrosion induced by exposure to magmatic gasses may extend beyond direct damage Zones 1 and 2 and could potentially affect all remaining waste packages in Zone 3 (CRWMS M&O, 2000e). If all waste packages in Zone 2 are wholly damaged, there is likely a one order-of-magnitude increase in the source term for subsequent hydrologic transport (CRWMS M&O, 2000e,f). This increase in source term may increase probability-weighted risk significantly above 10 μSv/yr [1 mrem/yr]. The current information and the agreements reached between DOE and NRC (Section 3.3.2.5) are sufficient to ensure the necessary information will be available at the time of a potential license application to address these concerns.

3.3.2.4.1.2 Data Are Sufficient for Model Justification

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.2.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess mechanical disruption of engineered barriers (i.e., igneous intrusion) with respect to sufficient data for model justification.

To support models for waste package resilience during igneous events, data are needed for proposed waste package alloys for the following conditions:

- Material strength properties at magmatic temperatures {i.e., around 1,100 °C [2,012 °F]} for dynamic load conditions appropriate for the potential duration of basaltic igneous events {i.e., recurring pressure variations on order of 0.1–10 MPa [14.5–1,450 psi]}

- Changes in waste package material properties caused by continued exposure to magmatic conditions for the likely duration of basaltic igneous events (i.e., time of exposure at least 500 hours)

- Geochemical effects on waste package properties from cooling and degassing magma in direct contact with waste packages and for waste packages located beyond the zone of direct magma contact

[6]Reamer C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

Limited available data indicate internal gas pressurization and differential thermal expansion at beyond design temperatures, coupled with the dynamic load of the overlying magma, secondary phase precipitation, and potential geochemical effects, appear sufficient to breach currently proposed waste packages located in Zone 2 (CRWMS M&O, 2000d) during basaltic igneous events. In addition, gasses produced from cooling magma appear potentially corrosive to proposed waste package alloys (CRWMS M&O, 2000e). These gasses will likely affect long-term performance of waste packages located in Zone 3 (CRWMS M&O, 2000e). To address these concerns, DOE agreed[7] to evaluate waste package response to stresses from thermal-mechanical effects associated with exposure to basaltic magma. This evaluation will include (i) appropriate at-conditions waste package material strength properties and magma flow paths for the likely duration of an igneous event, (ii) aging effects on waste package material strength properties when exposed to basaltic magmatic conditions for the likely duration of an igneous event, and (iii) evaluation of the response of Zone 3 waste packages, or waste packages covered by backfill or rockfall, if exposed to magmatic gasses at conditions appropriate for an igneous event.

In summary, data used by DOE are insufficient to justify model conclusions for limited waste package damage in Zone 2 of an igneous event or to evaluate the extent of waste package degradation caused by magmatic degassing following an igneous event (e.g., CRWMS M&O, 2000e,h). In addition, currently available data (e.g., Summers, et al., 1999; Rebak, et al., 2000; Haynes International, 2001) do not evaluate conditions representative of basaltic igneous events. DOE plans to provide an additional evaluation of thermal-mechanical effects on waste package damage in an update to CRWMS M&O (2000e). The current information and the agreements reached between DOE and NRC (Section 3.3.2.5) are sufficient to ensure the necessary information will be available at the time of a potential license application to address these concerns.

3.3.2.4.1.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.2.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess mechanical disruption of engineered barriers (i.e., igneous intrusion) with respect to the characterization and propagation of data uncertainty through the model abstraction.

The number of waste packages directly intersected by a basaltic dike is calculated using a range of dike characteristics summarized in CRWMS M&O (2000a,g). Current total system performance assessment models sample a range of dike length and orientations and the number of dikes per igneous event. These parameter ranges appear reasonably consistent with the underlying technical basis (CRWMS M&O, 1996). Using simple geometric relationships, models then calculate the number of drifts intersected by each sampled dike

[7]Reamer C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

event. For each intersected drift, three waste packages on either side of the dike are assumed to fail on contact with basaltic magma (i.e., Zone 1), whereas the remaining waste packages in the drift (i.e., Zone 2) are assumed to have limited damage (CRWMS M&O, 2000e,g,h). The range sampled in CRWMS M&O (2000b) for the number of waste packages impacted in Zones 1 and 2 is the simple product of the number of drifts intersected per intrusive event and the number of waste packages within each defined geometric zone.

DOE performed a limited number of sensitivity calculations in the Total System Performance Assessment–Site Recommendation relative to mechanical disruption of engineered barriers (CRWMS M&O, 2000b). Small variations in the number of waste packages failed in Zone 1 (i.e., 108 at the 5th percentile, 219 at the 95th percentile) had about a factor of two variation in the probability-weighted dose. Based on this sensitivity, an order-of-magnitude increase in dose is likely for an order-of-magnitude increase in the number of waste packages wholly damaged during an intrusive igneous event. Varying the aperture of end-cap openings in Zone 2 packages from 3.5 to 30 cm^2 [0.54–4.7 in^2] had negligible effects on the probability-weighted dose (CRWMS M&O, 2000b). Large increases in the number of waste packages partially damaged in Zone 2 also had negligible effects on the probability-weighted dose.

Although the processes of magma-waste package interaction are highly complex, DOE developed a deterministic model for waste package damage (CRWMS M&O, 2000b,e,h). Uncertain parameter values, such as waste package material strength properties at sustained temperatures, are not sampled in these models. If DOE develops process-level models to evaluate waste package resilience to igneous events, data uncertainty will need to be evaluated. To address this concern, DOE agreed[8] to evaluate waste package response to stresses from thermal and mechanical effects associated with exposure to basaltic magma. This evaluation will include (i) appropriate at-conditions waste package material strength properties and magma flow paths for the likely duration of an igneous event and (ii) aging effects on waste package material strength properties when exposed to basaltic magmatic conditions for the likely duration of an igneous event.

3.3.2.4.1.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.2.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess mechanical disruption of engineered barriers (i.e., Igneous Intrusion) with respect to the characterization and propagation of model uncertainty through the model abstraction.

CRWMS M&O (2000b) presents several alternative conceptual models for magma flow into open or backfilled drifts. The performance implications of these alternative models, however, are not discussed. For example, CRWMS M&O (2000e) discusses multiple-flow modes that

[8]Reamer C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

pyroclastic flows or liquid magma could follow that result in different rates and extent of magma interaction within and between proposed repository drifts. Only one of those models is evaluated within Total System Performance Assessment–Site Recommendation, which is for flow into and repressurization within each discretely intersected drift (CRWMS M&O, 2000b). A model is developed in CRWMS M&O (2000e) for evolution of potentially corrosive gasses from cooling basaltic magma. Although the gas-flow rate is probably overestimated in this model, the process of degassing-induced corrosion appears supportable based on this model (CRWMS M&O, 2000e). The potential effects of degassing-induced corrosion on waste package performance, however, are not evaluated. Calculations in CRWMS M&O (2000h) assume the waste package walls are a single metal alloy and, thus, do not evaluate the potential for differential thermal expansion or consider that waste packages will be subjected to igneous conditions for many hundreds of hours during the intrusive event. Each of these models has clear alternatives, such as the use of different composition alloys for canister walls and prolonged exposure to igneous conditions, which are expected to affect total system performance significantly. To address these concerns, DOE agreed[9] to evaluate waste package response to stresses from thermal and mechanical effects associated with exposure to basaltic magma. This evaluation will include (i) appropriate at-conditions waste package material strength properties and magma flow paths for the likely duration of an igneous event and (ii) aging effects on waste package material strength properties when exposed to basaltic magmatic conditions for the likely duration of an igneous event.

In summary, alternative conceptual models consistent with available information are not evaluated within the context of total system performance. Uncertainties with existing conceptual models are not quantified or discussed, and the potential effects of these uncertainties are not evaluated . The current information and the agreements reached between DOE and NRC (Section 3.3.2.5) are sufficient to ensure the necessary information will be available at the time of a potential license application to address these concerns.

3.3.2.4.1.5 Model Abstraction Output Is Supported by Objective Comparisons

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.2.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess mechanical disruption of engineered barriers (i.e., igneous intrusion) with respect to model abstraction output being supported by objective comparisons.

Models relevant to igneous effects on mechanical disruption of waste packages in CRWMS M&O (2000a,b,d–i) have not been compared to detailed process-level models, appropriate laboratory or field tests, or natural analogs. Models for the flow of magma into repository drifts (CRWMS M&O, 2000g) are critically dependent on sustaining a debris plug at the end of each intersected drift. The abstracted models used to calculate pressures in the magma-drift system will need to be supported in conjunction with an analysis of debris plug

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[9]Reamer C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

strength before magma flow can be wholly restricted to within an intersected drift. Models that conclude only a limited extent of damage to Zone 2 waste packages will need significant support, including evaluation of an appropriate range of physical conditions and duration of conditions associated with basaltic igneous events. The potential effects of degassing-induced corrosion will also need to be evaluated and verified for all potentially impacted waste packages. Once potential inconsistencies between the abstracted models and comparative data are explained and quantified, the resulting uncertainties will need to be included in total system performance assessment model results.

To address these concerns, DOE agreed[10] to evaluate waste package response to stresses from thermal and mechanical effects associated with exposure to basaltic magma. In addition, DOE agreed to evaluate the response of Zone 3 waste packages, or waste packages covered by backfill or rockfall, if exposed to magmatic gasses at conditions appropriate for igneous events.

3.3.2.4.2 Faulting

3.3.2.4.2.1 System Description and Model Integration Are Adequate

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.2.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess mechanical disruption of engineered barriers (i.e., faulting) with respect to system description and model integration.

DOE excluded all effects of faulting from consideration in the Total System Performance Assessment–Site Recommendation based on the features, events, and processes analyses (CRWMS M&O, 2000c). The exclusion of features, events, and processes related to faulting is primarily based on DOE conclusions of low probability. DOE assumes design parameters can be used to screen features, events, and processes based on low probability if the repository design eliminates or alleviates the features, events, and processes (CRWMS M&O, 2000c, Assumption 5.2). For faulting, the design parameters are fault-setback distances. DOE will position emplacement drifts and waste packages away from faults with future fault slip potential. The setback distance will have to be enough to ensure that faulting will not impact the engineered components. The amount of setback was determined from mechanical and theoretical considerations of fault zone behavior (CRWMS M&O, 2000j).

Determination of appropriate design parameters for faulting, including setback distances, was derived using results from the DOE fault displacement hazard assessment. The probabilistic fault displacement hazard assessment was constructed through the expert elicitation used by DOE to develop a probabilistic seismic hazard analysis (CRWMS M&O, 1998; Stepp, et al., 2001). The expert elicitation results were based on the findings of six expert teams, each consisting of three geoscientists. Fault displacement analyses evaluates the

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[10]Reamer C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

potential hazards of an active fault intersecting vital components of the engineered barrier subsystem, especially waste packages.

For this evaluation of faulting, principal and secondary (or distributed) faulting were considered (as defined in dePolo, et al., 1991). Principal faulting refers to displacement along the main fault zone responsible for the release of seismic energy (i.e., an earthquake). At Yucca Mountain, principal faulting is assumed to occur only along principal faults, mainly block-bounding faults like the Solitario Canyon and Paintbrush Canyon faults. In contrast, secondary or distributed faulting is defined as a rupture of smaller faults, such as the Ghost Dance fault, that occurs in response to the rupture in the vicinity of the principal fault. These two subsets of faults are not mutually exclusive. Faults capable of principal rupture also can undergo secondary faulting in response to faulting on another principal fault. Because principal and secondary faults pose a potential risk to repository performance, both types were considered by DOE. NRC (1999b) provided a review of the methodology used by the DOE expert elicitation to develop an appropriate probabilistic fault displacement hazard assessment.

Staff consider that DOE used conservative assumptions for estimating the probability of faulting and the associated effects on waste packages (NRC, 1999b). The current screening argument used by DOE to exclude faulting from the total system performance assessment in the features, events, and processes analyses and the inputs of fault displacement to the setback calculations (CRWMS M&O, 2000j), however, does not provide an adequate technical basis for staff to consider this subissue closed. In the screening, DOE (CRWMS M&O, 2000c, Assumption 5.5), assumes the median fault displacement values, rather than the mean values, are a more accurate predictor of faulting for low probability faulting events. Assumption 5.5 defined low probability events as those with annual probabilities less than 10^{-6} per year. To address this concern, DOE[11] agreed to provide the appropriate technical basis for use of the median or reevaluation of the features, events, and processes screening based on the mean values, according to the Structural Deformation and Seismicity Key Technical Issue Technical Exchange agreements.

3.3.2.4.2.2 Data Are Sufficient for Model Justification

Overall, the current information is sufficient to assess mechanical disruption of engineered barriers (i.e., faulting) with respect to sufficient data for model justification.

DOE adequately evaluated the nature and amount of faulting and the appropriate range of both principal and secondary faulting hazard sources within the repository block. In addition, DOE adequately determined fault geometry applicable to developing the probabilistic fault displacement hazard assessment. Given present knowledge, the DOE interpretations of faulting from surficial and underground mapping, as presented in CRWMS M&O (1998), are geologically consistent and reasonable.

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[11]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Structural Deformation and Seismicity (October 11–12, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

The experts adequately noted faults as primary or secondary for probabilistic fault displacement hazard assessment. Some fault data taken by DOE from surface outcrops and from the exploratory studies facilities have been confirmed by independent checks by the NRC staff (NRC, 1999b). The variation of fault orientation data is within acceptable limits for normal geologic work. Field checks of fault locations, orientations, displacements, and other selected geometric features are generally in close agreement with the DOE observations and interpretations.

3.3.2.4.2.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.2.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess mechanical disruption of engineered barriers (i.e., faulting) with respect to the characterization and propagation of data uncertainty through the model abstraction.

DOE has not yet provided information needed to justify the probability distributions and bounding assumptions of faulting or reasonably to account for the associated uncertainties and variabilities. DOE developed models of faulting (CRWMS M&O, 2000j) based on a probabilistic fault displacement hazard assessment (CRWMS M&O, 1998; Stepp, et al., 2001). In those models, values for fault displacements for probabilities less than 10^{-6} annual exceedance per year are based on the median rather than the mean values from the probabilistic fault displacement hazard assessment curves (CRWMS M&O, 2000c, Assumption 5.5). As discussed in Section 3.3.2.4.2.1, use of the median rather than the mean values is not supported by sufficient technical basis (also see Section 3.2.2). To address this concern, DOE agreed[12] to provide the necessary information or use the mean in future analyses.

3.3.2.4.2.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

At the time this report was prepared, the effects of faulting were excluded from the total system performance assessment model abstraction for disruptive events. DOE agreed to address the NRC concerns, as discussed in Sections 3.3.2.4.2.1 and 3.3.2.4.2.3. Depending on the resolution of these concerns, the effects of faulting will be included or excluded from the total system performance assessment model abstraction for disruptive events.

3.3.2.4.2.5 Model Abstraction Output Is Supported by Objective Comparisons

At the time this report was prepared, the effects of faulting were excluded from the total system performance assessment model abstraction for disruptive events. DOE agreed to address the NRC concerns, as discussed in Sections 3.3.2.4.2.1 and 3.3.2.4.2.3. Depending on the

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[12]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Structural Deformation and Seismicity (October 11–12, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

resolution of these concerns, the effects of faulting will be included or excluded from the total system performance assessment model abstraction for disruptive events.

3.3.2.4.3 Seismicity

3.3.2.4.3.1 System Description and Model Integration Are Adequate

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.2.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess mechanical disruption of engineered barriers (i.e., seismicity) with respect to system description and model integration.

The DOE calculation of seismic effects on the engineered barrier subsystem relies in part on the input seismic loads calculated from the DOE probabilistic seismic hazard analysis. The probabilistic seismic hazard analysis methodology has been identified by NRC in 10 CFR 100.23 as an appropriate approach to address uncertainties associated with ground motions. DOE outlined the methodology used for its probabilistic seismic hazard analysis in DOE (1994), which was accepted, in principle, by NRC.[13] The methodologies discussed in NRC (1997) also offer acceptable approaches for evaluating the probabilistic seismic hazard at Yucca Mountain. For postclosure performance, the seismic hazard curve is an important input parameter for assessing rockfall and drift collapse in the emplacement drifts because of earthquake-induced ground shaking.

See the discussion on Effect of Rockfall and Drift Collapse in Section 3.3.2.4.4.1 for the staff assessment of consequences to the engineered barrier subsystem components caused by seismic events. The following sections discuss the elements of a probabilistic seismic hazard analysis.

<u>Seismic Source Characterization</u>

DOE characterized seismic sources in CRWMS M&O (1998) and in Stepp, et al. (2001). In this analysis, DOE used six teams of experts. Each team consisted of three specialized geoscientists with expertise in either paleoseismology, Basin and Range structural geology, or Basin and Range seismology. To assess seismic sources, the teams mainly relied on information provided by the U.S. Geological Survey, DOE, and related Yucca Mountain studies augmented by published literature. In addition, the teams assembled for six workshops, at which the experts exchanged information on seismic sources and participated in additional discussions with other external experts. Details of the workshops are given in the probabilistic seismic hazard analysis final report (CRWMS M&O, 1998; Stepp, et al., 2001). Elicitation methodology and related issues are treated separately in Section 5.4, Expert Elicitation Acceptance Criteria.

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[13]Bell, M.J. “Issue Resolution Status Report on Methodology to Assess Fault Displacement and Vibratory Ground Motion Hazard at Yucca Mountain.” Letter (July 25) to S. Brocoum, DOE. Washington, DC: NRC. 1996.

Geologic and Tectonic Setting: The expert teams considered all the viable tectonic models, and aspects of all the modes were incorporated into all the expert elicitation teams' identifications of seismic sources. The teams relied, to varying degrees, two tectonic models: (i) seismogenic detachment faults as potential seismic sources (i.e., Deep Detachment Fault Tectonic Model) and (ii) hidden or buried strike-slip faults with associated cross-basin faults as potential seismic sources (i.e., Amargosa Desert Fault Model). In addition, planar-block bounding faults were also considered in the assessments made by the six expert elicitation teams. Although presented to the experts at the workshops, strain rate values derived from global positioning satellite measurements were not explicitly considered by any teams as a viable alternative to estimations of the seismic hazard.

Fault and Areal Sources: Seismic sources in CRWMS M&O (1998) and in Stepp, et al. (2001) consisted of two types: fault sources and areal source zones. The approach used by DOE to identify potential seismic sources follows standard practice for seismic hazard assessments of sites west of the Mississippi River where better exposure of bedrock and greater tectonic activity make identification of fault sources easier to discern.

Fault sources are used in the hazard assessment to account for expected seismicity on known or suspected fault traces. Uncertainty in fault sources is accounted for by alternative interpretations of fault length, fault dip, closest approach to the site, depth within the seismogenic crust, and possible kinematic linkage with other faults. In the probabilistic seismic hazard analysis calculations, earthquakes are assumed to occur randomly along the fault surface, constrained by the size of the rupture area. Rupture area and rupture dimensions are specified by empirical relationships based on magnitude (e.g., Wells and Coppersmith, 1994).

Fault sources were identified by the expert teams from published U.S. Geological Survey and DOE maps and reports (U.S. Geological Survey, 1996; Piety, 1995; Anderson, et al., 1995a,b; Simons, et al., 1995), published scientific literature (Scott, 1990; Zhang, et al., 1990; Reheis and Dixon, 1996; Reheis and Sawyer, 1997), and CNWRA publications (Ferrill, et al., 1996; McKague, et al., 1996). In addition, the experts benefitted from detailed discussions at several of the probabilistic seismic hazard analysis workshops, in which summaries of fault sources and tectonic models were presented by project and external scientific experts. The expert teams also visited many of the sources during a field trip held as part of Probabilistic Seismic Hazard Analysis Workshop #3 (November 18–21, 1996).

Local and regional Yucca Mountain tectonics also were considered when identifying potential fault sources. Considerations included sources from proposed buried or otherwise cryptic strike-slip faults (Schweickert and Lahren, 1997) and seismogenic detachment faults (Wernicke, 1995). Uncertainty in the sources, both in geometric characteristics and likelihood of activity, was accounted for by the logic tree structure of the probabilistic seismic hazard analysis, in which various models of faulting and fault activity were weighted according to the opinions of the experts.

The expert teams considered 87 fault sources or combinations of fault sources (CRWMS M&O, 1998, Table 4-2). These sources included 30 faults or combinations of fault sources local to Yucca Mountain (within Yucca Mountain or in the adjacent basins), 51 regional faults or combinations of faults in the Yucca Mountain region {generally within a radius of approximately 100 km [62 mi] of the site}, and 6 faults or combinations of fault sources inferred from the tectonic models. Included in this list are faults identified through independent analysis of Type I faults by NRC and CNWRA staffs (McKague, et al., 1996, Section 4.1.1). For example, one of the expert teams considered 41 faults as individual fault sources (CRWMS M&O, 1998, Tables AAR–1 and AAR–4). All are Type I faults. This same expert team also demonstrated how nonindividual Type I fault sources contribute to seismicity as background or areal seismic sources.

In contrast to fault sources, areal sources represent areas of distributed or background seismicity in which no geologic or geophysical evidence can tie earthquakes to known faults. In this way, areal sources account for earthquakes that occur on unidentified or unidentifiable fault sources. Areal sources are typically developed to represent earthquakes with magnitudes that may not necessarily cause surface rupture.

In the DOE probabilistic seismic hazard analysis (CRWMS M&O, 1998; Stepp, et al., 2001), experts relied on empirical relationships that relate surface rupture to earthquake magnitude (e.g., Wells and Coppersmith, 1993, 1994; dePolo, 1994; U.S. Geological Survey, 1996; CRWMS M&O, 1998, Figure 4-11; Stepp, et al., 2001). Given these data, there is greater than an 80-percent probability that **M**6.5 earthquakes will rupture the surface, while there is less than a 20-percent chance that **M**5.5 earthquakes will rupture the surface.

The boundaries of areal sources are drawn to define areas with relatively uniform seismicity and maximum magnitude, generally defined by the historic seismic record. All expert teams considered one to three areal source zones. For most teams, the source zones were used to capture background seismicity; and, thus, the maximum magnitude for areal sources close to Yucca Mountain was less than for those sources farther away thus the expert teams felt the fault source characterization at Yucca Mountain was superior to that in the surrounding regions. Some of the expert teams also included an explicit volcanic areal source term to explicitly account for seismic activity related to volcanism.

<u>Historic Seismicity:</u> The DOE facilitation team provided a single earthquake catalog to the expert teams. This catalog was compiled from 12 regional catalogs (CRWMS M&O, 1998, p. G-2). The initial catalog contained 271,223 earthquakes of **M**0.5 and larger for the period 1868–1996. This initial catalog was modified in three ways. First, all the magnitudes were converted to moment magnitude ($\mathbf{M}_\mathbf{W}$). Second, information on earthquakes from nuclear testing was removed based on compilations of all known nuclear tests. Third, foreshocks and aftershocks information was removed using two standard declustering methods (Youngs, et al., 1987; Veneziano and van Dyck, 1985). The Little Skull Mountain sequence was used to test the effectiveness of the two declustering techniques. Results show that the Veneziano and van Dyck (1985) method was better able to isolate foreshocks and aftershocks. After modifications, the resulting catalogs contained between 26,250 [Veneziano and van Dyck (1985) method] and 31,147 [Youngs, et al. (1987) method] earthquakes covering a circular area with a 300-km [186-mi] radius centered on Yucca Mountain.

<u>Maximum Magnitude:</u> The maximum magnitude earthquake is the largest earthquake that can be produced on a fault or in an areal source, regardless of its frequency of occurrence. For

fault sources, the expert teams used empirical scaling relationships that relate maximum magnitude to the physical dimensions of the fault. Maximum magnitude was derived from fault length, rupture area, maximum surface displacement, and average surface displacement. In some cases, the expert teams modified their maximum magnitude estimate by considering slip rate as well as rupture dimensions following Anderson, et al. (1996). In addition, the experts considered rupture area and average slip on the fault to estimate seismic moment, which was then converted to maximum magnitude using the relationships in Hanks and Kanamori (1979). For areal sources, the experts estimated the maximum magnitude earthquake based on the largest fault in the areal source not explicitly modeled as a fault source. Alternatively, the experts relied on the empirical relationships that relate surface rupture to earthquake magnitude based on empirical data (e.g., Wells and Coppersmith, 1994; dePolo, 1994; U.S. Geological Survey, 1996; CRWMS M&O, 1998, Figure 4-11).

Incorporation of Alternatives and Uncertainty: The elicitation used a standard logic tree approach to delineate the alternative interpretations into a coherent framework and to incorporate uncertainty. The first branch of the tree identified alternatives of faults based on different interpretations of local and regional tectonics derived from the suite of viable tectonic models. Subsequent branches evaluated alternatives in fault-specific characteristics such as fault linkage, segmentation, maximum magnitude, activity rate, and seismogenic depth (CRWMS M&O, 1998, Figures 4-2 and 4-3, example logic tree representations).

Earthquake Recurrence

The recurrence rates for the faults were estimated using either recurrence intervals or slip rates. Recurrence and slip rates were primarily derived from paleoseismic data obtained by the U.S. Geological Survey in detailed investigations of faulting in the Yucca Mountain region (CRWMS M&O, 1998). Additional constraints were derived from geologic data that estimate longer-term slip rates (e.g., Stamatakos, et al., 1997).

For fault sources, two methods were used by the experts to estimate recurrence. The first was to estimate the frequency of the largest earthquakes on the fault, and then specify the magnitude distribution function for the remaining earthquakes based on a particular recurrence model. The experts used three such recurrence models: (i) characteristic (Schwartz and Coppersmith, 1984), (ii) truncated exponential (Gutenberg and Richter, 1954), and (iii) modified truncated exponential. The second approach was to translate the slip rate into a seismic moment rate, and then partition the moments into earthquakes of various magnitudes according to a magnitude distribution model (Wesnousky, 1986).

For areal sources, the expert teams used the earthquakes in the catalog of historic earthquakes. The distribution of earthquake magnitudes in each areal source zone was interpreted following an exponential distribution (Gutenberg and Richter, 1954). Recurrence relationships for each zone were then estimated following a truncated exponential magnitude distribution to account for the maximum magnitude earthquake (Cornell and Van Marke, 1969).

Ground Motion Attenuation

In a probabilistic seismic hazard analysis, ground motion attenuation models (i.e., mathematical relationships between ground motion and earthquake magnitude, distance, site conditions, and style of faulting) are required to estimate the levels of ground motion that may occur at a site. An expert elicitation process was used (CRWMS M&O, 1998) to develop ground motion estimates for the Yucca Mountain probabilistic seismic hazard analysis. Because of the limited availability of sufficient strong motion data to develop robust empirical ground models specific to the regional and local geologic conditions at Yucca Mountain and the seismologic characteristics of nearby active faults, a group of ground motion experts convened to evaluate input for developing a probabilistic ground motion model specific to the regional conditions of the western Basin and Range, in proximity to Yucca Mountain. In the context of these circumstances, expert elicitation is reasonable and appropriate (NRC, 1997). In addition, an expert elicitation provides the opportunity to incorporate supplementary sources of information into the development of ground motion models such as expert interpretations of related and indirect information on strong ground motion.

In the Yucca Mountain probabilistic seismic hazard analysis, the experts were to provide input (i.e., data, scientific interpretations, and estimates of parameter uncertainties) as part of the development of a probabilistic ground motion attenuation model. Consistent with the overall approach in the probabilistic seismic hazard analysis, the probabilistic ground motion attenuation model includes estimates of aleatory and epistemic uncertainties in ground motion levels. The aleatory uncertainty quantifies the inherent or natural randomness of ground motions (e.g., variability not explained by the ground motion model). The aleatory or random uncertainty is a probabilistic variable that results from natural physical processes and is inherent to the unpredictable nature of future events. For example, the size, location, and time of the next earthquake and the details of the ground motion are examples of quantities considered aleatory. Aleatory uncertainty cannot be reduced by collecting additional data. Epistemic uncertainty quantifies the uncertainty associated with the estimate of model parameters that are the result of limited data and lack of knowledge about parameters such as the physical processes involved in fault rupture and its energy release properties and the resultant wave propagation characteristics. In the Yucca Mountain probabilistic seismic hazard analysis, a probabilistic ground motion model was developed by each of seven ground motion experts. In aggregate, the seven models were intended to represent (probabilistically) the current state of knowledge with regard to ground motions that can occur at the Yucca Mountain site because of earthquakes.

Elements of the Probabilistic Ground Motion Model

The probabilistic ground motion model used in the Yucca Mountain probabilistic seismic hazard analysis predicts aleatory and epistemic uncertainties in ground motion as a function of earthquake magnitude, source-site distance, and style of faulting. The model consists of the following the elements:

- Ground Motion—Mathematical relationship that defines the variation of the mean log (median) ground motion (denoted as $\boldsymbol{\mu}$) as a function of earthquake magnitude, source-site distance, and style of faulting. The relationship is defined by model coefficients derived from input provided by ground motion experts.

- Aleatory Model—The aleatory variability in ground motion is defined by a lognormal distribution whose parameters are a median (of 1.0) and a logarithmic standard deviation (denoted as $\boldsymbol{\sigma}$).

- Epistemic Model—This model consists of two parts. The first part defines the epistemic uncertainty in the parameters of the median ground motion and aleatory model. Uncertainty in the model parameters, $\boldsymbol{\mu}$ and $\boldsymbol{\sigma}$, is defined by lognormal distributions for each. The second part of the epistemic model is the uncertainty that arises from the alternative ground motion models as derived from the input provided by each of the ground motion experts.

In aggregate, the probabilistic ground motion model is intended to provide a measure of the state of knowledge with respect to the assessment of ground motions at Yucca Mountain. To be valid, expert judgments in an expert elicitation must be traceable and technically defensible (NRC, 1996, 1997).

Spectral Decay (Kappa)

During review of the probabilistic seismic hazard analysis, specific issues were raised regarding the definition of the shallow crustal velocity near the free surface and the value of crustal kappa used for ground motion estimation at Yucca Mountain. These issues were raised because of the differences between the site condition at Yucca Mountain and the representations of the empirical strong motion database used (mainly California). There is a great difference in shear wave velocities, deep crustal damping [Q(f)], and shallow crustal {top 1–2 km [0.62–1.24 mi]} damping value (kappa) between California and Yucca Mountain. Kappa, defined as the spectral decay, is primarily caused by subsurface geological structures near the site. It is a smaller value for hard rock sites than for soft rock sites. The value of kappa estimated by Su, et al. (1996) for the southwestern part of the Nevada Test Site ranged from 0.005 to 0.024 seconds. In the probabilistic seismic hazard analysis, a value of 0.0186 second was used. DOE agreed[14] that if new studies find the median value of kappa for material with shear wave velocity below 1,900 m/s [6,234 ft/s] is different from 0.0186 second, median attenuation model will be adjusted. Potential adjustment of the median attentuation model will be addressed by DOE in Topical Report #3.

Vibratory Ground Motion Hazard Results

Median and fractile ground acceleration and aleatory and epistemic uncertainties for various earthquake magnitudes, sources-to-site distances, and different fault styles were estimated by the experts. Uncertainties in seismic source characterization and ground motion attenuation

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[14]Schlueter, J.R. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Structural Deformation and Seismicity (October 11–12, 2000).” Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

relations were quantified by considering inputs from six seismic source fault displacement expert teams and seven ground motion experts. Each team and each expert provided their own assessment of uncertainty. The moment magnitude, $\mathbf{M_w}$, used in the probabilistic seismic hazard analysis ranged from 5.0 to 8.0 for normal and strike-slip faulting, and the distances examined were from 1 to 160 km [0.62 to 99 mi].

The probabilistic hazard for vibratory ground motion was calculated for peak ground acceleration, peak ground velocity, uniform hazard spectrum, and spectral accelerations at frequencies ranging 0.3–20 Hz. It was found that at 5–10 Hz, or high frequencies, the ground motions are dominated by earthquakes of magnitudes less than 6.5 and distances less than 15 km [9.3 mi]. At lower frequencies, 1–2 Hz, the ground motions are dominated by large events beyond distances of 50 km [31 mi]. The recurrence models contributed most to the uncertainty in the ground motion hazard, while geometric fault parameters were minor contributors to uncertainty. It was found that at 10 Hz, the dominant sources for seismic hazard ground motion are Paintbrush Canyon, Iron Ridge, and Solitario Canyon faults, and the host areal seismic source zone. For 1-Hz ground motion, the dominant seismic hazard comes from Death Valley–Furnace Creek faults.

The vibratory ground motion hazard calculations were performed for each expert proposed attenuation equation and seismic source parameters. In general, the most ground motion contributors to uncertainty in the hazard were σ_μ and σ_σ, within expert uncertainties, rather than expert-to-expert uncertainties. The total uncertainty caused by ground motion is larger than the uncertainty caused by the seismic source characterization. Combining the experts' hazard curves, giving each expert equal weight, a set of integrated hazard curves were produced. The integrated results, based on input from the six expert teams and the seven ground motion expert represent the seismic hazard and its associated uncertainty at Yucca Mountain. The separation between the 15^{th}- and 85^{th}-percentile curves conveys the effects of the epistemic uncertainty on the calculated hazards. It should be noted these hazard curves were estimated at a reference rock outcrop on the surface, on a reference site at the same elevation as the repository.

Seismic Hazard Analysis

An evaluation of the seismic and ground motion characterization of CRWMS M&O (1998) and Stepp, et al. (2001) concluded that the seismic source characterization is adequate, and sufficient information exists for staff to review this aspect of the probabilistic seismic hazard analysis for a potential license application.

The ground motion characterization component of the Yucca Mountain seismic hazard analysis cannot be closed, however, until additional information is provided by DOE. Specifically, DOE agreed[15] to provide information to address staff concerns regarding (i) the ground motion expert elicitation process (see the discussion in Section 5.4, Expert Elicitation); (ii) site specific seismic

[15]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Structural Deformation and Seismicity (October 11–12, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

data, including input to the site response model (to be documented in the Seismic Design Inputs Report and Seismic Topical Report #3); (iii) Assumption 5.5 of CRWMS M&O (2000c), which assumes the median fault displacement values, rather than the mean values, are more accurate predictors of faulting for annual probabilities less than 10^{-6} per year (see earlier discussion of faulting); and (iv) incorporation of seismicity into cladding failure scenarios (see discussion in Section 3.3.4.4.3). Staff review of the Yucca Mountain ground motion models raises questions about the scientific basis for several of the expert ground motion assessments and the completeness elicitation feedback process. In particular, examination of several expert ground motion models illustrates that large differences exist between the experts, regarding predicted ground motions and epistemic and aleatory uncertainties. In some cases, staff noted wide diversity between experts and large variability within individual expert models. For instance, the 5-and 95-percent confidence limits pertaining to the estimate of the median ground motion for an earthquake of a given magnitude-distance and style of faulting for two cases of the expert models are shown in Table 3.3.2-1.

Table 3.3.2-1. Epistemic Uncertainty in the Median Peak Ground Acceleration Ground Motion—One Expert Model

				Fractiles Based on Epistemic Uncertainty—Median (g)			
Case	Magnitude	Distance (km) [mi]	Style of Faulting	0.05	0.50*	0.95	Ratio (95/5)
1	6.5	1 [0.62]	Normal	0.11	0.50	2.33	21.2
2	6.5	10 [.62]	Normal	0.11	0.28	0.73	6.64

*Median scaled from attenuation model plots in CRWMS M&O, “Probabilistic Seismic Hazard Analyses for Fault Displacement and Vibratory Ground Motion at Yucca Mountain, Nevada, Final Report.” WBS Number 1.2.3.2.8.3.6. Las Vegas, Nevada: CRWMS M&O. 1998.

The results provided in Table 3.3.2-1 suggest a large uncertainty in the estimate on the median ground motion. For instance, in Case 1, the expert suggests there is a 5-percent chance the true estimate of the median ground motion at a site 1 km [0.62 mi] from the **M**6.5 event is greater than 2.33g. In other words, the entire attenuation relationship shifts upward to this ground motion level. A similar conclusion can be derived for the lower estimate of the median ground motions. That is, there is a 5-percent chance the true estimate of the median ground motion at a site 1 km [0.62 mi] from the **M**6.5 event is less than 0.11g. For this expert, this observation is particularly interesting because his median estimate for the cases considered in the table is also the highest among the seven experts. In addition, the epistemic uncertainty provided by this expert is significantly larger than the variation in the range of median values predicted by the other experts.

As a measure of the technical integrity of the expert elicitation process and the scientific evaluation of individual expert assessments, and in light of these observations about the variability of their results, staff examined the bases for the ground motion models and results as documented in available reports (e.g., CRWMS M&O, 1998). The review raised a series of questions about the feedback-documentation part of the probabilistic seismic hazard analysis expert elicitation process

- Did the process involve training the experts, and were measures taken to demonstrate the experts understood, with reasonable assurance, the applicable probabilistic concepts and their implementations in the ground motion model?

- What was the process (i.e., technical evaluations) the experts undertook individually and within the context of workshops to affirm their understanding and concurrence with the probabilistic ground motion model derived from their input, and was the process adequate? For instance, did the facilitation teams provide the experts with an accurate awareness of the 5–95 fractile estimates of the median ground motion?

- In the example just given, where is the specific documentation of the scientific basis for the experts' agreement with the results? Although such information may exist, it is not available in CRWMS M&O (1998).

At the Structural Deformation and Seismicity Technical Exchange Meeting (October 2000), DOE provided a brief summary of the approach to expert elicitation used in the ground motion part of the probabilistic seismic hazard analysis. As part of the agreements made at that technical exchange, DOE agreed[16] to provide additional information about the ground motion elicitation process.

The information provided by many of the experts at the April 1997 workshop (mentioned previously) is a description of the procedure they followed to generate their inputs rather than providing the scientific basis for their assessments. In CRWMS M&O (1998), the individual ground motion expert reports contain statements the experts accepted the models derived by the facilitation team from their input. There was, however, no information provided as part of CRWMS M&O (1998) or later submissions that indicated the experts evaluated or reviewed the acceptability of the probabilistic ground motion models developed from their ground motion input parameters. In the absence of the necessary documentation, two questions remain unanswered:

- Were the experts aware the 5-and 95-percent confidence limits predicted by their models led to high estimates of median ground motion?

- Did the experts make an attempt to critically examine the distribution on their median ground motion (for a given ground motion measure) such that they were aware of the range and meaning of the epistemic uncertainty? The information presented in CRWMS M&O (1998) neither demonstrates the experts' understanding of the probabilistic ground motion model derived from their input nor describes the

[16]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Structural Deformation and Seismicity (October 11–12, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

methodology each expert used to assess the probabilistic estimates of ground motions made by their model.

In summary, to address the aforementioned concerns, DOE agreed[17] to provide the appropriate technical bases and document the process used to provide feedback to experts following the elicitation process.

3.3.2.4.3.2 Data Are Sufficient for Model Justification

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.2.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess mechanical disruption of engineered barriers (i.e., seismicity) with respect to sufficient data for model justification.

The seismic sources identified by the experts in the probabilistic seismic hazard analysis adequately characterize the potential sources of seismicity that will contribute to the anticipated peak and spectral ground motions at Yucca Mountain resulting from future earthquakes in the Yucca Mountain region based on the following observations:

- The seismic source characterization adequately incorporated the geologic and tectonic settings of the region into the probabilistic seismic hazard analysis. The range of tectonic models and the implications of those models to the probabilistic seismic hazard analysis are geologically consistent and entirely compatible with the current understanding of the Yucca Mountain tectonic framework and with the Basin and Range.

- Fault and areal sources were adequately identified by DOE. For example, comparison of Type I faults (McKague, et al., 1996) with the DOE lists of relevant faults (U.S. Geological Survey, 1996) shows general agreement, especially on the most important sources to the overall seismic hazard. DOE (U.S. Geological Survey, 1996) uses the terms relevant and potentially relevant in describing faults. At this time, staff consider all known candidate Type I faults in the Yucca Mountain region have been evaluated adequately. Staff found differences between DOE and NRC classifications of particular faults rooted in three parameters: fault trace length, attenuation function, and use of median or 84th percentile groundmotion: for identification of those faults that will exceed the 0.1g cutoff criterion. These differences lead to only minor differences in predicted ground motions (<0.1g) and are not considered significant to overall estimates of repository performance.

- The earthquake historical data and paleoseismicity were adequately characterized by DOE on the site and in the region. That record included approximately 30,000 earthquakes from historical earthquake catalogs used by the experts in the

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[17]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Structural Deformation and Seismicity (October 11–12, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

probabilistic seismic hazard analysis. The earthquake magnitudes used in the analysis were corrected to a common moment magnitude ($\mathbf{M}_W$) and ranged from $\mathbf{M}_w$5.0 to $\mathbf{M}_w$8.0. Information on earthquakes from nuclear testing was removed based on compilations of all known nuclear tests. Foreshocks and aftershocks information was removed using standard declustering methods (Youngs, et al., 1987; Veneziano and van Dyck, 1985). The declustering techniques were tested for effectiveness by analysis of the Little Skull Mountain sequence, which had independently known foreshock, main shock, and aftershock sequences. Staff consider maximum magnitudes are reasonable for the fault sources based on established and published scaling relationships of rupture dimensions of the source. For example, empirical relationships between magnitude versus rupture length, rupture area, and maximum surface displacement (e.g., Wells and Coppersmith, 1994) were appropriately used to estimate maximum magnitude. Estimates of the rupture area and average slip on the fault were used by the experts to calculate the maximum magnitude event (Anderson, et al., 1996). For areal sources, the maximum magnitude earthquake was based on the maximum earthquake to occur within the area. The magnitude ranges used by the experts were based on moment magnitude (i.e., $\mathbf{M}_w$).

- Activity and fault slip rates were reasonably estimated by DOE. For example, recurrence and slip rates were primarily derived from paleoseismic data obtained by the U.S. Geological Survey detailed investigations of faulting in the Yucca Mountain region (CRWMS M&O, 1998). Additional constraints were derived from geologic data that estimated longer-term slip rates (e.g., Stamatakos, et al., 1997).

- Clustered events were adequately considered by DOE. For example, multiple rupture scenarios were derived (U.S. Geological Survey, 1996) and incorporated by the experts in the probabilistic seismic hazard analysis (CRWMS M&O, 1998).

In contrast, additional information pertaining to ground motion modeling is needed before staff can consider this acceptance criterion closed for seismicity (see the discussion in Section 3.3.2.4.3.1). To address this concern, DOE agreed[18] to provide the needed information.

3.3.2.4.3.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.2.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess mechanical disruption of engineered barriers (i.e., seismicity) with respect to the characterization and propagation of data uncertainty through the model abstraction.

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[18]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Structural Deformation and Seismicity (October 11–12, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

DOE has not provided information to justify the probability distributions and bounding assumptions of ground motion or to account reasonably for the associated uncertainties and variabilities. Similar to faulting, DOE developed models for seismicity and ground motion based on a probabilistic seismic hazard analysis (CRWMS M&O, 1998; Stepp, et al., 2001). In those models, values for ground motion probabilities less than 10^{-6} annual exceedance per year are based on the median rather than the mean values from the probabilistic seismic hazard analysis curves (CRWMS M&O, 2000c, Assumption 5.5). As discussed in Section 3.3.2.4.3.1, the adequacy of the characterization and propagation of uncertainty associated with the use of the median rather than the mean values is not supported by sufficient technical basis (also see Section 3.2.2).

In addition, staff review of the probabilistic seismic hazard analysis noted insufficient technical bases with regard to the ground motion expert elicitation (see Section 3.3.2.4.3.1). To address this concern, DOE agreed[19] to provide this information.

3.3.2.4.3.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

At the time this report was prepared, effects of seismicity were excluded from the total system performance assessment model abstraction for disruptive events. DOE agreed to address the NRC concerns, as discussed in Sections 3.3.2.4.3.1, 3.3.2.4.3.2, and 3.3.2.4.3.3. Depending on the resolution of these concerns, the effects of seismicity will be included or excluded from the total system performance assessment model abstraction for disruptive events.

3.3.2.4.3.5 Model Abstraction Output Is Supported by Objective Comparisons

At the time this report was prepared, effects of seismicity were excluded from the total system performance assessment model abstraction for disruptive events. DOE agreed to address the NRC concerns, as discussed in Sections 3.3.2.4.3.1, 3.3.2.4.3.2, and 3.3.2.4.3.3. Depending on the resolution of these concerns, the effects of seismicity will be included or excluded from the total system performance assessment model abstraction for disruptive events.

3.3.2.4.4 Rockfall and Drift Collapse

3.3.2.4.4.1 System Description and Model Integration Are Adequate

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.2.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess mechanical disruption of engineered barriers (i.e., rockfall and drift collapse) with respect to system description and model integration.

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[19]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Structural Deformation and Seismicity (October 11–12, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

Repository Safety After Permanent Closures

According to CRWMS M&O (2000c; 2001a,b), the consequences of rockfall and drift collapse are not being considered in the mechanical disruption of engineered barriers model abstraction for the DOE Total System Performance Assessment Code. The technical bases for this screening decision are provided in an analysis and model report (CRWMS M&O, 2000k) and CRWMS M&O (1999, 2000l) calculation reports. The detailed discussion that follows conveys the results of the staff review of these documents and the rationale for their findings. In summary, the staff review determined DOE

- Underestimated the probability, size, and areal coverage of potential discrete rock blocks that may be dislodged from the drift wall during earthquakes or from natural degradation of the drift wall rock mass

- Underestimated the probability, magnitude, and areal coverage of potential drift collapse

- Did not consider, in an acceptable manner, the potential consequences of rockfall and drift collapse on the engineered barrier subsystem

The effects of rockfall and drift collapse on repository performance will be manifested through changes in seepage characteristics and engineered barrier subsystem component temperatures, seismic response characteristics, near-field chemistry, corrosion rates, and functional capabilities (e.g., water infiltration pathways through breached drip shields).

Occurrence of Rockfall and Drift Collapse

The current DOE position on the occurrence of rockfall and drift collapse (CRWMS M&O, 2000m) is summarized as follows:

> Assuming complete degradation of the ground-support system at closure, time-dependent reduction in joint cohesion, thermal stresses, and seismic events combined will generate rockfall in less than 2.5 percent of the total length of emplacement drifts within 10,000 years after closure.

The DOE position contrasts with the following opinion of a DOE expert panel on drift stability:

> All drifts are likely to collapse in the fullness of time because of the severity of the THM [thermal-hydrological-mechanical]-driving gradients after emplacement ... (Brekke, et al., 1999, p. 3-16).

The DOE position was based on analyses documented in the CRWMS M&O (2000k) analysis and model report, which concluded that the emplacement drifts would experience only negligible rockfall and would essentially retain their as-built shape and size through the 10,000-year period of regulatory concern. The analyses were conducted using a computer code based on the key-block model in which a rock mass intersected by an opening is modeled as a network of rigid blocks and block-bounding fractures. In the model, a block may slide along its bounding fractures being influenced by gravitational force if sliding of the block is kinematically possible. The blocks exposed at the intersection with the opening and geometrically constrained in such a way that their sliding into the opening is kinematically possible are referred to as the key blocks. The sliding of all other blocks is kinematically impossible because of being restrained directly or indirectly by the key blocks. Therefore, the stability of the opening can be assured by preventing failure (i.e., sliding and eventual detachment from the network) of the key blocks. The driving force that may cause failure arises from gravity, and the resistance to failure is provided by the shear strength of the block-bounding fracture surfaces.

The key-block model does not have a mechanism to include a system of internal forces, such as may arise from a temperature distribution (thermal loading), earthquake (seismic loading), or other kinds of stress-generating processes. Furthermore, because blocks are treated as rigid in the mathematical formulation of the key-block model, the potential fracturing of blocks, which can have a significant effect on failure modes in a highly stressed rock mass, and the internal deformation of blocks, which has significant effects on fracture-surface stress, are not accounted for in key-block analysis. DOE indicated that some shortcomings of the key-block model were overcome in the drift degradation analysis (CRWMS M&O, 2000k) through the following procedures.

- The value of the cohesion parameter for fracture surfaces (i.e., the shear-strength intercept parameter of the Mohr-Coulomb strength criterion) was reduced from 0.86 Mpa [125 psi] to 0.01 MPa [1.45 psi] to represent thermal loading and time-dependent degradation of fracture surfaces.

- The value of friction angle for fracture surfaces was reduced by 8.0, 16.7, and 23.3 degrees, to represent seismic ground motions with 0.14, 0.30, and 0.43g peak ground accelerations. The value of friction-angle reduction in each case was calculated as the arc tangent of the respective peak ground accelerations. The peak ground acceleration values of 0.14, 0.30, and 0.43g are intended to represent the 1,000-, 5,000-, and 10,000-year earthquakes.

The rationale for the DOE approach is that the additional shear stress induced on fracture surfaces from a temperature distribution (thermal loading) or earthquake (seismic loading) and the weakening of fracture surfaces by time-dependent degradation can all be represented by the specified reduction of the cohesion and friction-angle parameters. DOE did not present a satisfactory mathematical basis to relate the cohesion reduction to the temperature distribution or the friction-angle reduction to the seismic loading to support an argument that the applied fracture-strength reductions appropriately represent the thermal and seismic loadings for the proposed repository.

Although it is theoretically possible to represent the effect of thermally induced shear stress on a fracture surface through a reduction of the fracture-surface strength, there are important requirements imposed by basic solid-mechanics principles that must be satisfied to apply the procedure satisfactorily. Because thermal stress is a tensor variable, the scalar parameter used to replace its effect must be mathematically tied to the components of the tensor, which, in turn, are dependent on the temperature, temperature gradient, mechanical boundary conditions, and mechanical properties. For this reason, the magnitude of the applied strength reduction would be expected to vary with the thermal load, time, location relative to the heated drift, fracture orientation, and rock-mass mechanical properties. As discussed in

Section 2.1.7.3 of this report, thermally induced rock failure at the proposed repository would likely be dominated by slip on subhorizontal fractures in the roof and floor areas of the drifts and in the pillars and slip on vertical fractures in the sidewall areas. The mechanisms of potential failure are controlled by the emplacement geometry, however, the actual occurrence of thermally induced rock failure would be determined by the strength and stiffness of the intact rock and fractures. None of these characteristics of thermally induced failure can be simulated correctly by representing thermal load as a constant cohesion reduction applied uniformly in a key-block model.

A similar argument can be made regarding the representation of seismic loading using a constant friction-angle reduction applied uniformly in the model. The appropriate friction-angle reduction would vary with the fracture orientation and with several characteristics of seismic ground motion that cannot be represented with peak ground acceleration only (e.g., frequency, duration, and direction of the associated particle motion).

The DOE expert panel on drift stability also noted the limitations of key-block modeling. Having identified rock raveling of small pieces of rock around the boundary of the drifts as a potentially important failure mechanism, the panel noted (referring to a set of illustrative numerical analyses conducted by the panel)

> These analyses do not support the application of key-block modeling to evaluate potential excavation degradation. The key-block approach does not examine subsequent behavior of a system of blocks or redistribution of loads. The raveling degradation may progress as a consequence of stress and/or temperature changes and other factors, which cannot be directly represented in a key-block model (Brekke, et al., 1999, p. 3–18).

Because of these shortcomings, the CRWMS M&O (2000k) analysis and model report does not provide the technical bases to support the current assessment of the effects of thermal loading, seismic loading, or time-dependent degradation of rock on the behavior of underground openings at Yucca Mountain. Further, the current assessment of drift stability is not consistent with the current state of knowledge on the behavior of underground openings in fractured rock [i.e., that the majority of the drifts are likely to collapse within a relatively short time (compared to the 10,000-year period of regulatory concern) after the cessation of maintenance]. This interpretation of the current state of knowledge is consistent with the DOE expert panel conclusion on drift stability (Brekke, et al., 1999, p. 3–16) and is supported by recent analyses of the behavior of unsupported drifts in fractured rock during seismic loading from an earthquake (Hsiung and Shi, 2001).

There are also concerns with the seismic and fracture data used for the drift degradation analysis. The seismic data used for the drift degradation analysis were the design basis seismic ground motions for both Categories 1 and 2 events. These seismic ground motion parameters are appropriate for preclosure-related design and analysis but are not proper for any postclosure considerations. DOE agreed[20] to address this concern in Seismic Topical Report #3. Development of the fracture data is documented in the fracture geometry analysis and model report (CRWMS M&O, 2000n), which, as previously noted,[21] contains the following implicit or explicit assumptions requiring technical justification:

- Volume sample from full periphery maps eliminates directional bias in the fracture distributions
- Fractures in the Exploratory Studies Facility and cross drift are representative of fracturing throughout the proposed emplacement volume at Yucca Mountain
- Lithology is the sole influence on fracture set characteristics
- Consideration of only fractures more than 1 m [3.3 ft] in length is representative or perhaps conservative with respect to rockfall and drift collapse
- Orientation variation within fracture sets is not important to drift stability
- Curvilinear trace length measured along the tunnel walls is representative of fracture size
- Strike and dip direction of shallowly dipping (<30 degrees) fractures is not important to drift stability
- The number of samples analyzed gives statistically significant results

To address the NRC concerns related to the occurrence of rockfall and drift collapse, as outlined in this section, DOE agreed[22] to

- Provide revised drift degradation analyses using an appropriate range of mechanical and strength properties for rock joints and account for their long-term degradation
- Provide an analysis of block sizes based on the full distribution of joint trace length data from the fracture geometry analysis and model report (CRWMS M&O, 2000n), including small joints trace lengths

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[20]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Repository Design and Thermal-Mechanical Effects (February 6–8, 2001).” Letter (February 28) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[21]Ferrill D., W. Dunne, S. Hsiung, and A. Morris. Review of Analysis and Model Report entitled “Fracture Geometry Analysis for the Stratigraphic Units of the Repository Host Horizon.” Letter Report to NRC (December 27). San Antonio, TX: CNWRA. 2000.

[22]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Repository Design and Thermal-Mechanical Effects (February 6–8, 2001).” Letter (February 28) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

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- Verify the results of the revised drift degradation analyses using (i) appropriate boundary conditions for thermal and seismic loading, (ii) critical fracture patterns from the fracture-network simulations used for the drift degradation analyses (at least two patterns for each rock unit), (iii) consistent thermal and mechanical properties for rock blocks and joints, (iv) long-term degradation of rock block and joint strength parameters, and (v) site-specific ground motion time histories appropriate for the postclosure period

- Provide the technical basis for the effective maximum rock size, including consideration of the effect of variation of the joint dip angle, to be used in assessing the response of the drip shield to rock block impacts

- Provide a detailed documentation of the analysis results

- Evaluate the uncertainties related to the rockfall and drift-collapse analyses and the importance of the outcome of the analyses to the performance of the repository

Staff reviewed DOE documentation of the fracture geometry parameters relevant to rockfall analyses of the repository host horizon rock units (CRWMS M&O, 2000n). Results of this review were documented in an NRC letter dated August 3, 2001,[23] and are summarized as follows.

- Directional Bias: Provide a technical basis for the conclusion that fracture geometry parameter values for the repository host horizon are correct; provide a set of data corrected for these sampling biases, along with a description of the methodology used for sampling bias correction; or risk inform the results to demonstrate that bias does not impact performance of the repository.

- Representativeness of Fracture Parameters: Provide a technical basis or rationale to support the extrapolation of fracture parameters to the repository footprint area. This extrapolation needs to account for heterogeneities in the repository host horizon and uncertainties in the fracture characteristics and their distribution. This technical basis is required to support the models and calculations used to select the new emplacement drift alignment and for the key-block analyses. Similarly, adequate technical rationales should be developed to support the use of the active fracture model and calculations that import or abstract fracture spacing data from the repository host horizon fracture analysis and model report (CRWMS M&O, 2000n).

- Misrepresentation of Aggregated Fracture Characteristics: Provide an adequate technical basis and rationale for the selection of fracture sets (i.e., sets based on orientation and lithology, rather than on origin) and provide statistics that represent the parameter distributions within each fracture set, or risk inform the aggregated characteristics.

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[23]Reamer, C.W. "Structural Deformation and Seismicity Key Technical Issue Agreements: Additional Information Needed." Letter (August 3, 2001) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

- Fractures More Than One Meter [3.3 ft] in Length: Provide an adequate technical basis for the fracture-length database used in rockfall analyses and other calculations, especially for the one-meter [3.3 ft] truncation. This technical basis should be adequate to support DOE key-block analyses for the Topopah Spring Tuff crystal-poor lower lithophysal unit. Alternatively, DOE could risk inform the fracture-length database.

- Orientation Variation Within Fracture Sets: Describe the procedure for defining fracture sets, explain the use of single-values to represent mean fracture set orientations, provide statistics that represent the range or variation in mean fracture orientations distribution of within each fracture set, or risk inform the fracture-orientation variation database.

- Fracture Trace Length and Fracture Shape: Provide an adequate technical basis for the method used to measure fracture lengths in tunnels and drifts and the potential fracture shapes and the significance, if any, to performance. Alternatively provide a risk-informed analysis of fracture trace length and fracture shape data and assumptions.

- Strikes of Shallowly Dipping Fractures: Provide a technically defensible distribution of fracture orientations and related population statistics for subhorizontal fractures used or assumed for tunnel stability analysis or risk inform the current uses or assumptions.

- Statistical Significance of Fracture Populations in the Exploratory Studies Facility and Enhanced Characterization of the Repository Block: Provide a population statistical analysis, unit by unit and set by set, of the fracture data and results and provide the character statistics, or risk inform the current assumptions.

Alternatively, DOE could explain the currently unsupported assumptions using a risk-informed approach. For example, with the absence of complete and persuasive evidence supporting the DOE assumptions of a uniform distribution of fracture characteristics throughout the repository, DOE could develop viable fracture models and use those models to develop a range of representative fracture characteristics most important to repository performance.

Effect of Rockfall and Drift Collapse

Finite Element Modeling Methodology: The process-level models used to approximate the response of the drip shield and waste package to various disruptive events are based on the finite element method. The finite element method is ideally suited to perform these analyses because it can readily account for the combined effects of nonlinear material behavior, nonlinear boundary conditions, and nonlinear geometry (i.e., large strains and large displacements). An important aspect of constructing finite element models, however, is the level of mesh discretization needed to achieve the requisite resolution of the results. To date, DOE has not provided any studies that demonstrate the finite element models used to simulate the functionality of the waste package and drip shield are sufficient to capture highly localized phenomena. For example, complex deformations of the waste package outer barrier in the immediate region of the waste package pallet support are expected. As a result, the finite element discretization will have to be sufficiently refined to capture adequately the localized stress states. Reasonable approximations of the stress are needed to assess the susceptibility of the various engineered barrier subsystem components to stress corrosion cracking.

Drip Shield: The finite element analysis models used by DOE to assess the structural integrity of the drip shield when subjected to rock block impacts (CRWMS M&O, 2000l) do not employ (i) appropriate boundary conditions, (ii) material properties corresponding to the expected emplacement drift environment and the effects of various material degradation processes, or (iii) acceptable criteria for assessing material failure and susceptibility to stress corrosion cracking.

Even though the drip shield is intended to be a free-standing structure, the DOE finite element model uses fixed displacement boundary conditions at its base. In addition, the finite element model did not account for (i) the potential interaction between the drip shield and gantry rails, (ii) the effect of the invert floor moving vertically upward as a result of the seismic excitation that may occur concurrently with rockfall, or (iii) the degradation of the carbon steel structural framework of the invert. These boundary conditions have a significant influence on the overall structural behavior of the drip shield when subjected to rock block impacts. As a result, the location and magnitude of the maximum stresses experienced by the drip shield when subjected to rockfall have not been adequately determined. DOE also assumed in these models that the contact area between the impacting rock block and drip shield will encompass at least 3 m [9.9-ft] length of the drip shield. Distributing the impact load over a relatively large surface area of the drip shield significantly reduces the magnitude of stress that would be experienced by the drip shield if the initial contact area was consistent with localized, point-type impacts.

DOE indicated the drip shield will be fabricated using Titanium Grades 7 and 24. The constitutive relationships used for these two materials within the finite element models simulating the drip shield and rock block impacts were derived from empirical data obtained at room temperature {i.e., approximately 20 °C [68 °F]}. The mechanical material properties for Titanium Grade 7 (American Society of Mechanical Engineers, 1995, 2001), however, are strongly dependent on temperature. The temperature-dependent values for the yield stress, ultimate tensile strength, and Young's modulus of Titanium Grades 5 or 24 are not provided in the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. Note that the compositions of Titanium Grades 5 and 24 are the same except Grade 24 contains 0.04–0.08-percent palladium. As a result, it is expected these two grades will exhibit similar mechanical behavior (i.e., mechanical properties). The U.S. Department of Defense (1998) and ASM International (1994) provide extensive material data for Titanium Grade 5. The Titanium Grade 5 values for the yield stress, ultimate tensile strength, and Young's modulus extracted from graphical data provided in U.S. Department of Defense (1998) are also strongly dependent on temperature. Even though Titanium Grade 5 exhibits much higher strengths than Titanium Grade 7, the relative effects of temperature are still significant and must be considered when assessing the ability of the drip shield to withstand rock block impacts.

In addition to temperature effects, DOE has not adequately addressed the influence of (i) welding flaws and defects, (ii) hydrogen entry into metal, and (iii) fluoride on the corrosion rate of titanium when assessing the ability of the drip shield to perform its intended functions after rockfall and seismic events. Enhanced susceptibility of the titanium drip shield to cracking may occur through hydrogen generated from the galvanic coupling of titanium with degraded carbon steel ground support materials such as rock bolts, steel mesh, or steel sets (CRWMS M&O, 2000o), or the gantry rail. The subsequent uptake of hydrogen into the titanium drip shield materials may reduce the ductility of the titanium drip shield. In addition, corrosion rates of titanium alloys are strongly dependent on fluoride concentration. Groundwater compositions in the emplacement drifts may have elevated fluoride concentrations as a result of evaporation (CRWMS M&O, 2000p). Elevated fluoride concentrations can result in accelerated corrosion of the titanium drip shield and increased hydrogen uptake that, in turn, may increase the susceptibility of the titanium drip shield to either mechanical failure or hydrogen-induced cracking.

No discussion was provided in the CRWMS M&O (2000l) report detailing which components or types of strain measure were used to conclude that "… no crack develops in the drip shield due to the dynamic impact of a rock on the drip shield for any of the rock sizes … ." For generalized three-dimensional stress states, failure criteria for metals are typically based on maximum shear stress, octahedral shear stress, Tresca stress, Von Mises stress, or strain-energy density. These measures are used because they can be readily employed to discern failure when complex stress states exist using data derived from simple tension tests.

The finite element analysis results obtained from the drip shield and rock block impact simulations were also used to assess the potential for the initiation of stress corrosion cracking in the drip shield. The results indicated that the drip shield residual stresses developed as a consequence of the rock block impact may be sufficient to cause stress corrosion cracking. No discussion was provided in the report detailing which components or types of stress were used in making this assessment. For example, no information was provided that addresses the recommended procedure for how generalized three-dimensional stress states obtained from engineering analyses should be interpreted to determine whether the initiation stress threshold for stress corrosion cracking has been exceeded. In addition, given the significant reduction in yield stress for Titanium Grades 7 and 24 at emplacement drift temperatures relative to the corresponding values at room temperature, the assumed initial stress threshold for the stress corrosion cracking criterion does not appear to be conservative.

The potential effects of dead loads on the drip shield caused by rockfall and drift collapse have not been adequately considered by DOE when assessing the performance capabilities of the drip shield. These effects include, but may not be limited to, changes to the dynamic response of the drip shield when subjected to seismic excitation, buckling, and creep.

It can be reasonably assumed that the effective mass of the drip shield will increase without appreciably changing its structural stiffness when supporting dead loads. The natural frequencies of the drip shield, therefore, will be reduced. Reduction in the drip shield natural frequencies is a concern because earthquake loads typically resonate structures with natural frequencies below 33 Hz. As a consequence, the drip shield may respond to seismic excitation by oscillating with displacements large enough to cause repeated impacts with a waste package, resulting in damage presently not accounted for.

Under static conditions, dead loads may also cause the drip shield to buckle or experience large plastic deformations, potentially transferring the dead loads from the drip shield directly to a waste package.

Because the reductions in yield stress and ultimate tensile strength for Titanium Grades 7 and 24 resulting from elevated emplacement drift temperatures are significant, there is some concern by the staff that these materials will also be susceptible to creep-related failures arising from the support of dead loads (e.g., fallen rock blocks or drift collapse). This concern is further substantiated by information provided in a U.S. Department of Defense handbook which states

> Below about 149 °C [300 °F], as well as above about 371 °C [700 °F], creep deformation of titanium alloys can be expected at stresses below the yield strength. Available data indicate that room-temperature creep of unalloyed titanium may be significant (exceed 0.2-percent creep-strain in 1,000 hours) at stresses that exceed approximately 50 percent F_{ty} [tensile yield stress], … (U.S. Department of Defense, 1998, p. 5-2).

Moreover,

> The alpha-beta alloys [Titanium Grade 24] have good strength at room temperature and for short times at elevated temperature. They are not noted for long-time creep strength. (U.S. Department of Defense, 1998, p. 5-51).

Room-temperature creep has been investigated for a variety of alpha or near-alpha (hexagonal closed packed) and alpha-beta (hexagonal closed packed-body centered cubic) titanium alloys. Significant room-temperature creep can occur in alpha or near-alpha titanium alloys, whereas, alpha-beta titanium alloys are not as susceptible to this degradation mechanism. Chu (1970) reported considerable creep strains for a near-alpha T1-6Al-2Cb-1Ta-0.8 Mo alloy at room temperature when the applied stress was above 80 percent of the yield strength. In contrast, the creep strains observed for alpha-beta Ti-6Al-4V at 90 percent of the yield strength are low (Odegard and Thompson, 1974) but dependent on the microstructure of the alloy (Imam and Gilmore, 1979). Tests conducted on as-welded Ti-6Al-4V showed similar behavior to the base alloy with the exception of a decrease in the yield strength for the as-welded material (Odegard and Thompson, 1974).

DOE has neither referenced specific creep data for Titanium Grades 7 and 24 nor provided adequate analyses demonstrating that dead loads caused by fallen rock blocks and drift collapse will not occur. Creeping of the drip shields subjected to dead loads can reduce the clearance between the drip shield bulkhead and the waste package. Given time, the dead loads may ultimately be supported by the waste package directly, or during a seismic event, the clearance may have been sufficiently reduced to the point that the drip shield will repeatedly impact the waste package, resulting in damage presently not accounted for.

DOE proposed an evaluation of the drip shield static loading (CRWMS M&O, 2000q) using a procedure based on Rankine’s theory of earth pressure (e.g., Terzaghi, et al., 1996). The proposed approach, however, is inappropriate because it does not account for the dead weight of fallen rock that may rest directly on the drip shield, and it does not adequately represent the lateral loads arising from naturally occurring or engineered backfill.

To address NRC concerns related to the effect of rockfall and drift collapse on the drip shield, as outlined in this section, DOE agreed[24,25,26] to

- Perform drip shield seismic evaluations that include the effects of static loads from fallen rock

- Perform drip shield rockfall evaluations that include the effects of (i) wall thinning caused by corrosion, (ii) hydrogen embrittlement, and (iii) multiple rock blocks falling simultaneously

- Provide (i) the justification for not including the rockfall effect and drift collapse loads on stress corrosion cracking of the drip shield and (ii) the documentation for the point loading rockfall analyses

- Demonstrate how the Tresca Failure criterion bounds a fracture mechanics approach to calculating the mechanical failure of the drip shield. Provide a technical basis for a stress measure that can be used as the equivalent uniaxial stress for assessing the susceptibility of titanium to stress corrosion cracking. The proposed equivalent uniaxial stress measure must be consistent and compatible with the methods proposed by DOE to assess stress corrosion cracking of the containers in WAPDEG. A detailed discussion of how the equivalent uniaxial stress measure will be used to determine nucleation of stress corrosion cracks in the calculations performed to evaluate the stress corrosion cracking criterion for the drip shield should be included

- Clarify why the effects of seismicity and large block rockfall are not considered in the Total System Performance Assessment Code (features, events, and processes numbers 1.2.03.02.00 and 2.1.07.01.00) [when providing this clarification, DOE should include analyses of the drip shield subjected to rock block impacts and seismic loads using boundary conditions that (i) represent the drip shield as a free-standing structure, (ii) account for the potential interactions between the drip shield and gantry rails (and any other relevant structures, systems, or components), and (iii) include the effects of seismic ground motion at the invert floor and take into account welding flaws and defects and the reduced mechanical strength of titanium commensurate with anticipated temperatures]

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[24]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000)." Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[25]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Repository Design and Thermal-Mechanical Effects (February 6–8, 2001)." Letter (February 28) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[26]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration—Features, Events, and Processes (May 15–17, 2001)." Letter (May 30) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

- Provide technical basis for the screening argument pertaining to creeping of metallic materials in the engineered barrier subsystem (features, events, and processes number 2.1.07.05.00)

Waste Package: The finite element analysis models used by DOE to assess the structural integrity of the waste package when subjected to rock block impacts (CRWMS M&O, 1999) do not employ (i) boundary conditions between the inner and outer barriers of the waste package consistent with the current waste package design, (ii) material properties corresponding to the expected emplacement drift environment and the effects of various material degradation processes, or (iii) acceptable criteria for assessing material failure and susceptibility to stress corrosion cracking.

Furthermore, DOE has not performed an assessment of the stresses generated in the waste package outer barrier near the pallet support caused by rock block impacts and seismic excitation. Specific aspects of the new waste package design and analyses of concern to the NRC staff are (i) the assumption that the inner and outer barriers can be treated as a single composite component in the DOE finite element models, (ii) the potential loss of material ductility in the immediate area of the closure lid welds, (iii) the design provisions that do not properly account for the difference in thermal expansion between the inner and outer barriers of the waste package, and (iv) the failure criteria used to assess the structural integrity of the waste package.

DOE has not adequately addressed the effects of welding flaws and defects and waste package degradation processes such as uniform corrosion, localized corrosion, stress corrosion cracking, and the possible decreased ductility as a result of container fabrication or long-term thermal aging that may reduce the ability of the waste package to withstand rockfall or seismic events. Penetration of the waste package outer barrier by localized corrosion or stress corrosion cracking will result in the exposure and subsequent degradation of the inner stainless steel container. In addition, the effects of container fabrication, thermal aging, or an increase in the exposure temperature as a result of volcanic activity may result in the formation of brittle phases that reduce the ductility of the waste package materials.

To address the NRC concerns related to the effects of rockfall and drift collapse on the waste package, as outlined in this section, DOE agreed[27,28,29,30] to

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[27]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000)." Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[28]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Repository Design and Thermal-Mechanical Effects (February 6–8, 2001)." Letter (February 28) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[29]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration–Features, Events, and Processes (May 15–17, 2001)." Letter (May 30) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[30]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

- Perform waste package rockfall evaluations that include the effects of (i) potential waste package closure weld material embrittlement after stress annealing and (ii) multiple rock blocks falling simultaneously

- Provide the documentation for the waste package point loading rockfall analyses

- Demonstrate how the Tresca Failure criterion bounds a fracture mechanics approach to calculating the mechanical failure of the waste package. Provide a technical basis for a stress measure that can be used as the equivalent uniaxial stress for assessing the susceptibility of Alloy 22 to stress corrosion cracking. The proposed stress measure must be consistent and compatible with the methods proposed by DOE to assess stress corrosion cracking of the containers in WAPDEG. A detailed discussion of how the stress measure will be used to determine nucleation of stress corrosion cracks in the calculations performed to evaluate the stress corrosion cracking criterion for the waste package should be included).

- Clarify why the effects of seismicity and large block rockfall are not considered in the Total System Performance Assessment Code (features, events, and processes numbers 1.2.03.02.00 and 2.1.07.01.00) [when providing this clarification, DOE should include analyses of the waste package that consider the effects of (i) temperature-dependent material properties, (ii) uniform and localized corrosion, (iii) welding flaws and defects, (iv) differential thermal expansion effects, and (v) susceptibility of the outer barrier to stress corrosion cracking where potential interactions with the drip shield may have occurred and in the immediate contact region with the pallet support]

- Clarify the description of the primary features, events, and processes (number 1.2.03.02.00, seismic vibration causes container failure)

- Provide the technical basis for the screening argument pertaining to the differing thermal expansion of repository components (features, events, and processes number 2.1.11.05.00)

3.3.2.4.4.2 Data Are Sufficient for Model Justification

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.2.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess mechanical disruption of engineered barriers (i.e., rockfall and drift collapse) with respect to sufficient data for model justification.

The fracture contact stiffness and strength properties used to support the drift degradation analysis (CRWMS M&O, 2000k) are not sufficient. These properties were determined based on 12 laboratory shear tests of fractures from the Topopah Spring densely welded devitrified lithophysal-poor Tuff. No distinction was made on the fracture properties among the three subunits of the Topopah Spring densely welded devitrified lithophysal-poor Tuff thermal-mechanical unit (CRWMS M&O, 2000k,r). Furthermore, the fracture shear stiffness (an important parameter for the verification studies) is not available and was assumed in the analysis (CRWMS M&O, 2000r). DOE agreed[31] to address these concerns.

See Sections 2.1.7.3, 3.3.1.4.1.2, and 3.3.1.4.2.2 of this report for comments related to data being sufficiently characterized and propagated for model justification for this topic area.

3.3.2.4.4.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

At the time this report was prepared, the effects of rockfall and drift collapse were excluded from the total system performance assessment model abstraction for disruptive events. DOE agreed to address the NRC concerns, as discussed in Sections 3.3.2.4.4.1, 3.3.2.4.4.2, and 3.3.2.4.4.3. Depending on the resolution of these concerns, the effects of rockfall and drift collapse will be included or excluded from the total system performance assessment model abstraction for disruptive events.

3.3.2.4.4.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

At the time this report was prepared, the effects of rockfall and drift collapse were excluded from the total system performance assessment model abstraction for disruptive events. DOE agreed to address the NRC concerns, as discussed in Sections 3.3.2.4.4.1, 3.3.2.4.4.2, and 3.3.2.4.4.3. Depending on the resolution of these concerns, the effects of rockfall and drift collapse will be included or excluded from the total system performance assessment model abstraction for disruptive events.

3.3.2.4.4.5 Verification of Model Abstraction

At the time this report was prepared, the effects of rockfall and drift collapse were excluded from the total system performance assessment model abstraction for disruptive events. DOE agreed to address the NRC concerns, as discussed in Sections 3.3.2.4.4.1, 3.3.2.4.4.2, and 3.3.2.4.2.3. Depending on the resolution of these concerns, the effects of rockfall and drift collapse will be included or excluded from the total system performance assessment model abstraction for disruptive events.

3.3.2.4.5 Criticality

3.3.2.4.5.1 System Description and Model Integration Are Adequate

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.2.5), is sufficient to conclude that the necessary information will be available at the

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[31]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Repository Design and Thermal-Mechanical Effects (February 6–8, 2001)." Letter (February 28) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

time of a potential license application to assess mechanical disruption of engineered barriers (i.e., criticality) with respect to system description and model integration.

DOE screened the occurrence of nuclear criticality for commercial spent nuclear fuel from consideration in the Total System Performance Assessment Code based on no waste package breach or failure and a low probability of critical configuration formation at any time during the postclosure period (CRWMS M&O, 2000s,t). DOE recently indicated (Bechtel SAIC Company, LLC, 2001a,b), however, there would be waste package failures prior to 10,000 years caused by improper heat treatment during fabrication. In addition, DOE has yet to demonstrate adequately the waste packages can satisfactorily maintain confinement from either direct or indirect effects that can be attributed to mechanically disruptive events or various corrosion processes (see Section 3.3.1). As a result, DOE agreed[32] to reexamine the screening arguments for postclosure criticality.

For criticality induced by seismic loading, the methodology for estimating the probability of a criticality event (DOE, 2000) will first identify and evaluate the waste package configurations that could become critical or supercritical as a result of being subjected to seismic loads. These configurations are called seismic predecessor configurations. To determine the probability of a criticality event initiated by seismic loads, the probability of any given seismic predecessor configuration will be multiplied by the probability of a seismic event that has a magnitude capable of taking such a configuration to criticality.

The methodology for estimating the probability of an igneous-induced criticality begins by identifying the potential critical configurations that can be created by an igneous event. The criticality potentials of these configurations are then evaluated according to the process described in the topical report.

DOE used the methodology for estimating the probability of criticality induced by an igneous event for waste packages containing pressurized water reactor spent nuclear fuel (CRWMS M&O, 2000t). To obtain this probability estimate, DOE evaluated the criticality configuration potential pertaining to the complete destruction of the seven waste packages located in Zone 1 (CRWMS M&O, 2000e). The result indicated that the system would be subcritical for the range of pellet spacings and fuel and magma volumes considered in the analyses. The analysis did not include any other waste package types containing high-enriched fuel (e.g., U.S. Navy and DOE-owned spent nuclear fuel). As for the Zone 2-type damages (CRWMS M&O, 2000e) (i.e., partial damage of the remaining waste packages in any drift intersected by an igneous intrusion), DOE calculated the probability for criticality to be 1.8×10^{-7} over 10,000 years, which is smaller than 1×10^{-4} over 10,000 years (screening criteria per 10 CFR 63.113). Similar to the Zone 1 analysis, DOE only evaluated waste packages containing pressurized water reactor spent nuclear fuel. Staff do not believe DOE can screen out igneous-induced criticality by evaluating only one waste package and fuel type. Therefore, the approach should include the probability and configurations for all potential waste

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[32]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Range of Operating Temperatures (September 18–19, 2001).” Letter (October 2) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

package and fuel types. In the Range of Thermal Operating Modes Technical Exchange, DOE agreed[33] to update the probability estimates for criticality by performing analyses that include different waste package and fuel types.

Because of the large uncertainty associated with calculating criticality probabilities, DOE also agreed[34] to perform a what-if criticality consequence analysis using a revised methodology (DOE, 2000), which is presently being reviewed by NRC, to determine the potential effects of criticality on meeting repository performance requirements.

3.3.2.4.5.2 Data Are Sufficient for Model Justification

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.2.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess mechanical disruption of engineered barriers (i.e., criticality) with respect to sufficient data for model justification.

DOE indicated relevant data pertaining to seismicity, faulting, volcanism, and rockfall used in criticality models will be consistent with data used in other areas of the total system performance assessment, where appropriate (DOE, 2000). Other significant data will be contained in the validation reports for the inventory, neutronics, and geochemistry computer codes that will be used in the criticality modeling. DOE agreed[35] to provide these validation reports to NRC prior to submission of any license application for the Yucca Mountain repository.

3.3.2.4.5.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.2.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess mechanical disruption of engineered barriers (i.e. criticality) with respect to the characterization and propagation of data uncertainty through the model abstraction.

DOE indicated that uncertainty distributions of parameters associated with seismicity, faulting, volcanism, and rockfall used in criticality models will be consistent with other areas of the total system performance assessment where appropriate (DOE, 2000). The validation reports for the inventory, neutronics, and geochemistry computer codes will quantify the effect of data

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[33]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Range of Operating Temperatures (September 18–19, 2001)." Letter (October 2) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[34]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Criticality (October 23–24, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[35]Ibid.

uncertainty on the results of these computer codes. DOE agreed[36] to provide these validation reports to NRC prior to submission of any license application for the Yucca Mountain repository.

3.3.2.4.5.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

At the time this report was prepared, the effects of criticality were excluded from the total system performance assessment model abstraction for disruptive events. DOE agreed to address the NRC concerns, as discussed in Sections 3.3.2.4.5.1, 3.3.2.4.5.2, and 3.3.2.4.5.3. Depending on the resolution of these concerns, the effects of criticality will be included or excluded from the total system performance assessment model abstraction for disruptive events.

3.3.2.4.5.5 Model Abstraction Output Is Supported by Objective Comparisons

At the time this report was prepared, the effects of criticality were excluded from the total system performance assessment model abstraction for disruptive events. DOE agreed to address the NRC concerns, as discussed in Sections 3.3.2.4.5.1, 3.3.2.4.5.2, and 3.3.2.4.5.3. Depending on the resolution of these concerns, the effects of criticality will be included or excluded from the total system performance assessment model abstraction for disruptive events.

3.3.2.5 Status and Path Forward

Table 3.3.2-2 provides the status of all key technical issue subissues, referenced in Section 3.3.2.2, for the Mechanical Disruption of Engineered Barriers Integrated Subissue. The table also provides the related DOE and NRC agreements pertaining to the Mechanical Disruption of Engineered Barriers Integrated Subissue. The agreements listed in the table are associated with one or all five generic acceptance criteria discussed in Section 3.3.2.4. Note that the status and the detailed agreements (or path forward) pertaining to all the key technical issue subissues are provided in Table 1.1-3 and Appendix A.

[36]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Criticality (October 23–24, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

Table 3.3.2-2. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreements*
Container Life and Source Term	Subissue 1—Effects of Corrosion Processes on the Lifetime of the Containers	Closed-Pending	CLST.1.13 CLST.1.14 CLST.1.16 CLST.1.17
	Subissue 2—Effects of Phase Instability of Materials and Initial Defects on the Mechanical Failure and Lifetime of the Containers	Closed-Pending	CLST.2.01 through CLST.2.09
	Subissue 5—Effect of In-Package Criticality on Waste Package and Engineered Barrier Subsystem Performance	Closed-Pending	CLST.5.01 CLST.5.03 CLST.5.06 CLST.5.07
	Subissue 6—Effect of Alternate of Engineered Barrier Subsystem Design Features on Container Lifetime and Radionuclide Release from the Engineered Barrier Subsystem	Closed-Pending	None
Igneous Activity	Subissue 1—Probability of Igneous Activity	Closed-Pending	None
	Subissue 2—Consequences of Igneous Activity	Closed-Pending	IA.2.10 IA.2.18 IA.2.19 IA.2.20
Repository Design and Thermal-Mechanical Effects	Subissue 1—Design Control Process	Closed	None
	Subissue 2—Seismic Design Methodology	Closed-Pending	RDTME.2.01 RDTME.2.02
	Subissue 3—Thermal-Mechanical Effects	Closed-Pending	RDTME.3.03 RDTME.3.15 To RDTME.3.19
Structural Deformation and Seismicity	Subissue 1—Faulting	Closed-Pending	SDS.1.02
	Subissue 2—Seismicity	Closed-Pending	SDS.2.01 SDS.2.03 SDS.2.04
	Subissue 3—Fracturing and Structural Framework of the Geologic Setting	Closed-Pending	SDS.3.04

Table 3.3.2-2. Related Key Technical Issue Subissues and Agreements (continued)			
Key Technical Issue	**Subissue**	**Status**	**Related Agreements***
Structural Deformation and Seismicity	Subissue 4—Tectonic Framework of the Geologic Setting	Closed	None
Total System Performance Assessment and Integration	Subissue 1—System Description and Demonstration of Multiple Barriers	Closed-Pending	None
	Subissue 2—Scenario Analysis and Event Probability	Closed-Pending	TSPAI.2.02 TSPAI.2.04
	Subissue 3—Model Abstraction	Closed-Pending	TSPAI.3.06
	Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards	Closed-Pending	None

*Related DOE and NRC agreements are associated with one or all five generic acceptance criteria.

Note: Key Technical Issue Agreement GEN.1.01 pertains to multiple integrated subissues, as well as some specific issues related to this integrated subissue.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (through specific data collection, testing, and analyses), acceptably addresses the NRC questions so that no information beyond that provided, or agreed to, will likely be required at the time of a potential license application.

3.3.2.6 References

American Society of Mechanical Engineers. *ASME Boiler and Pressure Vessel Code*. New York City, New York: American Society of Mechanical Engineers. 1995.

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

Anderson, J.G., S.G. Wesnousky, and M.W. Stirling. “Earthquake Size As a Function of Fault Slip Rate.” *Bulletin of the Seismological Society of America.* Vol. 86. pp. 683–690. 1996.

Anderson, R.E., R.C. Buckman, A.J. Crone, K.M. Haller, M.N. Machette, S.F. Personius, T.P. Barnhard, M.J. Cecil, and R.L. Dart. “Characterization of Quaternary and Suspected Quaternary Faults, Regional Studies, Nevada and California.” U.S. Geological Survey Open-File Report 56. pp. 95–599 plus appendixes. 1995a.

Anderson, R.E., A.J. Crone, M.N. Machette, L.A. Bradley, and S.F. Diehl. “Characterization of Quaternary and Suspected Quaternary Faults, Amargosa Area, Nevada and California.” U.S. Geological Survey Open-File Report 41. pp. 95–613 plus appendixes. 1995b.

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

Bechtel SAIC Company, LLC. "FY01 Supplemental Science and Performance Analyses." Vol. 2: Performance Analyses. TDR–MGR–PA–000001. Revision 00. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2001a.

———. "FY01 Supplemental Science and Performance Analyses." Vol. 1: Scientific Bases and Analyses. TDR–MGR–MD–000007. Revision 00 ICN 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2001b.

Brekke, T.L., E.J. Cording, J. Daemen, R.D. Hart, J.A. Hudson, P.K. Kaiser, and S. Pelizza. "Panel Report on the Drift Stability Workshop, Las Vegas, Nevada, 9–11 December 1998." Las Vegas, Nevada: DOE, Yucca Mountain Site Characterization Project. 1999.

Budnitz, R.J., G. Apostolakis, D. M. Boor, L.S. Cluff, K.J. Coppersmith, C.A. Cornell, and P.A. Morris NUREG/CR–6372. "Senior Seismic Hazard Analysis Committee, Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts." Washington, DC: NRC. April 1997.

Chu, H.P. "Room Temperature Creep and Stress Relaxation of a Titanium Alloy." *Journal of Materials JMLSA.* Vol. 5, No. 3. pp. 633–642. 1970.

Cornell, C.A. and E.H. Van Marke. "The Major Influences on Seismic Risk." Proceedings, Third World Conference on Earthquake Engineering. Santiago, Chile: pp. 3–14. 1969.

CRWMS M&O. "Probabilistic Volcanic Hazards Analysis for Yucca Mountain, Nevada." BA0000000–1717–2200–00082. Revision 0. Las Vegas, Nevada: DOE, Yucca Mountain Site Characterization Office. 1996.

———. "Probabilistic Seismic Hazard Analyses for Fault Displacement and Vibratory Ground Motion at Yucca Mountain, Nevada, Final Report." WBS Number 1.2.3.2.8.3.6. Las Vegas, Nevada: CRWMS M&O. 1998.

———. "Rock Fall and Vertical Drop Calculations of Waste Packages." BBA000000–01717–0210–00058. Revision 01. Las Vegas, Nevada: CRWMS M&O. 1999.

———. "Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations." TDR–WIS–RL–000001. Revision 04 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000a.

———. "Total System Performance Assessment–Site Recommendation." TDR–WIS–PA–000001. Revision 00 ICN1. North Las Vegas, Nevada: TRW Environmental Safety Systems, Inc. 2000b.

———. "Features, Events, and Processes: Screening for Disruptive Events." ANL–WIS–MD–000005. Revision 00 ICN 1. Las Vegas, Nevada: CRWMS M&O. 2000c.

———. “Disruptive Events Process Model Report.” TDR–NBS–MD–000002. Revision 00 ICN 1. North Las Vegas, Nevada: DOE, Yucca Mountain Site Characterization Office. 2000d.

———. “Dike Propagation Near Drifts.” ANL–WIS–MD–000015. Revision 00 ICN 1. North Las Vegas, Nevada: DOE, Yucca Mountain Site Characterization Office. 2000e.

———. “Igneous Consequence Modeling for the Total System Performance Assessment–Site Recommendation.” ANL–WIS–MD–000017. Revision 00 ICN 1. Las Vegas, Nevada: DOE, Yucca Mountain Site Characterization Office. 2000f.

———. “Characterize Eruptive Processes at Yucca Mountain, Nevada.” ANL–MGR–GS–000002. Revision 00. Las Vegas, Nevada: DOE, Yucca Mountain Site Characterization Office. 2000g.

———. “Waste Package Behavior in Magma.” CAL–EBS–ME–000002. Revision 00. Las Vegas, Nevada: DOE, Yucca Mountain Site Characterization Office. 2000h.

———. “Number of Waste Packages Hit by Igneous Intrusion.” CAL–WIS–PA–000001. Revision 01. Las Vegas, Nevada: DOE, Yucca Mountain Site Characterization Office. 2000i.

———. “Effects of Fault Displacement on Emplacement Drifts, Analysis and Models Report.” ANL–EBS–GE–000004. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000j.

———. “Drift Degradation Analysis.” ANL-EBS–MD–000027. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000k.

———. “Rock Fall on Drip Shield.” CAL–EDS–ME–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000l.

———. “Engineered Barrier System Degradation, Flow, and Transport Process Model Report.” TDR–EBS–MD–000006. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000m.

———. “Fracture Geometry Analysis for the Stratigraphic Units of the Repository Horizon.” ANL–EBS–GE–000006. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000n.

———. “Longevity of Emplacement Drift Ground Support Materials.” ANL–EBS–GE–000003. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000o.

———. “Environment on the Surfaces of the Drip Shield and Waste Package Outer Barrier.” ANL–EBS–MD–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000p.

———. “Structural Calculations of the Drip Shield Statically Loaded by the Backfill and Loose Rock Mass.” CAL–XCS–ME–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000q.

———. “Ground Control for Emplacement Drifts for Site Recommendation.” ANL–EBS–GE–000002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000r.

———. “Features, Events, and Processes: System-Level and Criticality.” ANL–WIS–MD–000019. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000s.

———. “Probability of Criticality Before 10,000 Years.” CAL–EBS–NU–000014. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000t.

———. “Features, Events, and Processes Screening of Processes and Issues in Drip Shield and Waste Package Degradation.” ANL–EBS–PA–000002. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2001a.

———. “Engineered Barrier Subsystem Features, Events, and Processes/Degradation Modes Abstraction.” ANL–WIS–PA–000002. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2001b.

dePolo, C.M. “The Maximum Background Earthquake in the Basin and Range.” *Bulletin of the Seismological Society of America.* Vol. 84. pp. 466–472. 1994.

dePolo, C.M., D.G. Clark, D.B. Slemmons, and A.R. Ramelli. “Historical Surface Faulting in the Basin and Range Province, Western North America, Implications for Fault Segmentation.” *Journal of Structural Geology*. Vol. 13. pp. 123–126. 1991.

DOE. “Methodology to Assess Fault Displacement and Vibratory Ground Motion Hazards at Yucca Mountain.” Yucca Mountain/TR–002–NP. Revision 0. Las Vegas, Nevada: DOE. 1994.

———. “Disposal Criticality Analysis Methodology Topical Report.” YMP/TR–004Q. Revision 01. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 2000.

Ferrill, D.A., G.L. Stirewalt, D.B. Henderson, J.A. Stamatakos, K.H. Spivey, and B.P. Wernicke. NUREG/CR–6401, “Faulting in the Yucca Mountain Region, Critical Review and Analyses of Tectonic Data from the Central Basin and Range.” Washington, DC: NRC. April 1996.

Gutenberg, B. and C.F. Richter. *Seismicity of the Earth*. Princeton, New Jersey: Princeton University Press. 1954.

Hanks, T.C. and H. Kanamori. “A Moment–Magnitude Scale.” *Journal of Geophysical Research.* Vol. 84. pp. 2,348–2,350. 1979.

Haynes International. Hastelloy Alloy C-22 product literature. 1988.

———. http://www.haynesintl.com/mini/C22s/C22.htm. 2001.

Hill, B.E. and C.B. Connor. “Technical Basis for Resolution of the Igneous Activity Key Technical Issue.” San Antonio, TX: CNWRA. 2000.

Hsiung, S.M. and G.H. Shi. "Simulation of Earthquake Effects on Underground Excavations Using Discontinuous Deformation Analysis (DDA)." *Proceedings 38th U.S. Rock Mechanics Symposium*. Washington, DC: CNWRA. 2001.

Imam, M.A. and C.M. Gilmore. "Room Temperature Creep of Ti-6Al-4V." *Metallurgical Transactions.* Vol. 10A. pp. 419–425. 1979.

McKague, H.L. J.A. Stamatakos, and D.A. Ferrill, "Type I Faults in the Yucca Mountain Region." CNWRA 96-007. Revision 1. San Antonio, Texas: CNWRA. 1996.

NRC. NUREG–1563, "Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program." Washington, DC: NRC, Division of Waste Management. 1996.

———. "Issue Resolution Status Report, Key Technical Issue: Igneous Activity." Revision 2. Washington, DC: NRC, Division of Waste Management. 1999a.

———. "Issue Resolution Status Report, Key Technical Issue: Structural Deformation and Seismicity." Revision 2. Washington, DC: NRC, Division of Waste Management. 1999b.

———. "Issue Resolution Status Report, Key Technical Issue: Repository Design and Thermal-Mechanical Effects." Revision 3. Washington, DC: NRC, Division of Waste Management. 2000a.

———. "Issue Resolution Status Report, Key Technical Issue: Total System Performance Assessment Integration." Revision 3. Washington, DC: NRC, Division of Waste Management. 2000b.

———. "Issue Resolution Status Report, Key Technical Issue: Container Life and Source Term." Revision 3. Washington, DC: NRC, Division of Waste Management. 2001.

———. NUREG–1804, "Yucca Mountain Review Plan—Draft Report for Comment." Revision 2. Washington, DC: NRC. March 2002.

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

Piety, L.A. "Compilation of Known and Suspected Quaternary Faults Within 100 km of Yucca Mountain." U.S. Geological Survey Open-File Report 94-112. Scale 1:250,000. 1995.

Rebak, R.B., T.S.E. Summers, and R.M. Carranza. "Mechanical Properties, Microstructure, and Corrosion Performance of C-22 Alloy Aged at 260 °C to 800 °C." Scientific Basis for Nuclear Waste Management XXIII. R.W. Smith and D.W. Shoesmith, eds. Symposium Proceedings 608. Warrendale, Pennsylvania: Materials Research Society. pp. 109–114. 2000.

Reheis, M.C. and T.H. Dixon. “Kinematics of the Eastern California Shear Zone: Evidence for Slip Transfer from Owens and Saline Valley Fault Zones to Fish Lake Valley Fault Zone.” *Geology.* Vol. 24. pp. 339–342. 1996.

Reheis, M.C. and T.L. Sawyer. “Late Cenozoic History and Slip Rate of the Fish Lake Valley. Emigrant Peak, and Deep Springs Fault Zones, Nevada and California.” *Geological Society of America Bulletin.* Vol. 109. pp. 280–299. 1997.

Rousseau, J.P. E.M. Kwicklis, and D.C. Giles. “Hydrogeology of the Undersaturated Zone, North Ramp Area of the Exploratory Studies Facility, Yucca Mountain, Nevada.” U.S. Geological Survey Water Resources Investigations Report 98-4050. 1999.

Schwartz, D.B. and K.J. Coppersmith. “Fault Behavior and Characteristic Earthquakes: Examples from the Wasatch and San Andreas Faults.” *Journal of Geophysical Research.* Vol. 89. pp. 5,681–5,698. 1984.

Schweickert, R.A. and M.M. Lahren. “Strike-Slip Fault System in Amargosa Valley and Yucca Mountain, Nevada.” *Tectonophysics.* Vol. 272. pp. 25–41. 1997.

Scott, R.B. “Tectonic Setting of the Yucca Mountain Region, Southwest Nevada.” *Geological Society of America Memoir* 176. Boulder, Colorado: Geological Society of America. pp. 251–282. 1990.

Simons, W.F., J.W. Whitney, K. Fox, A. Ramelli, J.C. Yount, M.D. Carr, C.D. Menges, R. Dickerson, and R.B. Scott. “Map of Fault Activity of the Yucca Mountain Area, Nye County, Nevada.” U.S. Geological Survey Miscellaneous Investigations Series, Map I–2520. Scale 1:24,000. 1995.

Stamatakos, J.A., P.S. Justus, D.A. Ferrill, R. Chen, and G.I. Ofoegbu. NUREG/CR–6513, No. 1, CNWRA 96–OIA, “Structural Deformation and Seismicity. NRC High-Level Radioactive Waste Program Annual Progress Report: Fiscal Year 1996.” B. Sagar, ed. Washington, DC: NRC. January 1997.

Stepp, C.J., I. Wong, J. Whiteny, R. Quittmeyer, N. Abrahamson, G. Toro, R. Youngs, K. Coppersmith, J. Savy, T. Sullivan, and Yucca Mountain Probabilistic Seismic Hazard Analysis Project Members. “Probabilistic Seismic Hazard Analyses for Ground Motions and Fault Displacement at Yucca Mountain, Nevada.” *Earthquake Spectra.* Vol. 17. pp. 113–151. 2001.

Su, F., J.G. Anderson, J.N. Brune, and Y. Zeng. “A Comparison of Direct S Wave and Code Wave Site Amplification Determined from Aftershocks of the Little Skull Mountain Earthquake.” *Bulletin of the Seismological Society of America.* Vol. 86. pp. 1,006–1,018. 1996.

Summers, T.S.E., M.A. Wall, M. Kumar, S.J. Matthews, and R.B. Rebak. "Phase Stability and Mechanical Properties of C-22 Alloy Aged in the Temperature Range 590 to 760 °C for 16,000 Hours." Scientific Basis for Nuclear Waste Management XXII. D. Wronkiewics and J. Lee eds. Symposium Proceedings 556. Warrendale, Pennsylvania: Materials Research Society. pp. 919–926. 1999.

Terzaghi, K., R.B. Peck, and G. Mesri. *Soil Mechanics in Engineering Practice.* 3rd Edition. New York City, New York: John Wiley and Sons, Inc. 1996.

U.S. Department of Defense. "Military Handbook: Metallic Materials and Elements for Aerospace Vehicle Structures." MIL–DBK–5H. Washington, DC: U.S. Department of Defense. 1998.

U.S. Geological Survey. "Seismotectonic Framework and Characterization of Faulting at Yucca Mountain, Nevada." J.W. Whitney, coord. Denver, Colorado: U.S. Geological Survey. 1996.

Veneziano, D. and J. van Dyck. "Statistical Discrimination of 'Aftershocks' and Their Contribution to Seismic Hazard. Seismic Hazard Methodology for Nuclear Facilities in the Eastern United States." EPRI Research Project No. P1010–29, Appendix A–4. pp. A121–A186. 1985.

Wells, D.L. and K.J. Coppersmith. "Likelihood of Surface Rupture as a Function of Magnitude." *Seismological Research Letters.* Vol. 64. p. 54. 1993.

———. "New Empirical Relationships Among Magnitude, Rupture Length, Rupture Width, Rupture Area, and Surface Displacement." *Bulletin of the Seismological Society of America.* Vol. 84. pp. 974–1,002. 1994.

Wernicke, B.P. "Low-Angle Normal Faults and Seismicity, A Review." *Journal of Geophysical Research*. Vol. 100, No. 20. pp. 159–20,174. 1995.

Wesnousky, S.G. "Earthquakes, Quaternary Faults, and Seismic Hazards in California." *Journal of Geophysical Research.* Vol. 91, No. B12. pp. 12,587–12,631. 1986.

Youngs, R.R., F.H. Swan, III, M.S. Power, D.P. Schwartz, and R.K. Green. "Probabilistic Analysis of Earthquake Ground-Shaking Along the Wasatch Front, Utah." W.W. Hays and P.L. Gori, eds. *Assessment of Regional Earthquake Hazard Drip Shield and Risk Along the Wasatch Front, Utah*. U.S. Geological Survey Open-File Report 87–585. Vol. II. pp. M–1 through M–100. 1987.

Zhang, P., M. Ellis, D.B. Slemmons, and F. Mao. "Right-Lateral Displacement and Holocene Slip Rate Associated with Prehistoric Earthquakes Along the Southern Panamint Valley Fault Zone: Implications for Southern Basin and Range Tectonics and Coastal California Deformation." *Journal of Geophysical Research*. Vol. 95. pp. 4,857–4,872. 1990.

3.3.3 Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms

3.3.3.1 Description of Issue

The Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms Integrated Subissue addresses features, events, and processes in the engineered barrier subsystem that may alter the chemical composition or volume of water present on the drip shield and waste package surfaces. To facilitate issue resolution, hydrologic processes affecting seepage rates are treated in the Flow Paths in the Unsaturated Zone Integrated Subissue, and quantity and chemistry of water inside breached waste packages are addressed by the Radionuclide Release Rates and Solubility Limits Integrated Subissue. Relationship of this integrated subissue to other subissues is depicted in Figure 3.3.3-1. The figure shows the relationship between this integrated subissue and the flow paths in the unsaturated zone (Section 3.3.6), mechanical disruption of engineered barriers (Section 3.3.2), radionuclide transport in the unsaturated zone (Section 3.3.7), degradation of engineered barriers (Section 3.3.1), and radionuclide release and solubility limits (Section 3.3.4) subissues. The overall organization and identification of all the integrated subissues are depicted in Figure 1.2-2.

3.3.3.2 Relationship to Key Technical Issue Subissues

The Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms Integrated Subissue incorporates subject matter previously captured in the following key technical issue subissues:

- Evolution of the Near-Field Environment: Subissue 1—Effects of Coupled Thermal-Hydrological-Chemical Processes on Seepage and Flow (NRC, 2000a)

- Evolution of the Near-Field Environment: Subissue 2—Effects of Coupled Thermal-Hydrological-Chemical Processes on the Waste Package Chemical Environment (NRC, 2000a)

- Evolution of the Near-Field Environment: Subissue 3—Effects of Coupled Thermal-Hydrological-Chemical Processes on the Chemical Environment for Radionuclide Release (NRC, 2000a)

- Evolution of the Near-Field Environment: Subissue 4—Effects of Thermal-Hydrological-Chemical Processes on Radionuclide Transport through Engineered and Natural Barriers (NRC, 2000a)

- Evolution of the Near-Field Environment: Subissue 5—Effects of Coupled Thermal-Hydrological-Chemical Processes on Potential Nuclear Criticality in the Near-Field (NRC, 2000a)

- Radionuclide Transport: Subissue 4—Nuclear Criticality in the Far Field (NRC, 2000b)

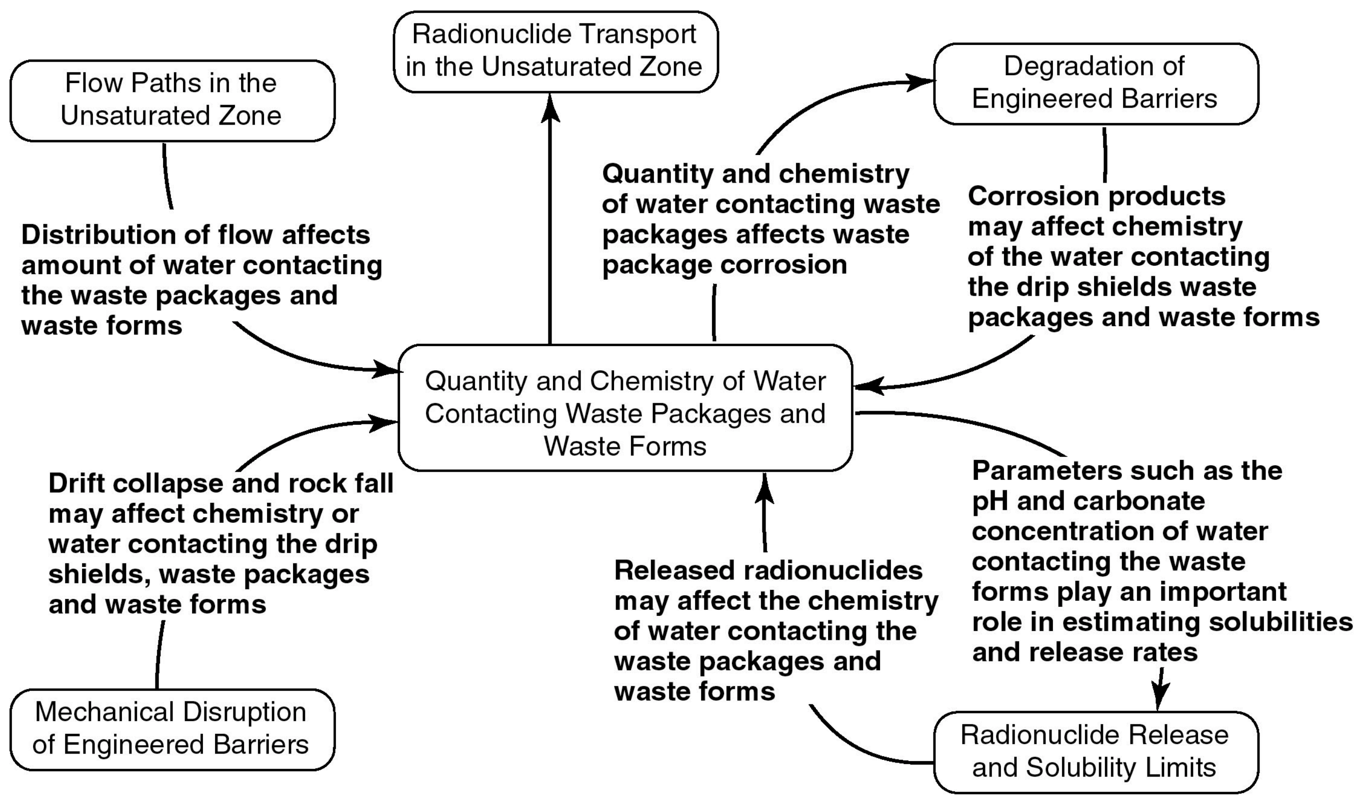

Figure 3.3.3-1. Diagram Illustrating the Relationship Between the Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms Integrated Subissue and Other Integrated Subissues

- Thermal Effects on Flow: Subissue 1—Features, Events, and Processes Related to Thermal Effects on Flow (NRC, 2000c)

- Thermal Effects on Flow: Subissue 2—Thermal Effects on Temperature, Humidity, Saturation, and Flux (NRC, 2000c)

- Container Life and Source Term: Subissue 1—The Effects of Corrosion Processes on the Lifetime of the Containers (NRC, 2000d)

- Container Life and Source Term: Subissue 3—The Rate at Which Radionuclides in Spent Nuclear Fuel are Released from the Engineered Barrier Subsystem through the Oxidation and Dissolution of Spent Nuclear Fuel (NRC, 2000d)

- Container Life and Source Term: Subissue 4—The Rate at Which Radionuclides in High-Level Waste Glass are Released from the Engineered Barrier Subsystem (NRC, 2000d)

- Container Life and Source Term: Subissue 5—The Effect of In-Package Criticality on Waste Package and Engineered Barrier Subsystem Performance (NRC, 2000d)

- Container Life and Source Term: Subissue 6—The Effects of Alternate Engineered Barrier Subsystem Design Features on Container Lifetime and Radionuclide Release from the Engineered Barrier Subsystem (NRC, 2000d)

- Repository Design and Thermal-Mechanical Effects: Subissue 3—Thermal-Mechanical Effects on Underground Facility Design and Performance (NRC, 2000e)

- Unsaturated and Saturated Flow Under Isothermal Conditions: Subissue 4—Deep Percolation (NRC, 2000f)

- Structural Deformation and Seismicity: Subissue 3—Fracturing and Structural Framework of the Geological Setting (NRC, 2000g)

- Total System Performance Assessment Integration: Subissue 1—System Description and Demonstration of Multiple Barriers (NRC, 2000g)

- Total System Performance Assessment Integration: Subissue 2—Scenario Analysis and Event Probability (NRC, 2000g)

- Total System Performance Assessment Integration: Subissue 3—Model Abstraction (NRC, 2000g)

- Total System Performance Assessment Integration: Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards (NRC, 2000g)

The key technical issue subissues formed the bases for the previous versions of the issue resolution status reports and also were the bases for technical exchanges with DOE, where agreements were reached on what additional information DOE needed to provide to resolve the subissue. The resolution status of this integrated subissue is based on the resolution status of each of the contributing key technical issue subissues. The subsequent sections incorporate applicable portions of these key technical issues subissue, however, no effort was made to explicitly identify each subissue.

3.3.3.3 Importance to Postclosure Performance

One aspect of risk informing the NRC review was to determine how this integrated subissue is related to the DOE repository safety strategy. DOE recognizes the importance of infiltration to repository performance at Yucca Mountain in the repository safety strategy for the postclosure safety case (CRWMS M&O, 2000a). Five of the DOE eight principal factors in the repository safety strategy can be related to the Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms Integrated Subissue. These principal factors are (i) seepage into emplacement drift, because this describes the quantity of water initially available to drip onto the drip shields and waste packages; (ii) performance of the drip shield/drift invert system, because performance depends on the quantity and chemistry of water contacting these materials; (iii) performance of the waste package, because performance depends on the quantity and chemistry of water contacting the waste package; (iv) radionuclide concentration limits in water, because radionuclide concentration limits in pure water may differ from the limits in the more complex water compositions expected to occur in an emplacement drift setting; and (v) radionuclide delay through the unsaturated zone, because the quantity and chemistry of water shed off the drip shield onto the inverts could influence the mobility of radionuclides by controlling precipitation and sorption processes.

3.3.3.4 Technical Basis

NRC developed a plan (2002) consistent with the acceptance criteria and review methods found in previous issue resolution status reports. A review of DOE approaches for including quantity and chemistry of water contacting the waste packages and waste forms in total system performance assessment abstractions is provided in the following subsections. The review is organized according to the five acceptance criteria identified in Section 1.5: (i) System Description and Model Integration Are Adequate, (ii) Data Are Sufficient for Model Justification, (iii) Data Uncertainty Is Characterized and Propagated Through the Model Abstraction, (iv) Model Uncertainty Is Characterized and Propagated Through the Model Abstraction, and (v) Model Abstraction Output Is Supported by Objective Comparisons.

3.3.3.4.1 System Description and Model Integration Are Adequate

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.3.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess quantity and chemistry of water contacting waste packages and waste forms with respect to system description and model integration.

The DOE technical bases for including or excluding the features, events, and processes related to the Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms Integrated Subissue are provided in the three analysis and model reports (CRWMS M&O, 2000b,c,d). Staff questions with the technical bases provided by DOE for several of these features, events, and processes. Staff comments on FEP 2.1.08.07.00 (Pathways for Unsaturated Flow and Transport in the Waste and Engineered Barrier Subsystem), address a key model integration/model abstraction concern and are most appropriately discussed in this section. The following paragraphs also provide review comments on the conceptual and modeling approach developed by DOE to integrate features, events, and processes affecting the quantity and chemistry of water in Total System Performance Assessment–Site Recommendation abstractions.

To develop the Total System Performance Assessment–Site Recommendation (CRWMS M&O, 2000e), site and design information were fed into detailed process-level models, the process-level models were abstracted for use in the Total System Performance Assessment–Site Recommendation and the inputs and outputs from the various model abstractions were integrated for internal consistency. Two of the nine groups of process-level model abstractions used in the Total System Performance Assessment–Site Recommendation directly relate to the Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms Integrated Subissue: (i) Unsaturated Zone Flow and (ii) Engineered Barrier Subsystem Environments.

The Unsaturated Zone Flow abstraction, presented in CRWMS M&O (2000f), outputs a seepage flux into the drift for the time the drift wall temperature is below boiling and, thus, provides the time-dependent quantity of seepage water that enters the emplacement drift for the majority of the 10,000-year compliance period. During the boiling period, seepage fluxes are calculated using two analysis and model reports (CRWMS M&O, 2000g,h) that evaluate the possibility that coupled effects on flow would significantly alter flow pathway, and conclude that secondary phases precipitate in volumes too small to alter rock permeabilities. Hence, seepage fluxes under both ambient and thermally perturbed conditions are taken directly from thermal-hydrological models without chemistry. Staff find this approach reasonable, but are concerned that current DOE models may not address all important features, events, and processes in models calculating seepage flux into the proposed emplacement drifts. Discussion of those concerns and associated DOE agreements follow.

DOE neglect of mineral precipitation in the vicinity of the emplacement drifts is based on the results of simulations described in the analysis and model report (CRWMS M&O, 2000h). These multiphase reactive transport simulations require special handling of mass transport and mineral reactions near computational cells that have dried completely because of vigorous heating. Some approaches to handling dry computational cells in reactive transport simulations artificially inhibit mineral precipitation at the position of the boiling front. CRWMS M&O (2000h) did not provide enough detail to determine if the simulations adequately represent mineral precipitation at the boiling front. DOE agreed[1] to provide additional documentation on the simulations pertaining to quantity of unreacted solute mass trapped in the dryout zone in TOUGHREACT simulations as well as on how this mass would affect precipitation and the resulting change in hydrologic properties.

The present DOE multiscale thermal-hydrological model approach does not adequately represent what may be called the cold-trap effect (i.e., mass movement along the length of drift, resulting from thermal gradients, causing condensation in cooler regions). This process may have occurred in the enhanced characterization of the repository block drift when it was isolated from the ventilation system by a bulkhead to allow re-equilibration to unventilated conditions. Dripping has been observed {e.g., ~10 to 30-cm [4- to 12-in]} diameter puddles, wet drip cloths, and corroded metal) in the sealed portion of the enhanced characterization of the repository block. This dripping may result from vapor-phase mobilization of water and condensation on surfaces such as rock bolts, ventilation ducts, and utility conduits under small thermal gradients. In an unventilated near-field environment where waste-canister heat causes spatial temperature variability, this process could result in significant localized dripping. It is likely that condensate would react with metal and grout at elevated but below-boiling temperatures. Alternatively, dripping in the enhanced characterization of the repository block may have resulted from seepage into the drift. DOE data at present are insufficient to distinguish what processes are primarily responsible for the observed dripping. Dripping from condensation may be masking observation of dripping from seepage. Current DOE testing in the Enhanced Characterization of the Repository Block is directed toward distinguishing the processes.

DOE has not provided an adequate evaluation of the potential cold-trap effect, but has provided a reasonable approach to do so by the time of license application, based on DOE agreements to provide additional documentation.[2] As agreed, DOE will represent the cold-trap effect in the appropriate models or provide the technical basis for exclusion of it in the various scale models (mountain, drift, and such) considering thermal effects on flow and other abstractions/models (e.g., chemistry). DOE will represent the cold-trap effect in the analysis and model report (CRWMS M&O, 2000g). This report will provide technical support for inclusion or exclusion of the cold-trap effect in the various scale models. The analysis will consider thermal effects on flow and the in-drift geochemical environment abstractions. In addition, DOE should assess the processes responsible for the observed evidence of dripping in the enhanced characterization of repository block (i.e., vapor transport and condensation or seepage) and incorporate those processes into model abstractions, if appropriate. Because the compositions of seepage and condensation water are likely to differ significantly, the additional documentation to be provided by DOE is expected to evaluate the impact of the cold-trap effect on water and gas compositions in the emplacement drifts. DOE agreed to provide a technical basis for representation of or the neglect of dripping from rockbolts in the Enhanced Characterization of

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[1]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001).” Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[2]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Thermal Effects on Flow (January 8–9, 2001).” Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

the Repository Block in performance assessment, including the impacts on hydrology, chemistry, and other impacted models at the Total System Performance Assessment and Integration Technical Exchange.[3]

In the Total System Performance Assessment–Site Recommendation (CRWMS M&O, 2000e), the composition of seepage waters is allowed to evolve in the engineered barrier subsystem environment through evaporation and salt formation processes and by variations in flow pathways within the engineered barrier subsystem. The analysis and model report (CRWMS M&O, 2001a) describes evaporation and salt formation processes in the engineered barrier subsystem by integrating two submodels, a high relative humidity model and a low relative humidity model. The high relative humidity model is represented by EQ3/6 Pitzer calculations. As these calculations are only verifiable up to an ionic strength of 10 molal, the high relative humidity model can only be used for relative humidities above 85 percent. At relative humidities lower than 85 percent, DOE employed a low relative humidity model, based on a mass balance approach. During the time relative humidity rises from 50 to 85 percent, the low relativity model simulates brine generation. This period is divided into equal time increments. For each time interval, DOE assumes that half the dissolved amount in the previous interval flows out of the local system or reactor. Currently, staff have no specific concerns related to system description/model integration for evaporation and salt formation. The staff, however, have general integration concerns related to the near-field environment, which are discussed later in this section.

DOE has not adequately defined the near-field geochemical environment that may be important to drip shield and waste package performances. Without a complete inventory of material that would be left in and surrounding the emplacement drifts after closure (i.e., a complete design), DOE predictions of the environment pertinent to drip shield and waste package performances are not adequate. Although current abstractions attempt to address some material that would be left in the emplacement drifts, elemental composition information for these materials is limited to major components and elements. DOE has not provided information on trace elements that may be important to the performances of the drip shield and waste package. In addition, because flow paths and the reaction pathways for fluid interaction are defined by local conditions, global and batch calculations do not capture the range of potential fluid compositions possible or the impact on repository performance. DOE agreed[4] to provide the technical basis for bounding the trace elements, including fluoride, for the geochemical environment affecting the drip shield and waste package, including the impact of engineered materials. DOE will document the concentrations of trace elements and fluoride in waters that could contact the drip shield and waste package in a revision to the analysis and model report (CRWMS M&O, 2000i). In addition, trace elements and fluoride concentrations in introduced materials in the engineered barrier subsystem (including cement grout, structural steels, and

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[3]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[4]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

other materials, as appropriate) will be addressed in a revision to the analysis and model report (CRWMS M&O, 2000j).

The technical basis for selecting, including, and excluding specific coupling relationships from the Total System Performance Assessment–Site Recommendation (CRWMS M&O, 2000e) is not transparent and traceable in all cases. One of the major assumptions of the DOE modeling approach for the total system performance assessment, for example, is that coupled thermal-hydrological-chemical processes can be decoupled, evaluated separately, and then recoupled, without adversely affecting predictions of repository performance. DOE has not yet provided a transparent list of the criteria used to distinguish between included and excluded couplings or an adequate technical basis for modeling decisions based on those criteria. DOE agreed[5] to identify specific coupling relationships included and excluded from total system performance assessment, including Onsager couples, and give technical bases for their inclusion or exclusion. The information will be documented in a revision to the process model report (CRWMS M&O, 2000k).

DOE has not yet provided a complete characterization of the dust expected to settle on engineered materials in the proposed repository drift environment or an analysis of how dust could affect the chemistry of water contacting the waste packages and drip shields. DOE agreed[6] to provide documentation regarding the deposition of dust and its impact on the salt analysis. DOE will document dust sampling in the Exploratory Studies Facility, analyze the dust, and evaluate its impact on the chemical environment on the surface of the drip shield and waste package in a revision to the analysis and model report (CRWMS M&O, 2000j).

Staff view integration between process model abstractions in the Total System Performance Assessment–Site Recommendation as well as between supporting process-level model analysis and model reports as a key factor in the robustness of the DOE safety case. Staff will, therefore, continue to evaluate the architecture of the Total System Performance Assessment–Site Recommendation (CRWMS M&O, 2000e) as new information becomes available.

Several integration concerns related to near-field environment models and data were expressed at the DOE and NRC technical exchange.[7] NRC was concerned with both integration between models and analyses and integration within models and analyses. The staff expressed concerns that the corrosion testing to define the potential (or lack thereof) for localized corrosion and the magnitude and variability in general corrosion was not sufficiently integrated with projections of potential in-drift environmental conditions. DOE and NRC reached

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[5]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[6]Ibid.

[7]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

agreement that DOE would complete corrosion testing in the predicted chemical environments or provide a technical basis as to why it is not needed. In addition, DOE would provide, in future documentation, a comparison of the environments predicted to those used in corrosion testing.

An area of concern for model integration and model abstraction is the screening status that DOE provided for FEP 2.1.08.07.00 (Pathways for Unsaturated Flow and Transport in the Waste and Engineered Barrier Subsystem). This feature is listed as included with regard to pathways for unsaturated flow and transport in the waste and engineered barrier subsystem in the DOE features, events, and processes database (CRWMS M&O, 2000c). This item evaluates unsaturated flow and radionuclide transport that may occur along preferential pathways in the waste form and engineered barrier subsystem. The technical basis DOE gives for the status of this item is that preferential pathways are already included via a series of linked one-dimensional flowpaths and mixing cells representing chemical evolution of the engineered barrier subsystem (CRWMS M&O, 2000c). Staff are concerned that preferred pathways in the engineered barrier subsystem are not being evaluated at the appropriate scale and, therefore, that potentially important aspects of this feature have not been included in the Total System Performance Assessment–Site Recommendation (CRWMS M&O, 2000e). Water has been observed to drip preferentially from grouted rock bolts in the enhanced characterization of the repository block (e.g., demonstrating that the introduced structures and materials themselves can influence the location of preferred flow pathways). Interactions with engineered materials, such as cementitious and metallic components, can have a significant effect on evolved water and gas compositions. DOE agreed[8] to address NRC concerns about screening arguments for features, events, and processes.

Also, in the DOE features, events, and processes database (CRWMS M&O, 2001b), the description for FEP 2.1.08.07.00 (Pathways for Unsaturated Flow and Transport in the Waste and Engineered Barrier Subsystem) includes the statement that physical and chemical properties of the engineered barrier subsystem and waste form, in both intact and degraded states, should be considered in evaluating (preferential) pathways. Hence, staff expect the screening arguments to be based on an evaluation of these topics (Evolution of the Near-Field Environment Issue Resolution Status Report Revision 03). DOE should explicitly include the possibility of localized flow pathways in the engineered barrier subsystem in total system performance assessment calculations, including the influence of introduced materials on these pathways, or provide adequate technical bases for not including this feature, event, and process. DOE agreed[9] to address NRC concerns about screening arguments for features, events, and processes.

[8]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[9]Ibid.

Features, events, or processes that DOE is evaluating according to DOE and NRC technical exchange agreements[10] for key technical issues other than evolution of the near-field environment, could, depending on the nature of the process-level model results, significantly alter water compositions in the evolution of the near-field environment. DOE should take a broad approach toward integrating key features, events, and processes between integrated subissues. The list of key technical issue subissues in Section 3.3.3.2 provides a useful resource for considering the appropriate extent of integration between the Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms Integrated Subissue and other areas of research. For clarity, a few examples of areas that may require integration efforts with the Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms Integrated Subissue are identified next.

If the drifts were to collapse onto the drip shields/waste packages, the lifetime of the drip shields/waste packages could be altered by local variations in relative humidity and chemical compositions of water and gas developed in voids between the collapsed rocks. Also, temperature versus time profiles for the drip shield/waste package surfaces may differ significantly for scenarios where the drifts do and do not collapse, resulting in different estimates of water chemistry. NRC staff review of the analysis and model report (NRC, 2000g) concluded that DOE has not provided a satisfactory basis for screening out drift collapse because thermal and seismic loadings were not represented satisfactorily in the documented analyses. DOE has agreed[11] to provide an adequate path forward for the analysis of thermal-mechanical effects, but does not relate those issues to the quantity and chemistry of water contacting the waste packages and waste forms. If future DOE analyses indicate that drift collapse is likely, the impact of drift collapse on water and gas chemistries in the engineered barrier subsystem would be evaluated. DOE agreed[12] to evaluate spatial heterogeneity on unsaturated zone flow, seepage into drifts, and transport for both ambient and drift collapse conditions.

Another concern is the thermal-mechanical effects on hydrological properties (see Section 3.3.2 of this report). DOE proposed an evaluation of thermal-mechanical effects on hydrological properties based on analyses of localized thermally induced rock response near a heated drift (CRWMS M&O, 2000l; DOE, 2001). An important case of fracture-aperture changes in the pillar between two heated drifts was not considered in the DOE analyses, however. An increase in the aperture of subhorizontal fractures in the pillar from thermal-mechanical effects is possible and would be important to cross-repository water flow because of the potential diversion of water flux from the pillar to one of the adjacent drifts, thereby focusing flux toward

[10]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[11]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Thermal Effects on Flow (January 8–9, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[12]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

the drift.[13] If future DOE evaluations of thermal-mechanical effects on hydrological properties indicate significant focused flow toward the drift, DOE would also evaluate the impact of the focused flow on repository performance with respect to the quantity and chemistry of water in the engineered barrier subsystem. DOE agreed[14] to consider this particular scenario.

In summary, system description and model integration for quantity and chemistry of water contacting waste packages and waste forms are not yet adequate. DOE agreed[15] to address these concerns in future documents.

3.3.3.4.2 Data Are Sufficient for Model Justification

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.3.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess quantity and chemistry of water contacting waste packages and waste forms with respect to data being sufficient for model justification.

In the Total System Performance Assessment–Site Recommendation (CRWMS M&O, 2000e), response surfaces describing temperature, humidity, liquid saturation, pH, total carbonate, ionic strength, and seepage flux are evaluated by the Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms Integrated Subissue. The process-level and conceptual models used to define these response surfaces depend on a wide range of information and data including waste form properties, engineered barrier subsystem material properties, drip shield and waste package design properties, repository design properties, site geohydrology, and site geochemistry. Because of the inherent interconnectedness between the Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms Integrated Subissue and other key technical issues and integrated subissues (see Section 3.3.3.2), DOE should evaluate staff comments raised in other sections of this report for applicability to the Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms Integrated Subissue.

Currently, insufficient data are being used to constrain the chemistry of brine solutions under low relative humidity conditions in the analysis and model report (CRWMS M&O, 2001a). Drip shield and waste package degradation are expected to be most active when the repository is still hot and the deliquescent humidity has been reached. The characterization of high ionic strength solution chemistries (e.g., greater than 10 molal) in complex natural environments exceeds the limitations of the DOE modeling approach and may be best characterized by experimental data. Interpolation techniques DOE used in the low relative humidity model are

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[13]Ofoegbu, G.I., S. Painter, R. Chen, R.W. Fedors, and D.A. Ferrill. “Geomechanical and Thermal Effects on Moisture flow at the Proposed Yucca Mountain Nuclear Waste Repository.” *Nuclear Technology*. Vol. 134. In press. June 2001.

[14]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001).” Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[15]Ibid.

insufficient to constrain the models. Modeling results, such as predicted concentrations of major anionic and cationic species, are inadequately described.

Also, current DOE technical bases are not adequate to justify the assumption that pure salts will define the deliquescent humidity. The minimum deliquescent humidity of a salt mixture is typically lower than the individual deliquescent humidity. Although 50 percent is the lowest deliquescent humidity for all the pure salts considered in CRWMS M&O (2001a), the deliquescent humidity of the salt mixture in the Yucca Mountain waters may be even lower.

To address the concerns in the preceding paragraphs, DOE agreed[16] to provide a revision of the analysis and model report (CRWMS M&O, 2001a) that includes (i) the major anionic (e.g., fluoride or chloride) and cationic species and (ii) additional technical bases for the low relative humidity model. The data should provide the technical basis why the assumption of the presence of sodium nitrate is conservative, when modeling and experimental results indicate the presence of other mineral phases for which the deliquescence points are unknown. DOE will provide additional information to constrain the low relative humidity salts model. The information will include the deliquescent behavior of mineral assemblages derived from alternative starting water compositions (including bulk water compositions and local variations associated with cement leaching or the presence of corrosion products) representing the range of potential water compositions in the emplacement drifts.

In view of safety insights achieved since the DOE and NRC technical exchange[17] staff reassessed the analysis and model report (CRWMS M&O, 2001a) and identified several repository performance concerns. Although DOE considers evaporation and salt formation processes in the engineered barrier subsystem throughout the 10,000-year compliance period, NRC staff continue to focus its review of these models on the initial deliquescent period, when brine solutions are likely to be most corrosive. Staff concerns are described in the following two paragraphs.

The analysis and model report (CRWMS M&O, 2001a) assumes that during discrete time intervals, half the dissolved amount in the previous time interval would flow out of the reactor as soon as a minimum relative humidity is reached. For soluble species such as nitrates, the fraction, $f_{i,\,(k\text{-}1)/2}$, in Eq. 3 (CRWMS M&O, 2001a) is unity, and all the accumulated nitrate salts would be dissolved in the liquid. Hence, DOE would predict that most accumulated salt will flow out of the reactor in a few time intervals. Staff are concerned that this may be an unrealistic artifact of the modeling approach. The following scenario was predicted by staff evaluations, and has raised concerns with the DOE models. In this scenario, only a small amount of concentrated liquid is present after the deliquescent humidity is reached. The concentrated liquid stays in the pores of the deposits until the liquid volume reaches a point where the solid deposit could no longer hold the liquid. This scenario does not agree with the rapid depletion of accumulated salts predicted by DOE models following the onset of deliquescence, and

[16]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[17]Ibid.

suggests DOE may be underestimating the amount of nitrate salts on the waste package. DOE should provide stronger technical bases for the approach used in the low relative humidity model during the initial stages of deliquescence, when the potential for corrosion is highest. As part of the DOE and NRC technical exchange,[18] DOE agreed to provide technical basis for the simplifications used when developing model abstractions.

In the analysis and model report (CRWMS M&O, 2001a), DOE assumes all accumulated nitrate salts are dissolved in the liquid as soon as a minimum relative humidity is reached. The total calculated volume of water is large at this time, causing concentrations for other important species such as Cl^- to be extremely low. Staff are concerned the DOE model may significantly underestimate concentrations of these aqueous species following deliquescence. DOE should provide additional technical bases explaining why these species concentrations are not limited by solubilities of the salts. DOE agreed[19] to provide additional technical bases for the low relative humidity salts model, including the major anionic and catronic species.

The data DOE used to calibrate and validate several process-level models providing input into total system performance assessment are not adequate, and the technical reliability and representativeness of these data have not been adequately evaluated. A significant amount of experimental data was collected for use in the analysis and model report (CRWMS M&O, 2000h). Insufficient analyses, however, were performed to interpret the data and to establish that parameter values are bounded. In addition, the criterion used to include and exclude individual water and gas measurements for use in these models has not been clearly documented. Similar concerns exist about the reliability data used to validate and calibrate the analysis and model report (CRWMS M&O, 2001a). Finally, validation efforts can only be as robust as the data they are being validated against. Thus, DOE should fully scrutinize the reliability of data used to validate this model and provide this information to NRC staff for review. DOE did not make a transparent distinction between calibration and validation efforts in either of these analysis and model reports. Data should be used to either calibrate or validate, but not to simultaneously calibrate and validate.

To address the previous concerns, DOE agreed[20] to provide additional documentation on the data used to calibrate models and to support model predictions. In addition, the DOE agreed to assess data uncertainty (e.g., sampling and analytical), including critical analyses of variables that affect the data measurements and their interpretations (e.g., drift-scale thermal and evaporation tests). DOE will provide documentation of data used to calibrate models and of data to support model predictions, together with an assessment of data uncertainties (e.g., sampling and analytical) in the areas of water and gas chemistries, from the drift-scale

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[18]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[19]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[20]Ibid.

thermal tests and evaporation tests. This documentation will be provided in revisions to the analysis and model report (CRWMS M&O, 2000i) or in another future document.

DOE has not demonstrated that water and gas chemistry analyses used as initial conditions in process-level models supporting total system performance assessment calculations are appropriately bounding. The level of detailed information DOE provided on the full water chemistry, including trace metals potentially important to drip shield and waste package performance, is insufficient.

To address the previous concern, DOE agreed[21] to provide additional information about the range of water composition that could contact the drip shield or waste package, including whether such waters are of the bicarbonate or chloride-sulfate type. DOE will describe the range of bulk composition for waters that could affect corrosion of the drip shield or waste package outer barrier in a revision to the analysis and model report (CRWMS M&O, 2000i).

DOE has not yet provided sufficient data to support models of coupled thermal-hydrological-chemical processes on the waste package environment. Silica mobility may play an important role in models predicting the quantity and chemistry of water contacting the waste packages, but kinetic parameters for the silicate phases present at Yucca Mountain are poorly understood. DOE has not sufficiently constrained coupled thermal-hydrological-chemical models of Yucca Mountain with site-specific experimental data.

To address the previous concern, DOE agreed[22] to provide documentation of the results obtained from the crushed tuff hydrothermal column experiment and of posttest analysis in new reports specific to the column test.

DOE has not adequately considered changes in local water and gas chemistries because of interactions with engineered materials, such as grouted rock bolts, along preferential flow pathways. The current total system performance assessment approach weights the volumetric contribution made by local variations in water and gas chemistries against bulk engineered barrier subsystem water and gas chemistries to evaluate the potential impact of local chemistry on repository performance. Staff are concerned this approach does not adequately address the potential impact that preferential pathways could have on the chemistry of water contacting the drip shield and waste packages, because it does not allow the composition of water moving along these pathways to deviate from the bulk engineered barrier subsystem composition.

[21]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[22]Ibid.

To address this concern, DOE agreed[23] to provide the analyses of laboratory solutions that have interacted with introduced materials. DOE will provide additional information about laboratory solutions that have interacted with introduced materials in a revision to the analysis and model report (CRWMS M&O, 2000i). DOE will also reevaluate the impact of these water compositions in the context of preferential flow pathways in the total system performance assessment and repository performance.

NRC staff expressed concern that unmeasured loss of mass from the heated drift complicates analysis of the Drift Scale Test results and may ultimately compromise utility of the Drift Scale Test for evaluating refluxing during the thermal phase of the proposed repository design. DOE maintained that “more accurate characterization of the heat loss through the bulkhead” is difficult, problematic, and unnecessary (CRWMS M&O, 1999). Because of concerns regarding these uncertainties, however, DOE decided to take a dual approach to quantifying mass and energy losses through the bulkhead (CRWMS M&O, 2000m). First, a proposal by the University of Nevada to measure losses in a manner requiring sealing of the cable bundles and other leakage through the bulkhead would be pursued. Second, the DOE thermal test team would deploy a series of humidity and temperature sensors along the drift immediately outside the bulkhead. Muffin fans would be used to ensure proper air movement and to prevent condensation. Both approaches would be implemented in fiscal year 2001 if funding were approved (CRWMS M&O, 2000m). DOE reversed this position at the DOE and NRC Thermal Effects on Flow Technical Exchange,[24] however, stating that measuring mass and energy losses through the bulkhead of the Drift Scale Test is not necessary for the intended use of the Drift Scale Test results.

To address these concerns, DOE agreed[25] to provide additional documentation to address mass and energy losses through the Drift Scale Test bulkhead. In addition, DOE will provide NRC with a White Paper on the technical basis for DOE understanding of heat and mass losses through the bulkhead. This White Paper will address uncertainty in the fate of thermally mobilized water in the Drift Scale Test and also the effect this uncertainty has on conclusions drawn from the Drift Scale Test results. NRC will provide comments on this white paper. DOE will analyze the effects of this uncertainty on the uses of the Drift Scale Test in response to NRC comments.

DOE data from ventilation testing are not sufficient to support the ventilation model. The design objective of maintaining pillar temperatures below boiling to allow for condensate drainage between emplacement drifts depends on the efficacy of the ventilation system. The analysis and model report (CRWMS M&O, 2000n) shows 70 percent heat removal by ventilation flow

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[23]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001).” Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[24]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Thermal Effects on Flow (January 8–9, 2001).” Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[25]Ibid.

rates between 10 and 15 m^3/s [350 and 530 ft^3/s]. This model involves simplifying assumptions, however, and is not supported by experimental data. Plans have been developed for a quarter-scale ventilation test to be conducted at the Engineered Barrier Subsystem Test Facility in North Las Vegas, Nevada (CRWMS M&O, 2000n). This test needs to be completed to provide data for support and verification of the ventilation model. To address these concerns, DOE will provide[26] the detailed test plan for Phase III of the ventilation test. NRC comments on the test plan will be considered by DOE before initiation. DOE will provide the analysis and model report (CRWMS M&O, 2000n) and CRWMS M&O (2000o). Test results will be provided in an update to CRWMS M&O (2000n).

In summary, data sufficiency and model justification for quantity and chemistry of water contacting waste packages and waste forms are not yet adequate, but DOE agreed[27] to address staff concerns in future documents.

3.3.3.4.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.3.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess quantity and chemistry of water contacting waste packages and waste forms with respect to data uncertainty being characterized and propagated through the model abstraction.

Uncertainties in data used to constrain individual in-drift geochemical submodels have yet to be adequately evaluated and documented, and the impact of these uncertainties on the predicted quantity and chemistry of water contacting the waste packages and waste forms has not yet been propagated through total system performance assessment calculations. DOE agreed[28] to evaluate data and model uncertainties for specific in-drift geochemical environment submodels used in total system performance assessment calculations and propagate those uncertainties. DOE will document the evaluation in an update to the analysis and model report (CRWMS M&O, 2000j) (or in another future document). DOE also agreed to address the various sources of uncertainty [e.g., model implementation, conceptual model, and data uncertainty (hydrologic, thermal, and geochemical)] in the thermal-hydrological-chemical model. DOE will evaluate the various sources of uncertainty in the thermal-hydrological-chemical process model, including details on how the propagation of various sources of uncertainty is calculated in a systematic uncertainty analysis, and document those sources in a revision to the analysis and model report (CRWMS M&O, 2000h) (or in another future document).

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[26]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[27]Ibid.

[28]Ibid.

Several concerns related to propagation of uncertainty in near-field environment models and data were expressed at the DOE and NRC technical exchange.[29] The DOE models on the near-field environment propagated a limited amount of uncertainty from upstream sources. Staff expressed concern that the uncertainty in the environmental conditions generated by the DOE models did not adequately propagate all significant sources of uncertainty, therefore, leading to an underprediction in the range of expected environmental conditions. DOE agreed[30] to address the NRC concerns in several agreements.

DOE thermal-hydrological calculations used to support seepage fluxes do not currently account for measurement error, bias, and scale dependence in the saturation, water potential, and pneumatic pressure data. Standard deviation of saturation data from cores was used to estimate weights for the weighted least-squares inverse algorithm (CRWMS M&O, 2000p), however, the effect of measurement errors on the resulting calibrated properties was not evaluated. Three types of data (matrix saturation from cores, water potential from boreholes, and pneumatic pressures) were measured on different scales ranging from a few centimeters for cores to several tens of meters or more for pneumatic pressures. Matrix saturations from core data were upscaled by arithmetic averaging, a process that tends to smooth out variability; but it is not clear how the scale dependence of the water potentials and pneumatic pressure data were treated. Pneumatic pressure data are known to be scale dependent because fracture permeabilities from barometric pumping responses tend to be about two orders of magnitude greater than fracture permeabilities determined from air-injection testing (CRWMS M&O, 2000p). This information is important because property sets developed in the calibrated properties analysis and model report are used in the unsaturated zone flow models (and multiscale thermohydrological model) essentially deterministically. That is, a single property set for each high-, median-, and low-infiltration condition is assumed to capture all the variability and uncertainty in the model. Propagation of uncertainty from unsaturated zone and multiscale thermohydrologic process models to model abstractions necessitates incorporating all sources of uncertainty.

The nonlinear least-squares maximum likelihood inverse method implemented in ITOUGH2 accounts for uncertainty through measurement error. Thus, the measurement error must be generalized to include other sources of uncertainty, such as scale dependence and modeling errors, because there is no other way to account for uncertainty in the least-squares inverse approach (McLaughlin and Townley, 1996).

DOE presented a discussion of the conceptual model used to develop the calibrated property sets used in the analysis and model report (CRWMS M&O, 2000o) stating that [h]eterogeneity of hydrologic properties is predominantly a function of geological layering, and therefore, each geological layer in the model is treated as homogeneous. The resulting average layer-calibrated, layer-averaged, drift-scale property sets for the basecase show fracture permeability in the Topopah Spring (Tsw34) unit to be 2.76E–13 m^2 [2.97E–12 ft^2] and in the Tsw35 unit to

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[29]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001).” Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[30]Ibid.

be 1.29E–12 m^2 [1.39E–11 ft^2]. For the upper-bound infiltration map, these change to 4.63E–13 m^2 [4.98 E–12 ft^2] and 5.09E–12 m^2 [5.48E–12 ft^2] and for the lower-bound, to 4.99E–13 m^2 [2.97E–12 ft^2] and 1.82E–12 m^2 [1.96E–11 ft^2] for the Tsw34 and Tsw35 units, respectively. Thus, all the variability and uncertainty in model layer fracture permeability for these two units ranges within approximately one order of magnitude. A statistical analysis of air-injection data collected from the niches in the Exploratory Studies Facility, however, found fracture permeabilities ranging from 1.53E–15 m^2 to 7.15E–10 m^2 [1.65E–14 to 7.70 E–9 ft^2]. These data, collected in the Tsw34 unit, indicate that heterogeneity of fracture permeability can range at least four orders of magnitude within a single geological layer. The DOE use of homogeneous layer properties in a model, with variability ranging only one order of magnitude, does not adequately represent variability and uncertainty that may range several orders of magnitude within a single geological layer.

To address these concerns, DOE agreed[31] to provide additional documentation to address data uncertainty that will be provided as part of the issue resolution process and, if provided by DOE by the time of any license application, should afford sufficient information for NRC to conduct its licensing review. DOE will provide documentation of analyses of spatially heterogeneous fracture permeability using refinement of the grid for the heterogeneous fields in three dimensions and will evaluate the effect of high-permeability features (e.g., faults) crossing the drifts. DOE will consider the NRC suggestion to compare the numerical model results with the Phillips (1996) analytical solution.

In summary, data uncertainty being characterized and propagated through the model abstraction with regard to quantity and chemistry of water contacting waste packages and waste forms is not yet adequate, but DOE agreed[32,33] to address all staff concerns in future documents.

3.3.3.4.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.3.5), is sufficient to ensure that the information will be available at the time of a potential license application to assess quantity and chemistry of water contacting waste packages and waste forms with respect to model uncertainty being characterized and propagated through the model abstraction.

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[31]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Thermal Effects on Flow (January 8–9, 2001).” Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[32]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001).” Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[33]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Thermal Effects on Flow (January 8–9, 2001).” Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

DOE has not yet documented how different flow pathways impact total system performance assessment predictions of the Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms Integrated Subissue. Even for ambient conditions (Browning, et al., 2000), water and gas compositions will vary, depending on the types of materials encountered along a particular flow pathway and the duration of those interactions. The Total-system Performance Assessment code evaluates several different flow pathways in the engineered barrier subsystem, but does not adequately consider local changes in water and gas chemistries that may result from interactions with engineered materials, corrosion products, or both (such as cement-grouted rock bolts) located above the drip shield. Staff are also concerned that water, affected by these interactions may impact performance.

To address these concerns, DOE agreed[34] to evaluate the impact of the range of local chemistry (e.g., dripping of equilibrated evaporated cement leachate and corrosion products) conditions at the drip shield and waste package, considering the chemical divide phenomena that may propagate small uncertainties into large effects. DOE should also evaluate the range of local chemical conditions at the drip shield and waste package (e.g., local variations in water composition associated with cement leaching or the presence of corrosion products), considering potential evaporative concentration. This evaluation will be documented in a revision to the analysis and model report (CRWMS M&O, 2000j). DOE should determine whether calculated water compositions for various flow pathways in the engineered barrier subsystem are significant, given the uncertainties in the data and models.

Inadequate technical bases have been provided for the DOE major assumption that all reactions proceed to equilibrium. The suppression of mineral precipitation in process-level models supporting total system performance assessment is an acknowledgment of the role of kinetics, but DOE has not yet documented the criteria used to identify which mineral reactions are suppressed and the conditions under which the suppression of these reactions is applicable. To address this concern, DOE agreed[35] to provide stronger technical bases for the suppression of individual mineral reactions predicted by equilibrium models in a revision to the analysis and model report (CRWMS M&O, 2000j). DOE also agreed[36] to provide the technical basis for current treatment of the kinetics of chemical processes in the in-drift geochemical models in a revision to the analysis and model report (CRWMS M&O, 2000j). The technical basis will include reaction progress simulation for laboratory evaporative concentration tests and appropriate treatment of time as related to the residence times associated with the abstractions used to represent in-drift processes in total system performance assessment.

Two different models were presented in the analysis and model report (CRWMS M&O, 2000h) that provide some insight into model uncertainty. The two models differ mainly in the number of minerals and dissolved elemental components considered, and the model results show that the

[34]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[35]Ibid.

[36]Ibid.

limited suite mineral model provides a closer match to Drift Scale Test data. DOE has not yet provided sufficient technical bases demonstrating the output from these two cases bound the quantity and chemistry of water contacting the waste packages and waste forms. In contrast to the DOE claim that infiltration rates are the major uncertainty in thermal-hydrological and thermal-hydrological-chemical models (CRWMS M&O, 2000e), NRC staff believe that uncertainties associated with the representation of complex chemical interactions are equally significant. Stronger technical bases are needed for the DOE exclusion of chemistry-related uncertainties in total system performance assessment abstractions of seepage compositions. Additional technical bases are also needed for the lack of spatial variability in Total System Performance Assessment–Site Recommendation Abstraction for Seepage Water and Gas Chemistries. To address these concerns, DOE agreed[37] to provide a revision of the analysis and model report (CRWMS M&O, 2000h) that includes information supporting both the limited suite mineral model and the more complete extended model. In addition, DOE should provide sufficient technical bases demonstrating that the output from these two cases bound predictions of the quantity and chemistry of water contacting the waste packages and waste forms, given that both models most closely approximate conditions near the center of the potential repository.

CRWMS M&O (2000g) uses only the drift-scale property sets to calculate thermohydrologic variables. It is not clear how this captures the variability and uncertainty seen in predictions using other property sets or the uncertainty in comparisons to actual test results. Note that, to date, all thermal tests at Yucca Mountain have been conducted in the middle nonlithophysal unit of the Tsw34, so all conclusions of the analysis and model report (CRWMS M&O, 2000q) apply only to that unit. Thus, it seems reasonable that if the analyses were performed on the remaining geological units, the predicted variability and uncertainty would be greater. Further, the analysis and model report (CRWMS M&O, 2000p) recommends that future studies should consider the use of Monte Carlo simulations to evaluate the appropriateness of using the prior information uncertainty for the calibrated properties. Such exercises would be useful for evaluating the propagation of uncertainty through the least-squares inverse approach, as discussed previously. This approach would not address the uncertainty inherent in spatial heterogeneity nor would it adequately address the uncertainty in the equally valid but significantly different models and property sets of CRWMS M&O (2000q). Additional studies applying generally accepted methods of stochastic subsurface hydrology, sensitivity, and bounding analyses would be required to address the data and model uncertainties.

To address this concern, DOE agreed[38] to provide additional documentation to address model uncertainty. DOE will represent the full variability/uncertainty in the results of the thermal effects on flow simulations in the abstraction of thermodynamic variables to other models or provide technical bases that a reduced representation is appropriate (considering risk

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[37]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001

[38]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Thermal Effects on Flow (January 8–9, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

significance). DOE will provide an updated calibrated properties analysis and model report (CRWMS M&O, 2000p) that incorporates uncertainty from all significant sources. DOE will consider model uncertainty, including (i) types of model uncertainty, (ii) flow conceptualization for ambient conditions, (iii) flow conceptualization for thermal conditions, (iv) fracture flow for ambient and thermal conditions, (v) fracture matrix interaction model evolution, (vi) discrete fracture description, and (vii) model uncertainty reduction.

In summary, characterization and propagation of model uncertainty through the model abstraction, as applied to quantity and chemistry of water contacting waste packages and waste forms, are not yet adequate. DOE agreed[39] to address staff concerns in future documents.

3.3.3.4.5 Model Abstraction Output Is Supported by Objective Comparisons

Overall, the current information along, with agreements reached between the DOE and NRC (Section 3.3.3.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess quantity and chemistry of water contacting waste packages and waste forms with respect to model abstraction output being supported by objective comparisons.

Although this integrated subissue deals with water in the drift, the composition of seepage water is likely influenced by the phases in the unsaturated fractured rock with which it reacts. Geochemical modeling has been used to predict the composition of the water seeping into the drifts. The predictions resulting from geochemical modeling are uncertain. Sources of uncertainty include the modeler decisions on components to include or exclude in the system studied, kinetics of reactions, surface areas of minerals and fractures and activity coefficients of species in the aqueous and solid phases. DOE agreed[40] to provide physical evidence to support the model of fracture/matrix interaction by overcoring in the Single Heater Test and side-wall sampling mineralogy/petrology of the Drift Scale Test. Comparison of pre and posttest mineral assemblages, looking for evidence of alteration, and redistribution can be used to support predictive models.

In addition to seepage water, increased attention is currently being paid to condensation. Evidence suggests that condensation is occurring behind the bulkhead of the Enhanced Characterization of the Repository Block, where conditions are unventilated, and relative humidity is high. DOE is conducting experiments to address concerns related to condensation. If the experiments suggest that a significant portion of the water that could contact the waste

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[39]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[40]Ibid.

packages and waste form is condensate, DOE agreed[41] to represent this process in appropriate models, including the thermal effects on flow and the in-drift geochemical environment.

DOE should provide model support by predicting thermohydrologic results of the Cross Drift Thermal Test to verify that the thermohydrologic model abstraction adequately represents the potential thermohydrologic conditions expected in the proposed repository. DOE should identify and implement a useful approach toward verifying total system performance assessment predictions of engineered barrier subsystem environments in the proposed repository setting. Numerical simulations are used to predict the occurrence (or lack) of mineral precipitation around the emplacement drifts. Conditions of above-boiling temperatures may persist for hundreds to thousands of years. Resulting from the numerous sources of uncertainty, strong model support is needed for the numerical result of no significant alteration around the emplacement drifts. Staff have discussed this concern with DOE. DOE stated that significant alteration has not been observed for the Drift Scale Test and that this provides sufficient support for the modeling result of limited mineral alteration around the emplacement drifts. The difficulty with using this piece of information as the primary support for the modeling result is the temporal scales associated with the processes. For instance, if the minerals in the Drift Scale Test were being altered at a rate of less than 1 percent per year, most of the characterization performed to date, or the observation of thermodynamic variables, would be unable to resolve the magnitude of the alteration. This amount of alteration during hundreds to thousands of years, however, could have significant impacts on the quantity and chemistry of water contacting the waste packages and waste forms. Additional sources of information (e.g., simulation of laboratory experiments on silica precipitation) should be used to provide additional model support. DOE agreed to address concerns related to model support in future documentation.[42]

In summary, DOE has not provided sufficient evidence, either through field tests or natural analogs, that modeling results of quantity and chemistry of water contacting waste packages and waste forms are sufficient for inclusion in license application. DOE agreed to address the concerns described previously with the results from field tests in the enhanced Characterization of the Repository Block, and from laboratory experiments.

3.3.3.5 Status and Path Forward

Table 3.3.3-1 provides the status of all key technical issue subissues, referenced in Section 3.3.3.2, for the Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms Integrated Subissue. The table also provides the related DOE and NRC agreements pertaining to the Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms Integrated Subissue. The agreements listed in the table are associated

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[41]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Thermal Effects on Flow (January 8–9, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[42]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

with one or all five generic acceptance criteria discussed in Section 3.3.3.4. Note that the status and the detailed agreements (or path forward) pertaining to all the key technical issue subissues are provided in Table 1.1-3 and Appendix A.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (through specified testing, analyses, and the like), acceptably addresses the NRC questions so that no information beyond that provided, or agreed to, will likely be required at the time of a potential license application.

Table 3.3.3-1. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreements*
Evolution of the Near-Field Environment	Subissue 1—Effects of Coupled Thermal-Hydrological-Chemical Processes on Seepage and Flow	Closed-Pending	ENFE.1.01 ENFE.1.03 through ENFE.1.07
	Subissue 2—Effects of Coupled Thermal-Hydrological-Chemical Processes on the Waste Package Chemical Environment	Closed-Pending	ENFE.2.01 ENFE.2.03 through ENFE 2.18
	Subissue 3—Effects of Coupled Thermal-Hydrological-Chemical Processes on the Chemical Environment for Radionuclide Release	Closed-Pending	ENFE.3.01 ENFE.3.02 ENFE.3.03 ENFE.3.05
	Subissue 4—Effects of Coupled Thermal-Hydrological-Chemical Processes on Radionuclide Transport through Engineered and Natural Barriers	Closed-Pending	ENFE.4.01 ENFE.4.02 ENFE.4.03 ENFE.4.04
	Subissue 5—Effects of Coupled Thermal-Hydrological-Chemical Processes on Potential Nuclear Criticality in the Near Field	Closed-Pending	ENFE.5.01
Thermal Effects on Flow	Subissue 1—Features, Events, and Processes Related to Thermal Effects on Flow	Closed-Pending	TEF.1.01
	Subissue 2—Thermal Effects on Temperature, Humidity, Saturation, and Flux	Closed-Pending	TEF.2.01 TEF.2.02 TEF.2.05 through TEF.2.08 TEF.2.10 TEF.2.11

Table 3.3.3-1. Related Key Technical Issue Subissues and Agreements (continued)			
Key Technical Issue	**Subissue**	**Status**	**Related Agreements***
Container Life and Source Term	Subissue 1—The Effects of Corrosion Processes on the Lifetime of the Containers	Closed-Pending	None
	Subissue 3—The Rate at Which Radionuclides in Spent Nuclear Fuel Are Released from the Engineered Barrier Subsystem through the Oxidation and Dissolution of Spent Nuclear Fuel	Closed-Pending	CLST.3.02 CLST.3.04
	Subissue 4—The Rate at Which Radionuclides in High-level Waste Glass Are Released from the Engineer Barrier Subsystem	Closed-Pending	CLST.4.02 CLST.4.04
	Subissue 5—The Effects of In-Package Criticality on Waste Package and Engineered Barrier Subsystem Performance	Closed-Pending	CLST.5.01 CLST.5.05
	Subissue 6—The Effects of Alternate Engineered Barrier Subsystem Design Features on Container Lifetime and Radionuclide Release from the Engineered Barrier Subsystem	Closed-Pending	None
Radionuclide Transport	Subissue 4—Nuclear Criticality in the Far Field	Closed-Pending	RT.4.03
Repository Design and Thermal-Mechanical Effects	Subissue 3—Thermal-Mechanical Effects on Underground Facility Design and Performance	Closed-Pending	RDTME.3.20 RDTME.3.21
Unsaturated and Saturated Flow Under Isothermal Conditions	Subissue 4—Deep Percolation	Closed-Pending	None
Structural Deformation and Seismicity	Subissue 3—Fracturing and Structural Framework of the Geological Setting	Closed-Pending	SDS.3.03 SDS.3.04

Table 3.3.3-1. Related Key Technical Issue Subissues and Agreements (continued)			
Key Technical Issue	**Subissue**	**Status**	**Related Agreements***
Total System Performance Assessment and Integration	Subissue 1—System Description and Demonstration of Multiple Barriers	Closed-Pending	None
	Subissue 2—Scenario Analysis and Event Probability	Closed-Pending	TSPAI.2.01 TSPAI.2.02
	Subissue 3—Model Abstraction	Closed-Pending	TSPAI.3.07 through TSPAI.3.13
	Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards	Closed-Pending	None

*Related DOE and NRC agreements are associated with one or all five generic acceptance criteria.

Note: Key Technical Issue Agreement GEN.1.01 pertains to multiple integrated subissues, as well as some specific issues related to this integrated subissue.

3.3.3.6 References

Browning, L., W.M. Murphy, B.W. Leslie, and W.L. Dam. "Thermodynamic Interpretations of Chemical Analyses of Unsaturated Zone Water from Yucca Mountain, Nevada." Scientific Basis for Nuclear Waste Management XXIII. R.W. Smith and D.W. Shoesmith, eds. Symposium Proceedings 608. Pittsburgh, Pennsylvania: Materials Research Society. pp. 237–242. 2000.

CRWMS M&O. "Thermal Test Progress Report #4." T200001170008. Las Vegas, Nevada: CRWMS M&O. 1999.

———. "Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations." TDR–BWI–SBR–LB000001. Revision 4 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000a.

———. "Features, Events, and Processes in UZ Flow and Transport." ANL–NBS–MD–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000b.

———. "Near-Field Environment Process Model Report." TDR–NBS–MD–000001. Revision 00 ICN 03. Las Vegas, Nevada: CRWMS M&O. 2000c.

———. "Miscellaneous Waste Form FEPs." ANL–WIS–MD–000009. Revision 00. Las Vegas, Nevada: CRWMS M&O: 2000c.

———. "EBS FEPs/Degradation Modes Abstraction." ANL–WIS–PA–000002. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000d.

———. "Total System Performance Assessment for the Site Recommendation." TDR–WIS–PA–000001. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000e.

———. "Abstraction of Drift Seepage." ANL–NBS–HS–000005. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000f.

———. "Multiscale Thermohydrologic Model." ANL–EBS–MD–000049. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000g.

———. "Drift-Scale Coupled Processes (DST and THC Seepage) Models." MDL–NBS–HS–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000h.

———. "Environment on the Surfaces of the Drip Shield and Waste Package Outer Barrier." ANL–EBS–MD–000001. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000i.

———. "Engineered Barrier Subsystem: Physical and Chemical Environment Model." ANL–EBS–MD–000033. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000j.

———. "Engineered Barrier Subsystem Degradation, Flow, and Transport Process Model Report." TDR–BEB–SBM–DB–000006. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000k.

———. "Calculation of Permeability Change Due to Coupled Thermal-Hydraulic-Mechanical Effects." CAL–NBS–MD–000002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000l.

———. "Thermal Test Progress Report #5." Las Vegas, Nevada: CRWMS M&O. 2000m.

———. "Ventilation Model." ANL–EBS–MD–000030. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000n.

———. "Pre-Test Predictions for Ventilation Tests." CAL–EBS–MD–000013. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000o.

———. "Calibrated Properties Model." ANL–NBS–HS–000003. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000p.

———. "Thermal Tests Thermal-Hydrological Analyses/Model Report." ANL–NBS–TH–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000q.

———. "In-Drift Precipitate/Salts Analysis." ANL–EBS–MD–000045. Revision 00 ICN 02. Las Vegas, Nevada: CRWMS M&O. 2001a.

———. "FEPs Database." TDR–WIS–MD–000003. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2001b.

DOE. “Subissue 3, Thermal-Mechanical Effects on Underground Facility Design and Performance—Component 3, Thermal-Mechanical Effects on Flow into Emplacement Drifts.” *Presentation at DOE and NRC Technical Exchange on the Key Technical Issue and Subissues Related to Repository Design and Thermal-Mechanical Effects*. 2001.

McLaughlin, D. and L.R. Townley. “A Reassessment of the Groundwater Inverse Problem.” *Water Resources Research*. Vol. 32, No. 5. pp. 1,131–1,161. 1996.

NRC. “Issue Resolution Status Report, Key Technical Issue: Evolution of the Near-Field Environment.” Revision 3. Washington, DC: NRC. 2000a

———. “Issue Resolution Status Report, Key Technical Issue: Radionuclide Transport.” Revision 2. Washington, DC: NRC. 2000b.

———. “Issue Resolution Status Report, Key Technical Issue: Thermal Effects on Flow, Revision 3. Washington, DC: NRC. 2000c.

———. “Issue Resolution Status Report Key Technical Issue: Container Life and Source Term.” Revision 3. Washington, DC: NRC. 2000d.

———. “Issue Resolution Status Report, Key Technical Issue: Repository Design and Thermal-Mechanical Effects. Revision 3. Washington, DC: NRC. 2000e.

———. “Issue Resolution Status Report, Key Technical Issue: Unsaturated and Saturated Flow Under Isothermal Conditions.” Revision 2. Washington, DC: NRC. 2000f.

———. “Issue Resolution Status Report, Key Technical Issue: Total System Performance Assessment and Integration.” Revision 3. Washington, DC: NRC. 2000g.

———. “Issue Resolution Status Report, Key Technical Issue: Structural Deformation and Seismicity.” Revision 3. Washington, DC: NRC. 2000h.

———. NUREG–1804. “Yucca Mountain Review Plan—Draft Report for Comment.” Revision 2. Washington DC: NRC. March 2002.

Phillips, O.M. “Infiltration of a Liquid Finger Down a Fracture Into Superheated Rock.” *Water Resources Research*. Vol. 32, No. 6. pp. 1,665–1,670. 1996.

3.3.4 Radionuclide Release Rates and Solubility Limits

3.3.4.1 Description of Issue

The Radionuclide Release Rates and Solubility Limits Integrated Subissue addresses the release of radionuclides from the engineered barrier subsystem to the geosphere. The relationship of this integrated subissue to other subissues are depicted in Figure 3.3.4-1. The overall organization and identification of all the integrated subissues are depicted in Figure 1.2-2. This section provides a review of the abstractions of radionuclide release rates and solubility limits incorporated by the DOE in its Total System Performance Assessment–Site Recommendation (CRWMS M&O, 2000a,t).

3.3.4.2 Relationship to Key Technical Issue Subissues

The Radionuclide Release Rates and Solubility Limits Integrated Subissue incorporates subject matter previously captured in the following key technical issue subissues:

- Container Life and Source Term: Subissue 3—The Rate at Which Radionuclides in Spent Nuclear Fuel Are Released from the Engineered Barrier Subsystem Through the Oxidation and Dissolution of Spent Nuclear Fuel (NRC, 2001)

- Container Life and Source Term: Subissue 4—The Rate at Which Radionuclides in High-level Waste Glass Are Leached and Released from the Engineered Barrier Subsystem (NRC, 2001)

- Container Life and Source Term: Subissue 5—The Effect of In-package Criticality on Waste Package and Engineered Barrier Subsystem Performance (NRC, 2001)

- Container Life and Source Term: Subissue 6—The Effects of Alternate Engineered Barrier Subsystem Design Features on Container Lifetime and Radionuclide Release from the Engineered Barrier Subsystem (NRC, 2001)

- Evolution of the Near-Field Environment: Subissue 3—Effects of Coupled Thermal-hydrologic-chemical Processes on the Chemical Environment for Radionuclide Release (NRC, 2000a)

- Evolution of the Near-Field Environment: Subissue 4—Effects of Coupled Thermal-hydrologic-chemical Processes on Radionuclide Transport Through Engineered and Natural Barriers (NRC, 2000a)

- Evolution of the Near-Field Environment: Subissue 5—Effects of Coupled Thermal-hydrologic-chemical Processes on Potential Nuclear Criticality in the near Field (NRC, 2000a)

- Total System Performance Assessment and Integration: Subissue 1—System Description and Demonstration of Multiple Barriers (NRC, 2000b)

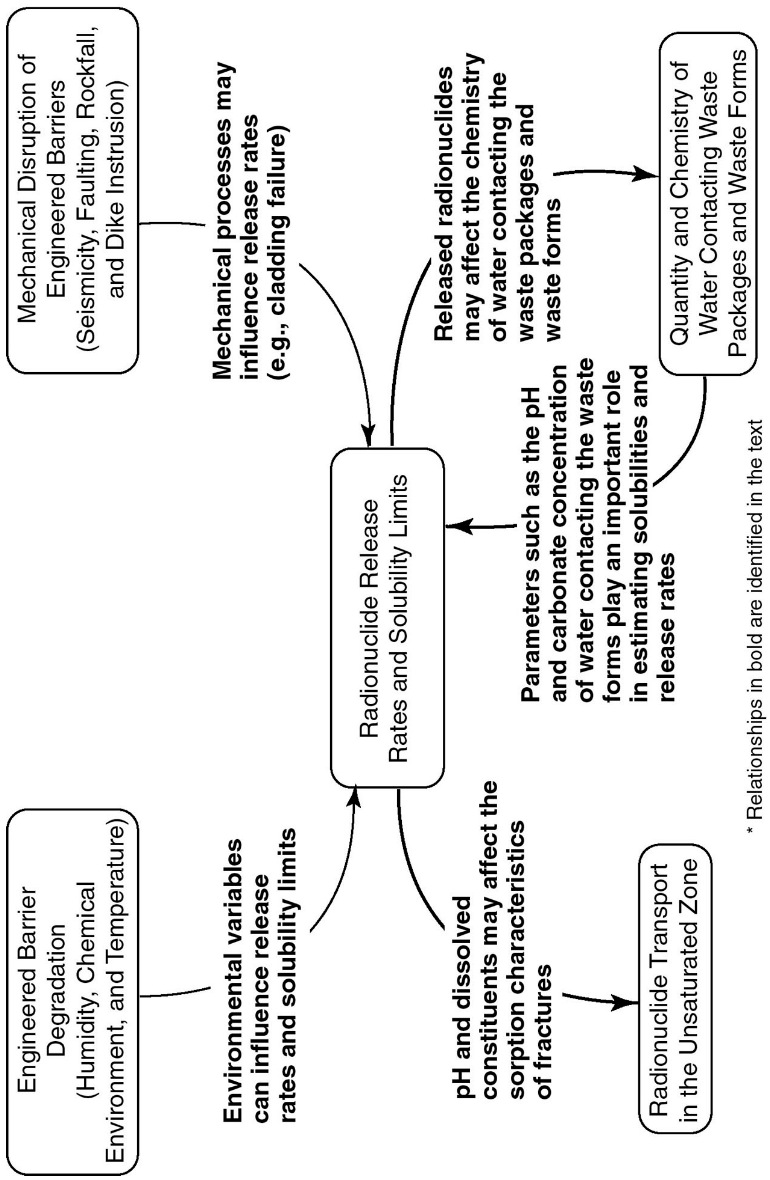

Figure 3.3.4-1. Diagram Illustrating the Relationship Between Radionuclide Release Rates and Solubility Limits and Other Integrated Subissues

- Total System Performance Assessment and Integration: Subissue 2—Scenario Analysis and Event Probability (NRC, 2000b)
- Total System Performance Assessment and Integration: Subissue 3—Model Abstraction (NRC, 2000b)
- Total System Performance Assessment and Integration: Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards (NRC, 2000b)

The key technical issue subissues formed the bases for previous versions of the issue resolution status reports and also were the bases for technical exchanges with DOE where agreements were reached on the additional information DOE needed to provide to resolve the subissue. The resolution status of each integrated subissue is based on the resolution status of each of the contributing key technical issue subissues. Discussions of issue resolution pertaining to the subissues on nuclear criticality are presented in Sections 3.3.1 and 3.3.7 and are not repeated here. The subsequent sections incorporate applicable portions of these key technical issue subissues, however, no effort was made to explicitly identify each subissue.

3.3.4.3 Importance to Postclosure Performance

The NRC review was to determine how this integrated subissue is related to the DOE repository safety strategy. The importance of radionuclide release rates and solubility limits to repository performance at Yucca Mountain is recognized by DOE. In CRWMS M&O (2000a), limited release of radionuclides from the engineered barriers is identified as one of five system attributes most important for predicting the performance of engineered and natural barriers. DOE considered the waste form itself, such as the irradiated uranium oxide pellets or the high-level waste glass, as one of the barriers to the release of radionuclides. DOE believed the concentration limits of radionuclides in water was another factor that constrained radionuclide release. For example, many radionuclides are sufficiently insoluble that they are not mobilized even if the waste form degrades. The transport behavior of radionuclides in the waste package and the engineered barriers outside the waste package also places constraints on radionuclide release. For limited flow conditions, DOE believes that radionuclide transport is limited by diffusion out of the waste package, a process that would be affected by the waste-generated heat that elevates temperatures and removes moisture. The invert material below the waste package could also limit the migration of radionuclides in the engineered barrier subsystem.

DOE considered radionuclide concentration limits in water as one of eight principal factors of the postclosure safety case in CRWMS M&O (2000a). This factor includes the limits for both dissolved radionuclides and those associated with colloidal suspensions. Other factors identified by DOE for the postclosure safety case, though given lower importance, include cladding performance and waste form performance. Cladding performance pertains to the role of cladding in limiting water contact and subsequent dissolution of the spent nuclear fuel waste form. Waste form performance relates to the rate of mobilization of radionuclides caused by degradation of the waste form itself (e.g., the irradiated uranium oxide matrix or high-level waste glass waste form).

3.3.4.4 Technical Basis

Radionuclide release from the engineered barrier subsystem will depend on several processes: (i) contact of water with the waste form, (ii) dissolution of the waste form, (iii) solubility limit of radionuclides, (iv) transport in water, and (v) interaction with engineered barrier materials. The waste form will begin to degrade once it comes into contact with air, water vapor, or water. Transport of radionuclides away from the waste form to the geosphere, however, generally requires a water pathway. In this regard, integrity of the cladding as an additional metallic barrier is an important factor considered by DOE in delaying water contact with the fuel matrix and degradation of commercial spent nuclear fuel. Radionuclides would be released from the waste form to the water within the waste package at a rate controlled by the (i) rate of waste form degradation (i.e., congruent dissolution), (ii) rate of dissolution of secondary minerals into which the radionuclides have become incorporated (e.g., schoepite or uranyl-hydrate), or (iii) solubility of the radionuclides themselves. Rates of dissolution vary for the different waste forms (e.g., spent nuclear fuel versus high-level waste glass). The rate of water flow through the waste package and the concentration of radionuclides in waste package waters ultimately control the release rate from the waste package (although molecular diffusion might be relatively important in a situation where flow rates are small). The solubility of radionuclide-bearing minerals could limit radionuclide concentrations in waste package water if the saturation index of the radionuclide-bearing mineral is positive. Colloid formation, especially from degradation of the high-level waste glass, however, is a potential process that could result in radionuclide concentration [(dissolved + colloid load)/volume] higher than the solubility limit. Once radionuclides are released from the waste package into the waste emplacement drifts, interaction with other engineered components could affect the release of radionuclides into the geosphere.

Near-field, coupled thermal-hydrological-mechanical-chemical processes will affect the environment for radionuclide release from the engineered barrier subsystem. Composition of the water entering the waste package will evolve as a function of time as a result of thermal-hydrological-chemical processes. As water interacts with the materials inside the waste package, the water chemistry will change. The dissolution rates of the waste form and engineered materials and the precipitation rates of alteration minerals are functions of temperature. In addition, as the materials degrade and alteration minerals are formed, the amount of water that can enter or exit the degraded waste package may change. The degradation rate of the Zircaloy cladding that surrounds the spent nuclear fuel and the dissolution rates of both the spent nuclear fuel and glass waste forms are functions of water chemistry. Other engineered materials in the emplacement drifts, including backfill, if present, and the drift invert, will be affected by coupled thermal-hydrological-mechanical-chemical processes. The coupled thermal-hydrological-mechanical-chemical processes could affect both hydraulic properties of the flow path from the waste package into the geosphere and the sorptive properties of the engineered materials. In addition, coupled thermal-hydrological-mechanical-chemical processes could affect the potential of near-field criticality.

Several factors need to be considered in abstractions of radionuclide release rates and solubility limits. In the specific case of spent nuclear fuel degradation, important factors are (i) spent nuclear fuel types, (ii) radionuclide inventory and distribution in the fuel, (iii) cladding performance, (iv) dry oxidation of the spent nuclear fuel and its effects on subsequent performance in an aqueous environment, (v) dissolution in an aqueous environment, (vi) solubility of radionuclides, (vii) secondary mineral formation and coprecipitation, (viii) formation of colloids, and (ix) conceptual models for release from waste packages. In the abstraction of glass waste form degradation, several factors are important: (i) high-level waste glass dissolution processes, (ii) formation of secondary minerals, (iii) effects of colloids and microbes, and (iv) conceptual models for release from the waste packages. Finally, the abstraction of the release of radionuclides from the waste package into the geosphere must consider (i) the hydrologic, chemical, and sorptive characteristics of engineered materials beneath the waste packages, such as backfill and invert, and (ii) the changes in the sorptive and hydraulic characteristics of engineered materials beneath the waste packages caused by coupled thermal-hydrological-mechanical-chemical processes.

The release rate of uranium and other species from breached waste packages containing spent nuclear fuel is controlled by a series of processes, such as the flux of water and oxidants, oxidative dissolution of spent nuclear fuel, secondary uranyl mineral precipitation, uranyl mineral dissolution or transformation, and transport of radionuclides, and is affected by the condition of the fuel cladding. The waste dissolution rate and elemental solubilities are key technical components affecting total system performance assessment (Electric Power Research Institute, 1998; DOE, 1998). The models used to describe waste form dissolution and the extent to which cladding can protect the spent nuclear fuel from contact with water are important determinants of total system performance assessment (Jarzemba, et al., 1999; Mohanty, et al., 1999; DOE, 1998). For example, four different spent nuclear fuel dissolution models, based on different assumptions of the chemical composition of the water contacting the waste form and the presence or absence of secondary uranium minerals, predict doses at 10,000 years that vary by one order of magnitude or more (Mohanty, et al., 1999).

The release of radionuclides from the waste package and engineered barriers is dependent on the concentration of radionuclides contained in the water of breached waste packages. Radionuclide release into water contacting the waste forms is, in turn, dependent on either the solubility of the individual radionuclide or the solubility of the waste matrix. In the absence of colloids, radionuclide solubilities represent the upper limit for individual radionuclide concentrations in the in-package water and depend on the physical and chemical conditions in the near-field environment.

A typical approach to analyze the radionuclide release rates and solubility limits in total system performance assessments is as follows. Waste form leach rate, combined with the amount of water in contact with the waste form, determines the fraction of radionuclide inventory released to waste package waters (NRC, 2001). If releases of radionuclides to waste package waters result in concentrations greater than the solubility limits, the radionuclide concentrations are set equal to the solubility limits (NRC, 2001). In this manner, both radionuclide solubilities and the waste form leach rate contribute to estimates of repository performance.

Total system performance assessment models can use a bathtub model, where a volume of water accumulates within a failed waste package, or a flow-through model, where water does not collect in the waste package (Mohanty, et al., 2000). Advective and diffusive releases from the waste package are estimated; both require estimation of time-dependent radionuclide concentrations in the water inside the waste package. In advective release, the rate at which water exits the waste package is multiplied by the radionuclide concentration to obtain a release rate for radionuclides from the waste package. In diffusive release, the concentration of radionuclides in waste package waters is used to estimate the concentration gradient necessary for calculating the diffusive flux of radionuclides from the waste package. Expressions for the dissolution rate of radionuclides in the waste form are used to estimate time-dependent radionuclide concentrations inside a breached waste package, Then, a mass balance is performed for the radionuclide concentration in the waste package water. The total release rate of radionuclides of higher solubility to waste package waters is the dissolution rate multiplied by the radionuclide inventory in the waste packages.

Radionuclides exiting the waste package will travel through the material that supports the waste package and lines the floor of the emplacement drifts (NRC, 2000a). These materials could sorb the radionuclides and decrease their release rate from the engineered barrier subsystem depending on whether matrix flow or fracture flow occurs through the materials (NRC, 2000a). The physical properties and sorptive capabilities of these materials may change as a result of coupled thermal-hydrological-mechanical-chemical processes. On the other hand, the sorption capabilities of the materials could increase the potential for near-field criticality.

Radionuclide solubilities are strongly dependent on the in-package environment. The chemistry of water contacting the waste form affects the oxidation state, the solubility, and the release rate of the radionuclides. In an oxidizing environment such as the Yucca Mountain proposed repository setting, uranium in the spent nuclear fuel may ultimately exist as U_3O_8 or UO_3, which has markedly different solubilities from UO_2. Similarly, technetium is soluble during oxidizing conditions but insoluble during reducing conditions. Other parameters dictated by the in-package chemistry also affect waste form degradation rates and radionuclide solubilities. For example, equations for the dissolution rate of spent nuclear fuel could have terms dependent on pH, temperature, carbonate, silica, and calcium concentrations.

Secondary minerals could precipitate on or near the spent nuclear fuel and mitigate radionuclide release by incorporating radionuclides into their structure or by reducing the spent nuclear fuel surface in contact with the water. For example, drip tests on spent nuclear fuel conducted at Argonne National Laboratory indicate that key nuclides, such as neptunium and cesium, can be concentrated in secondary mineral phases at the surface of the spent nuclear fuel (NRC, 2001). Periodic spallation of the secondary mineral layer, however, could expose a fresh surface of spent nuclear fuel for further dissolution.

Another aspect of radionuclide release is the possibility of precipitation of fissile isotopes after their release from the waste forms and waste packages, within the invert or on the drift floor, which could increase the potential of near-field criticality. More discussion on this subject is provided in Section 3.3.4.4.10.

NRC developed a plan (2002) consistent with the acceptance criteria and review methods found in previous issue resolution status reports. A review of DOE approaches for including radionuclide release and solubility limits in total system performance assessment abstractions is provided in the following subsections. Several DOE abstractions pertain to the Integrated Subissue on Radionuclide Release Rates and Solubility Limits. For clarity, the discussions in the following subsections are organized according to the specific topic of the DOE abstractions:

(i) Radionuclide Inventory, (ii) In-Package Chemistry, (iii) Degradation of Cladding on Commercial Spent Nuclear Fuel, (iv) Commercial Spent Nuclear Fuel Dissolution, (v) DOE Spent Nuclear Fuel Dissolution, (vi) High-Level Waste Glass Dissolution, (vii) Radionuclide Solubility, (viii) Colloidal Release, and (ix) Engineered Barrier Subsystem Flow and Transport. Staff comments for each topic are organized according to the five generic acceptance criteria identified in Section 1.5: (i) System Description and Model Integration Are Adequate, (ii) Data Are Sufficient for Model Justification, (iii) Data Uncertainty Is Characterized and Propagated Through the Model Abstraction, (iv) Model Uncertainty Is Characterized and Propagated Through the Model Abstraction, and (v) Model Abstraction Output Is Supported by Objective Comparisons.

3.3.4.4.1 Radionuclide Inventory

Radionuclide inventory is used for three purposes: (i) in a radionuclide screening evaluation to determine which radionuclides should be tracked for the total system performance assessment calculations, (ii) as input to the total system performance assessment calculations to determine the fuel heat generation rates and the radionuclide release rates, and (iii) in an evaluation to determine potential reconcentration of fissile materials that could form a critical mass. DOE accounts for the radionuclide inventories in commercial spent nuclear fuel assemblies, DOE spent nuclear fuel canisters, and defense high-level waste canisters (CRWMS M&O, 2000b). DOE derived representative radionuclide inventories, one for commercial spent nuclear fuel waste packages and another for codisposal waste packages, which contain both DOE spent nuclear fuel and high-level waste. The representative waste package inventories were developed based on a weighted average of the radionuclide inventories for all potential waste package loadings.

Radionuclide screening was performed to ensure all radionuclides that could contribute significantly to the dose were tracked in the total system performance assessment. This screening was performed by summing the product of the inventory of a radionuclide in a representative waste package and the inhalation or ingestion dose conversion factor for all radionuclides. The radionuclides that composed the upper 95 percent of this sum were screened into the analysis. This screening process was conducted at times between 100 and 10,000 years for the Total System Performance Assessment–Site Recommendation (CRWMS M&O, 2000c) analyses and up to 1,000,000 years for the final environmental impact statement analyses. Also, the process was repeated for subgroups of radionuclides based on their solubility and transport properties. Radionuclides were divided into two solubility groups (soluble and insoluble) and three transport groups (highly sorbing, mildly sorbing, and nonsorbing). This categorization identifies the important radionuclides for the nominal release scenario, the igneous activity scenario, and the human intrusion scenario.

The staff review regarding the DOE abstraction of radionuclide inventory follows.

3.3.4.4.1.1 System Description and Model Integration Are Adequate

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the effects of radionuclide inventory on radionuclide release rates and solubility limits with respect to system description and model integration.

The approach appears to account for all waste types that will be emplaced in the repository, with reasonable bases for the radionuclide source term in the various fuel types, and seems complete in this regard. Projections of radionuclide inventory include considering the current trend to increase burnup of commercial fuel in the nuclear industry.

3.3.4.4.1.2 Data Are Sufficient for Model Justification

Sufficient data are available on the inventory of radionuclides in the waste to support the numerical values used in the calculations. Fuel assembly characteristics such as burnup, enrichment, and cooling time for commercial spent nuclear fuel are derived from a 1995 data submittal from the commercial utilities that supplied historical information about reactor assembly discharges through December 1995 and forecasts about future discharges. These data were used to derive representative radionuclide inventories for commercial spent nuclear fuel waste packages. Inventory projections for DOE spent nuclear fuel were derived from ORIGEN–2 (Oak Ridge National Laboratory, 1983) runs of representative fuel types (CRWMS M&O, 1998a). Inventory projections for high-level waste are taken from the best available information for each vitrification site (DOE, 1999). With respect to sufficient data for model justification, no information (beyond that currently available) likely will be required for regulatory decision making at the time of a potential license application.

3.3.4.4.1.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

DOE uses values for radionuclide inventories that reasonably account for uncertainty and variability. No additional information is needed regarding the characterization and propagation of data uncertainty through the abstraction of waste inventory.

3.3.4.4.1.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

To generate radionuclide inventories, DOE uses models that are reasonable. No additional information is needed regarding the characterization and propagation of model uncertainty through the abstraction of the waste inventory.

3.3.4.4.1.5 Model Abstraction Output Is Supported by Objective Comparisons

The modeling of the radionuclide inventory by DOE in its total system performance assessment analyses for every type of DOE spent nuclear fuel is reasonable. No additional information is needed regarding model abstraction output that is supported by objective comparisons for the abstraction of DOE spent nuclear fuel inventory.

3.3.4.4.2 In-Package Chemistry

The estimation of the in-package chemical environment is integral to the calculations of commercial spent nuclear fuel, DOE spent nuclear fuel, high-level waste glass degradation, radionuclide solubility, and colloid availability and stability. The in-package chemistry component in the Total System Performance Assessment–Site Recommendation couples the seepage rate of water into the waste package, the degradation rate of the waste form and waste package components, and the cladding coverage of commercial spent nuclear fuel to the resulting effluent chemistry. The water chemistry parameters used in the DOE abstraction include pH, Eh, ionic strength, and total aqueous carbonate, fluoride, and chloride concentrations. DOE made two assumptions in its abstraction of the in-package chemistry. These assumptions are that the aqueous solution fills all the voids in the waste package and that solutions that drip into the package will have the composition of J–13 Well water (CRWMS M&O, 2000d). Other drip water compositions, such as evaporated J–13 Well and Drift Scale Test waters, were considered in the revised analysis and model report (CRWMS M&O, 2001a). For development of the in-package chemistry abstraction, the drip rate is assumed to range from 1.5 to 150 L/yr [0.4 to 40 gal/yr] (CRWMS M&O, 2000d) or from 0.15 to 15 L/yr [0.04 to 4 gal/yr] (CRWMS M&O, 2001a). Dripping water is assumed to enter and exit the waste package at the same rate and not interact to any significant degree with the waste package walls as it enters/exits the waste package. The interaction of water with the waste form and several waste package components, however, is considered (CRWMS M&O, 2000d; 2001a).

Two representative waste packages were modeled: a commercial spent nuclear fuel package and a DOE-owned spent nuclear fuel/high-level waste glass codisposal package (CRWMS M&O, 2000d). Commercial spent nuclear fuel waste packages are assumed to be made of several reactive components: aluminum alloy, 304L low-carbon stainless steel, A516 carbon steel, borated and nonborated Type 316 stainless steel containing $GdPO_4$, and zirconium-clad fuel rods. Commercial spent nuclear fuel is primarily UO_2. No interaction between the Zircaloy cladding and the internal environment is assumed, although sensitivity analyses include the percentage of fuel area exposed by breached cladding. Codisposal wastes compose a DOE spent nuclear fuel canister surrounded by five containers of high-level waste glass. The codisposal waste package was assumed to have the properties of a fast flux test facility waste package with six reactive components: A516 carbon steel, Type 316 stainless steel (with and without $GdPO_4$), 304L low-carbon steel, high-level waste glass, mixed oxide fuel (made of plutonium, uranium, and neptunium oxide), and UO_2 fuel.

In the in-package chemistry model, water is assumed to fill the void volume, and the waste package internal components are lumped into equivalent masses per unit volume for calculating the reaction products. EQ3/6 is used to calculate the time evolution of solution composition as a result of these interactions (CRWMS M&O, 2000d). The specific partial pressures of CO_2 and O_2 of the repository atmosphere are set to $10^{-3.0}$ and $10^{-0.7}$ atmosphere. A range of degradation rates was used for each component of the waste package.

Results of the EQ3/6 calculations indicate the reaction of waste package components with incoming fluids results in dramatic changes in solution chemistry. The pH decreases inside the waste package because of dissolution of stainless steel components, specifically because of the chromium oxidation to Cr^{6+} species. The pH increases because of the dissolution of the uranium oxide fuels, aluminum alloy, and high-level waste glass. Solution pH represents a dynamic balance between proton-producing and proton-consuming reactions. Relatively high rates of the proton-producing reactions lead to transiently low pH, whereas relatively high rates of the proton-producing reactions cause solution pH to be transiently high. Solution ionic strength for codisposal waste package effluents varied between 0.003 and ~5.8 M, however, ionic strengths of commercial spent nuclear fuel waste package effluents never exceeded 1.7 M.

Direct use of a complex code such as EQ3/6 within the Total System Performance Assessment–Site Recommendation analysis calculations was not practical because of computational constraints. Thus, DOE used abstractions of in-package processes based on a series of multiple linear regression analyses of the output from the EQ3/6 simulations (CRWMS M&O, 2000e, 2001b). Three-dimensional response surfaces establishing the pH boundary limits were determined using the extreme (high and low) values of the waste package corrosion rate. EQ3/6 simulation results were plotted in three-dimensional space, and the pH response surfaces were modeled as a planar surface. Data regression was performed using the equation of a plane: $z = y_o + ax + by$. Processes at times less than 1,000 years after breach of the waste package were abstracted separately from those at greater than 1,000 years postbreach. For each time phase and each package type (commercial spent nuclear fuel and codisposal), one pH surface was generated for a low waste package corrosion rate scenario and another pH surface was generated for a high waste package corrosion rate scenario. These surfaces constitute the boundaries of the range of in-package pH values. The waste package corrosion rates used in the Total System Performance Assessment–Site Recommendation analysis are randomly sampled from the range bounded by these low and high values (CRWMS M&O, 2000e, 2001b).

In-package chemistry parameters included in the Total System Performance Assessment–Site Recommendation abstraction are pH, Eh, ionic strength, total aqueous carbonate concentration, chloride concentration, and fluoride concentration. The pH is the most important in-package parameter. Thus, the time discretization used by DOE for all abstracted parameters was based on changes in pH. Both total carbonate and Eh are pH dependent and may be calculated directly from the abstracted pH value. The fluoride concentration and ionic strength were given a range of values to be sampled in the Total System Performance Assessment–Site Recommendation analysis. The chloride concentration and the O_2 and CO_2 fugacities were set to constant values.

The staff review regarding the DOE abstraction of in-package chemistry follows.

3.3.4.4.2.1 System Description and Model Integration Are Adequate

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the effects of in-package chemistry on radionuclide release rates and solubility limits with respect to system description and model integration.

The process model report (CRWMS M&O, 2000b) and analysis and model reports (CRWMS M&O, 2000d,e, 2001b) provide sufficient descriptions of the process-level and mathematical models used in estimating the in-package chemical environment and how the in-package chemistry abstraction is integrated with other abstractions in the total system performance assessment analyses. Assumptions are appropriate, clearly stated, and used consistently. In general, important physical phenomena and couplings are adequately incorporated or bounded.

DOE developed an in-package chemistry abstraction for the Total System Performance Assessment–Site Recommendation (CRWMS M&O, 2000t) that used different response surfaces depending on the time from waste package breach. An early abstraction was used for conditions when the average time since the first waste package failure was less than 1,000 years. A late abstraction was used for conditions when the average time since the first waste package failure was greater than 1,000 years. The staff review found that using the aforementioned methodology for implementation in the total system performance assessment was inconsistent with supporting documentation and was likely to underestimate projected doses.

Assuming corrosion is the only mechanism for degradation of the waste packages, breach of waste packages during the thermal period will not be significant, and high-temperature phenomena need not be considered in determining the initial conditions for the in-package chemistry model. The potentials for juvenile failure and for mechanical disruption of waste packages exist, however, and DOE will need to demonstrate that the probability of these other mechanisms is not high enough to warrant evaluating the consequences of these other processes. DOE agreed[1] to update the in-package chemistry model to account for scenarios, their associated uncertainties, and implementation in the total system performance assessment model.

3.3.4.4.2.2 Data Are Sufficient for Model Justification

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the effects of in-package chemistry on radionuclide release rates and solubility limits with respect to data sufficiency and model justification.

Sufficient data are available about the characteristics of the near-field environment and the engineered materials to establish initial and boundary conditions for conceptual models and simulations of coupled processes affecting the in-package chemical environment. Insufficient technical justification was provided by DOE for the assumed corrosion rates of waste package components, however, and the likely modes of corrosion that account for the rates were not identified. For example, the dissolution rate assumed for Type 316 stainless steel and the

[1]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

borated stainless steel is one order of magnitude lower than measured experimentally (Kirchheim, et al., 1989). Additionally, the lower dissolution rate assumed for the borated stainless steel compared to Type 316 stainless steel is counterintuitive. The presence of boron, in the form of second phase particles of borides, would be expected to result in a higher corrosion rate, especially in local zones around the boride particles. DOE agreed[2] to address concerns regarding the effect of corrosion rates on in-package chemistry.

3.3.4.4.2.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

DOE acknowledges there are large uncertainties in the thermodynamic and kinetic data and in the simplified approach used for the abstraction of in-package chemistry. DOE accounted for some uncertainties by varying the model parameters. For example, uncertainty in the dissolution rates of waste package materials is dealt with using high and low values. Drip rates onto the waste package ranged from 1.5 to 150 L/yr [0.4 to 40 gal/yr] (CRWMS M&O, 2000d) or 0.15 to 15 L/yr [0.04 to 4 gal/yr] (CRWMS M&O, 2001a). The composition of water entering the waste package was varied by using compositions similar to that of J–13 Well water, evaporated (50×) J–13 Well water, and Drift Scale Test water (CRWMS M&O, 2001a). Staff consider acceptable this approach to characterizing and propagating data uncertainty through the model abstraction.

3.3.4.4.2.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the effects of in-package chemistry on radionuclide release rates and solubility limits with respect to model uncertainty.

In NRC (2000a, 2001), staff commented that the DOE assumption the waste package components can be lumped into a single mass for estimating the in-package chemistry may lead to highly nonconservative estimates of pH values and asked DOE for further justification of its assumption. At issue is the effect of potential spatial variation in chemistry in the waste package leading to local pH values considerably more acidic than calculated, based on a volume-averaged mass. The pH in crevices and other tight spaces differs from bulk pH values because the dissolution reactions become spatially separated from the reduction reactions. For example, Cavanaugh, et al. (1983) reported that the pH values in corroding cavities of stainless steels range between 0 and 2, with the pH increasing with increasing molybdenum and decreasing chromium concentrations. The pH in crevices of aluminum alloys can be either acidic (pH 4) or alkaline (pH 9), depending on the initial pH and surface conditions. Therefore, the pH generated by localized dissolution of aluminum most likely would be influenced by the pH resulting from the corrosion of other components. Because the internal geometry of the

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[2]Schlueter, J. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000).” Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

waste package will have many tightly packed regions, local pH may affect the dissolution rate of spent nuclear fuel locally and, hence, the local release rate of highly soluble radionuclides such as Tc-99. Staff recommended alternative models that consider electrochemical reactions coupled to transport processes should be considered by DOE.

In its revised analysis and model report (CRWMS M&O, 2001a), DOE provided additional arguments to justify its assumption that all waste package components are in communication with fluids that completely fill the waste package. DOE argues that, although bypassing specific waste package components might lead to anomalous package fluid compositions, it also would lead to limited reaction with those components and possibly not all the reaction steps necessary to cause radionuclide release. Thus, DOE states that homogeneous flow, as used in its abstraction of in-package chemistry, is conservative because it involves complete reaction and maximal release of radionuclides. Staff consider the previous argument acceptable with respect to the potential effect of bypassing of flow inside the waste package. The potential formation of locally aggressive environments in crevices and tight spaces that could enhance waste form degradation and radionuclide solubility, however, has not been addressed sufficiently by DOE. DOE agreed[3] to provide analyses justifying the use of bulk chemistry as opposed to local chemistry for solubility and waste form degradation models in an update of the in-package chemistry analysis and model report (CRWMS M&O, 2001a).

3.3.4.4.2.5 Model Abstraction Output Is Supported by Objective Comparisons

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the effects of in-package chemistry on radionuclide release rates and solubility limits with respect to model abstraction output being supported by objective comparisons.

The EQ3/6 predictions of in-package chemistry have not been verified by empirical observations. DOE recognized the difficulties in modeling the detailed effects of geometry and corrosion reactions on the in-package chemistry. DOE did not consider, however, the potential local acidification resulting from spatially separated anodic and cathodic regions that may enhance the dissolution rate of the waste form and the solubility of radionuclides. Although there will be issues regarding the ability of experiments to adequately represent all the complexities in a waste package, experiments to simulate certain aspects of waste package geometry and materials may aid in gaining confidence in the model abstractions.

In the revised analysis and model report (CRWMS M&O, 2001a), DOE states that validation of the in-package chemistry model is incomplete. Planned DOE validation exercises will involve using EQ3/6 to model some combination of the following processes: (i) alteration observed during drip tests performed at Argonne National Laboratory; (ii) formation of ore deposits that might constitute natural analogues; and (iii) glass, mineral, and steel corrosion measurements

[3]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

performed in the laboratory. The planned DOE validation exercises are expected to address staff concerns.[4]

3.3.4.4.3 Degradation of Cladding on Commercial Spent Nuclear Fuel

DOE considered the most likely forms of degradation that may affect the integrity of the commercial spent nuclear fuel cladding during disposal conditions. DOE developed a model to evaluate cladding degradation as part of the waste form degradation model (CRWMS M&O, 2000b) to determine the rate at which the commercial spent nuclear fuel matrix is exposed to the in-package environment. This cladding degradation model represents a significant improvement with respect to that presented in the DOE (1998). The degradation of the commercial spent nuclear fuel cladding is assumed to occur in two stages (CRWMS M&O, 2000b,f). The first stage of degradation corresponds to rod failure as a result of cladding perforation. The second stage involves progressive exposure of the spent nuclear fuel matrix as a result of splitting (unzipping) of the cladding through oxidation of the irradiated UO_2 pellets either by air and moisture or by an aqueous environment.

Cladding perforation may occur before or after waste package emplacement. DOE evaluated the initial condition of the cladding and the percentage of rods perforated at the time of disposal, taking into account data obtained from reactor operation, pool storage, dry storage, and transportation, including fuel handling (CRWMS M&O, 2000g). A distribution of initially perforated Zircaloy fuel rods, expressed as a complementary cumulative distribution function, was developed from the available data. All the commercial spent nuclear fuel clad with stainless steel instead of Zircaloy (estimated to be approximately 1.1 percent of the total) was assumed initially perforated (CRWMS M&O, 2000g).

DOE used an empirical creep model developed by Matsuo (1987) to define the creep damage of the Zircaloy cladding prior to disposal. DOE computed the creep strain as a function of initial rod stress for cladding in dry storage alone and for dry storage plus transportation, using an assumed temperature history profile representative of dry storage and transportation conditions (CRWMS M&O, 2000g). DOE concluded that little creep occurs for rod stresses less than 80 MPa [11.6 ksi]. It is assumed that most creep occurs during dry storage, whereas only a small amount of creep occurs during transportation. The amount of creep strain accumulated is expected to be less than 1 percent at initial stresses less than 90 MPa [13.0 ksi] at 27 °C [81 °F]. A creep failure strain of 3.3 percent was established based on experimental results of tensile and creep tests. This creep failure strain led to a prediction of approximately 0.24 percent of failed rods by creep in dry storage and transportation, compared with an actual failure rate of 0.45 percent (CRWMS M&O, 2000g).

Cladding perforation after waste package emplacement is assumed caused by creep, stress corrosion cracking, mechanical failure (due through seismic events), and localized corrosion (CRWMS M&O, 2000f). To evaluate the possibility of creep and stress corrosion cracking for

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[4]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

disposal conditions, DOE estimated the temperature history of the cladding during storage and transportation and the evolution of temperature after waste package emplacement, as well as estimating the distribution of internal pressure and corresponding hoop stresses (CRWMS M&O, 2000f,g). The Murty's creep-versus-strain correlation was selected to evaluate creep rupture on the basis of experimental data for unirradiated cladding. It is claimed that the Murty's creep model is more accurate than other models because it includes Coble creep, a type of creep process important at low stresses and temperatures. The approach is considered conservative because irradiated cladding has a creep rate significantly lower than that of the unirradiated material. Nevertheless, the criterion for creep failure strain was developed based on data for irradiated cladding and is conservative with respect to other creep failure criteria. Based on distribution of hoop stresses, an abstraction was developed to provide the fraction of rods that failed by creep as a function of the peak waste package surface temperature.

Stress corrosion cracking was also a possibility, based on the calculated distribution of hoop stresses. The causative species for stress corrosion cracking of commercial spent nuclear fuel cladding is considered to be iodine, found free as a fission product in the pellet-cladding gap (CRWMS M&O, 2000f). Although the iodine concentration is asserted to be negligible, conservatively it is assumed to be above a certain critical concentration required to promote iodine-stress corrosion cracking. For stress corrosion cracking to occur, a critical stress level of 180 MPa [26.1 ksi] is selected as a threshold stress. This value is relatively high and can be attained by no more than a few rods.

Localized corrosion is also considered as a process leading to the perforation of the commercial spent nuclear fuel cladding (CRWMS M&O, 2000f). Fluoride is assumed the anionic species promoting accelerated corrosion on a relatively small area of cladding approximately 10 mm [0.39 in] in rod length. The fraction of fuel cladding surface on different fuel rods inside the same waste package is considered proportional to the volume of water entering the waste package in a flow-through scenario. This approach is considered a bounding analysis because it implicitly assumed 100-percent efficiency in the chemical reaction of fluoride with Zircaloy.

The DOE analysis of delayed hydride cracking is based on a fracture mechanics approach in which the cladding stress and crack depth were used to compute the model stress intensity factor of preexisting cracks in the cladding (CRWMS M&O, 2000g). The stress intensity factor, K_I, was taken to be the driving force for delayed hydride cracking and compared against the threshold stress intensity factor, K_{IH}. Failure by delayed hydride cracking is considered not to occur when K_I is lower than K_{IH}, but failure can occur when K_I is higher than K_{IH}. The DOE extensive review of the literature DOE (CRWMS M&O, 2000g) indicated that the minimum reported value of K_{IH} for zirconium cladding is 5 MPa•m$^{1/2}$ [4.55 ksi • in$^{1/2}$]. DOE analyzed delayed hydride cracking of existing cracks using distributed stresses and crack sizes (CRWMS M&O, 2000g). It was concluded that delayed hydride cracking can be ruled out as a possible mechanism for cladding failure of spent nuclear fuel in the proposed repository because the computed mean K_I value, 0.0016–2.7 MPa•m$^{1/2}$ [0.0015–2.5•in$^{1/2}$], was too low. DOE screened out failures of cladding by hydrogen or hydride embrittlement, delayed hydride cracking, and hydride reorientation as possible events in the repository (CRWMS M&O, 2000h). DOE considered stresses and temperatures of the cladding as too low for hydride reorientation to occur, and the cladding material would maintain sufficient strength that cladding failure would be unlikely, even if hydride reorientation did occur.

The remaining process that could lead to cladding perforation is mechanical failure caused by seismic events when the frequency of the events is on the order of 1×10^{-6} per/year. This type of event, which is considered in the DOE analysis as a disruptive event, perforates the cladding and initiates unzipping. Mechanical failure of the cladding as a result of rockfall is excluded from the model abstraction (CRWMS M&O, 2000h) using the screening argument that the waste package will remain intact for more than 10,000 years.

After cladding perforation, the inventory of radionuclides in the gap and in the grain boundaries of the irradiated fuel pellets is considered to experience fast release. The gap inventory of iodine and cesium is predicted to be released in proportion to the fission gas release fractions, while that in the grain boundaries is estimated from release experiments using intact and defective (i.e., with slits and holes) fuel rod samples. A cumulative distribution function for the fast release fraction of the radionuclides is used in the model abstraction.

Unzipping of the cladding during dry conditions is excluded from the model abstraction assuming the integrity of containers is maintained during the performance period (CRWMS M&O, 2000h). Only wet unzipping is assumed to occur. The time to unzip a fuel rod during wet conditions is estimated as a function of waste package temperature. Time also is a function of the in-package water chemistry, which, for this purpose, is defined by the pH, partial pressure of O_2, and carbonate concentration. Although DOE considered these criteria are conservative, and include the consideration of uncertainties, it argued the criteria are not as conservative as in previous total system performance assessments.

The staff review regarding the DOE abstraction of cladding degradation on commercial spent nuclear fuel follows.

3.3.4.4.3.1 System Description and Model Integration Are Adequate

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the effect of cladding degradation of commercial spent nuclear fuel on radionuclide release rates and solubility limits with respect to system description and model integration.

The system description and model integration for creep and mechanical failure are adequate. For mechanical failure, however, the abstraction is related to the evaluation of seismic events (see Section 3.2.2), and the exclusion of rockfall effects is related to the integrity of the waste package through the 10,000-year performance period (see Section 3.1.1). The system description and model integration used in the abstraction of stress corrosion cracking and localized corrosion are not sufficient because the abstraction does not consider the range of chemical conditions that may prevail in the in-package aqueous environment. DOE agreed[5] to establish a better technical basis for the abstracted in-package chemistry.

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[5]Schlueter, J. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000).” Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

Although it is a possible failure process (NRC, 2001), localized corrosion in the form of pitting promoted by chloride, DOE excluded (CRWMS M&O, 2000h) by assuming the (i) chloride concentration is lower than the minimum concentration required for pit initiation; (ii) concentrations of inhibiting anions such as nitrate, sulfate, and bicarbonate are sufficient to overcome the detrimental effect of chloride; and (iii) concentration of dissolved Fe^{3+} ions, considered to be the single species that may increase the corrosion potential of the cladding to more than the pitting potential, is assumed insufficient for the range of expected pH of the in-package water. Instead, DOE proposed accelerated corrosion by fluoride ions as the most plausible degradation process through a chemical reaction controlled by the volume of water entering the waste package in a flow-through scenario, the flow rate, and the concentration of fluoride in the water (CRWMS M&O, 2000f). The chloride concentration inside breached waste packages, however, has not been properly bounded in DOE analyses, and the presence of Fe^{3+} ions cannot be considered an absolute requirement because corrosion potentials higher than the pitting potential can be attained in the presence of other oxidizing species including radiolytic products such as H_2O_2. A detailed discussion, based mostly on data about commercial purity zirconium relevant to chemical processes and industry applications, has been provided in the analysis and model report devoted to localized corrosion (CRWMS M&O, 2000i) questioning the occurrence of pitting corrosion induced by chloride during repository conditions. It is claimed in the discussion that acidic pHs are not attained to maintain sufficient concentration of Fe^{3+} ions in solution. This analysis, however, contradicts screening arguments in several features, events, and processes (CRWMS M&O, 2000h) in which the existence of acidic conditions inside the waste packages is assumed to justify the screening arguments that acidic pHs may affect the occurrence of localized corrosion. DOE agreed[6] to address concerns of the effects of in-package chemistry on cladding degradation.

Stress corrosion cracking of Zircaloy cladding may occur in the presence of hoop stresses of sufficient magnitude for the same environmental and electrochemical conditions that promote pitting corrosion by chloride (NRC, 2001). As noted, instead of chloride, DOE considers iodine as the causative species for stress corrosion cracking (CRWMS M&O, 2000f). The possibility of stress corrosion cracking induced by iodine discussed in the process model report (CRWMS M&O, 2000b), however, does not appear so important because it is limited by the availability of iodine. The mechanism as such has been postulated as the cause of pellet cladding interaction failure for reactor operating conditions following steep power ramps, but it does not seem plausible for disposal conditions. The technical bases to support modeling of cladding degradation as a result of both the corrosion by fluoride and the internal stress corrosion cracking by iodine are limited (NRC, 2001). DOE agreed[7] to address concerns of the effects of in-package chemistry on cladding degradation.

In the process model report (CRWMS M&O, 2000b), the role of fluoride is emphasized as a species promoting accelerated corrosion in local areas, but insufficient technical basis is offered in CRWMS M&O (2000i). In addition, the analysis of the flow and volume of water contacting

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[6]Schlueter, J. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000)." Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[7]Ibid.

the fuel rods to evaluate the local attack by fluoride is limited and requires additional justification. Again, inconsistencies exist regarding evaluation of the in-package pH. A low pH is assumed for the attack by fluoride, whereas it is not taken into account to estimate the concentration in solution of Fe^{3+} ions that may promote the oxidizing conditions required for pitting corrosion in chloride solutions. DOE agreed[8] to address concerns of the effects of in-package chemistry on cladding degradation.

3.3.4.4.3.2 Data Are Sufficient for Model Justification

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the effect of cladding degradation of commercial spent nuclear fuel on radionuclide release rates and solubility limits with respect to sufficient data for model justification.

Currently, insufficient data have been presented to justify that accelerated corrosion by fluoride or internal stress corrosion cracking by iodine are the appropriate degradation processes that need to be included in the model abstraction for radionuclide release. DOE agreed[9] to address concerns of the effects of in-package chemistry on cladding degradation caused by localized corrosion and stress corrosion cracking.

Corrosion data, generated outside the Yucca Mountain program by Teledyne Wah Chang (a producer of zirconium alloys) and reported by Yau and Webster (1987), are presented in the analysis and model report (CRWMS M&O, 2000i) to support the localized corrosion failure model for Zircaloy-2 or -4 cladding. Most data provided are for commercial purity zirconium instead of Zircaloy. In the report, it is noted that the behavior of commercial purity zirconium (containing hafnium and lacking the Zircaloy alloying elements) is comparable to that of Zircaloy. Although an acceptable statement in general terms, there are no specific data provided for environments postulated to simulate the in-package water chemistry. Although data on localized corrosion by chloride anions are presented, it is claimed this process cannot occur because the pH is too high to maintain sufficient concentration of Fe^{3+} ions in solution, which implicitly assumes this cation is the single species able to increase the corrosion potential more than beyond the pitting potential. Instead, corrosion is assumed to be caused by fluoride anions only. Corrosion rate data from 24- to 72-hour tests in aqueous solutions containing fluoride and chloride were used to generate a parametric equation relating the corrosion rate to the concentration of these anionic species (CRWMS M&O, 2000i). The equation is not used in the model abstraction, however. In the analysis and model report (CRWMS M&O, 2000f), corrosion by fluoride to stoichiometrically form ZrF_4 is conservatively assumed to be determined by its concentration in the J–13 Well water, the volume of water entering the waste package, and the flow rate; however, the attack is confined to a small 1-cm [0.39-in] long cladding ring portion of the fuel rod.

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[8]Schlueter, J. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000)." Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[9]Ibid.

As noted in the analysis and model report (CRWMS M&O, 2000f), the model abstraction for stress corrosion cracking is based on the assumption that iodine as a fission product is the causative species. As a conservative approach, it is assumed that iodine concentration in the fuel matrix-cladding gap is higher than the threshold value of 5×10^{-6} g/cm^2 [7.1×10^{-8} lb/in^2] required for stress corrosion cracking. If the hoop stress is higher than 180 MPa [26.1 ksi], this form of internal stress corrosion cracking is assumed to occur. Although these values seem appropriate for evaluating iodine stress corrosion cracking and represent a lower bound, the data obtained for test conditions are not necessarily applicable to disposal conditions where stress corrosion cracking on the cladding outer surface may begin by other species present in the modified groundwater. In addition, an adequate technical basis should be provided for selection of the critical stress relevant to the environment in which external stress corrosion cracking may occur. DOE agreed[10] to address concerns of the effects of in-package chemistry and stress on cladding degradation caused by stress corrosion cracking.

In the assessment of hydride reorientation and delayed hydride cracking (CRWMS M&O, 2000j), the stress distribution reported for cladding corresponds to 27 °C [81 °F], which appeared to be the basis leading to the conclusion that stresses and temperatures in the cladding were too low to cause hydride reorientation. There is a concern that the proper cladding stress might not have been used in the analysis. For hydride reorientation, the relevant stress to consider is the cladding hoop stress at temperatures just below the solvus temperature, which is in the range 260–300 °C [500–572 °F], depending on the hydrogen content (Northwood and Kosasih, 1983). The peak cladding temperature for the design basis waste package was estimated to be 325 °C [617 °F] (CRWMS M&O, 2000j). The hydrogen solubility in Zircaloy-2 and -4 is approximately 90 ppm. Consequently, some of the circumferential hydrides in Zircaloy cladding would dissolve into the matrix and subsequently reorient and reprecipitate as radial hydrides for a tensile (hoop) stress when the cladding cools slowly in repository conditions below the solvus temperature. The DOE analysis of delayed hydride cracking is based on the properties of Zircaloys that contain circumferential hydrides, which would not be applicable if hydride reorientation occurs. The prediction of the lack of susceptibility to delayed hydride cracking based on a K_{IH} of 5 MPa•m$^{1/2}$ [4.55 ksi•in$^{1/2}$] might not be conservative if hydride reorientation occurs in the cladding. Thus, it is important to consider the distribution of cladding stresses and temperatures and their evolution following waste package emplacement in the repository. DOE agreed[11] to address concerns regarding hydrogen embrittlement as a mode of cladding degradation.

3.3.4.4.3.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the effect of cladding degradation of

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[10]Schlueter, J. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000)." Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[11]Ibid.

commercial spent nuclear fuel on radionuclide release rates and solubility limits with respect to the characterization and propagation of data uncertainty through the model abstraction.

Data uncertainty regarding stresses and temperatures of the cladding may affect the consideration of hydride reorientation and subsequent hydride embrittlement as potential cladding failure mechanisms that need to be included in the model abstraction for radionuclide release.

DOE considers that stresses and temperatures of the cladding are too low for hydride reorientation to occur and that the cladding material would maintain sufficient strength even if hydride reorientation occurred, hence, cladding failure would be unlikely (CRWMS M&O, 2000b,h). The DOE arguments are not consistent, however, with the cladding temperatures and stresses documented in the analysis and model report (CRWMS M&O, 2000f, pp.19 and 20). According to DOE analyses, the center rod in an average waste package will reach 308 °C [586 °F], and the outer rods will peak at 291 °C [556 °F]. The temperature uncertainty is assumed uniformly distributed throughout a range of ±13.5 percent. Thus, the hottest center rod in an average waste package could peak at 350 °C [662 °F], while the hottest outer rod could peak at 314 °C [597 °F]. Solubility values of hydrogen in Zircaloy are 80 and 120 ppm at 314 °C [597 °F] and 350 °C [662 °F] (CRWMS M&O, 2000h, p. 57), whereas the average hydrogen content in commercial spent nuclear fuel rods is approximately 400 ppm in the form of hydrides. As the fuel rod temperature increases to the peak temperature, some precipitated hydrides would dissolve, and hydrogen will return to solid solution. The dissolved hydrogen will reprecipitate as radial hydrides if the cladding stress exceeds a critical value during the precipitation process. The tensile stress for hydride reorientation is estimated to be between 69 and 208 MPa [10 and 30.2 ksi]. CRWMS M&O (2000j) and the DOE calculations of the cladding stresses for the temperature range 250–385 °C [482–725 °F] result in values ranging between 55 and 120 MPa [7.8 and 17.4 ksi]. This range of stresses is well within the minimum tensile stress for hydride reorientation to occur when the cladding cools slowly below the solvus temperature in the repository. Uncertainties regarding the calculated values of cladding temperatures and stresses, including uncertainties related to the temporal and spatial variations expected for thousands of waste packages, must be taken into account when considering hydride reorientation and hydride-induced failure. The DOE analysis of delayed hydride cracking was based on properties of Zircaloys that contain circumferential hydrides, which would not be applicable if hydride reorientation occurs. The prediction of the lack of potential for delayed hydride cracking based on a K_{IH} of 5 MPa•m$^{1/2}$ [4.55 ksi•in$^{1/2}$] might not be conservative if hydride reorientation occurs in the cladding. Thus, it is important to consider the distributions of cladding stresses and temperatures and their evolution on disposal in the repository considering spatial variations. The accuracy and validity of the stress and temperature data will determine if hydride embrittlement should be considered as an important failure process for spent nuclear fuel cladding to be incorporated into the model abstraction for radionuclide release. DOE agreed[12] to address concerns regarding cladding temperature and stress related to hydrogen embrittlement.

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[12]Schlueter, J. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000)." Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

3.3.4.4.3.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the effect of cladding degradation of commercial spent nuclear fuel on radionuclide release rates and solubility limits with respect to the characterization and propagation of model uncertainty through the model abstraction.

Current model uncertainty characterization and use are insufficient for certain aspects of commercial spent nuclear fuel cladding degradation. In particular, alternative models or consideration of model uncertainties are not sufficiently used for localized corrosion and stress corrosion cracking.

The DOE abstraction considered most forms of degradation that may affect the integrity of commercial spent nuclear fuel cladding during disposal conditions, including creep, localized corrosion, stress corrosion cracking, hydride reorientation and embrittlement, and mechanical failure (CRWMS M&O 2000b), which agree with the evaluation NRC conducted (2001). After comparing the results with various alternative creep models to define the creep damage in zirconium cladding on disposal, DOE used an empirical creep model developed by Matsuo (1987) and computed the creep strain as a function of initial rod stress for cladding in dry storage alone and in dry storage with transportation. An assumed temperature history profile representative of dry storage and transportation conditions was used (CRWMS M&O, 2000g). After an evaluation of six creep models against five sets of experimental data, DOE elected Murty's creep model rather than one of other five models, including the one by Matsuo (CRWMS M&O, 2000f). DOE claimed that Murty's creep equations are accurate at low stresses and low temperatures because the equations incorporate Coble creep, which is dominant at low stresses and low temperatures. In addition to Coble creep, Murty's creep equations include primary and steady-state creep by dislocation glide—the same creep mechanisms treated in Matsuo's model. [Model uncertainty in creep correlations of all five sets of experimental data as given by the weighted average of the relative error is 0.487 for Matsuo's model and 0.557 for Murty's model (CRWMS M&O, 2000f).] A critical strain criterion was used for creep failure. Upper and lower limits of rod failure by creep were computed based on creep failure strain limits of 0.4 and 11.7 percent. These creep failure strains were supported by experimental data of unirradiated Zircaloy and corresponded to an average creep failure strain of 3.3 percent used in an earlier analysis concerning cladding failure by creep during dry storage and transportation (CRWMS M&O, 2000g). The Murty's model and the creep strain criteria are acceptable because they both lead to conservative failure estimates.

In excluding hydride reorientation, DOE also argued that the fracture strength of zirconium cladding with reoriented hydrides remains high. There is a concern that a global stress failure based on fracture strength might not be appropriate for treating hydride embrittlement. The tensile ductility of zirconium is known to decrease with the length of radial hydrides. Puls (1988, Table IV) reported the tensile ductility of Zr-2.5 wt % Nb decreased from 12.8 to 1 percent when the hydride length increased from 20 to 150–450 μm [0.79 to 5.9–18 mils], even though the ultimate fracture strength only decreased from 866 to 715 MPa [125 to 104 ksi]. The slow cooling rate in the repository is conducive to the formation of long radial hydrides and a continuous hydride network (Chan, 1996). DOE should include hydride reorientation in its analyses of cladding failure and consider the possibility that hydride reorientation might lower the upper limit of the failure strain (11 percent) in the creep failure criterion and the K_{IH} {5 MPa•m$^{1/2}$ [4.55 ksi•in$^{1/2}$]} in delayed hydride cracking. In conclusion, the DOE analyses of delayed hydride cracking relied solely on a large crack fracture mechanics approach. In addition, no consideration was given to crack initiation at large hydrides. DOE discounted the importance of this failure event on the basis that this failure process can occur only for Zircaloy-4 cladding of pressurized water reactor fuel assemblies with a burnup exceeding 55 MWd/Kg [25 MWd/lb] uranium (CRWMS M&O, 2000f). The percentage of pressurized water reactor assemblies with burnup exceeding 55 MWd/Kg [25 MWd/lb] uranium, however, is approximately 15 percent (CRWMS M&O, 2000g). The possible failure rate of these high burnup fuel rods has not been considered. DOE agreed[13] to address concerns of hydrogen embrittlement.

Finally, no alternative models have been considered for localized corrosion and external stress corrosion cracking. This lack of alternative models can be acceptable if DOE demonstrates in the analysis and model report that the environmental conditions are not conducive to localized corrosion or stress corrosion cracking induced by chloride because (i) the chloride concentration is too low, (ii) the corrosion potential is lower than the pitting potential, or (iii) anionic species, such as nitrate, are present at a sufficiently high concentration ratio with respect to chloride that can act as efficient localized corrosion inhibitors. The hoop stress calculations used to evaluate creep are applicable to the assessment of chloride-induced stress corrosion cracking. DOE agreed[14] to address concerns of the effects of in-package chemistry on cladding degradation caused by localized corrosion and stress corrosion cracking.

3.3.4.4.3.5 Model Abstraction Output Is Supported by Objective Comparisons

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the effect of cladding degradation of commercial spent nuclear fuel on radionuclide release rates and solubility limits with respect to model abstraction output being supported by objective comparisons.

To date, adequate verification of the model abstraction for cladding degradation is not available. As noted before, DOE has not provided empirical demonstration through experiments, using simulated in-package environments, to verify that localized corrosion by fluoride anions is a valid process to be modeled and abstracted for incorporation into the DOE Total system Performance Assessment Code or at least bound the rate at which other corrosion processes may perforate the cladding. A similar argument is valid for the model abstraction of stress corrosion cracking in which only iodide is considered the causative agent for stress corrosion cracking. This internal stress corrosion cracking process has not been verified for the

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[13]Schlueter, J. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000).” Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[14]Ibid.

conditions expected in the repository. DOE agreed[15] to provide a technical basis for the various modes of cladding degradation.

3.3.4.4.4 Commercial Spent Nuclear Fuel Dissolution

The commercial spent nuclear fuel dissolution rates have been measured using a wide range of techniques, including flow-through experiments using spent nuclear fuel and UO_2 pellets, static tests in autoclaves, and unsaturated drip tests with spent nuclear fuel pellets contained in zirconium. Only data from the flow-through tests, however, are used to derive the dissolution rate model for total system performance assessment (CRWMS M&O, 2000b,j). Two regression equations are used in the DOE abstraction of the commercial spent nuclear fuel dissolution rate presented in the process model report (CRWMS M&O, 2000b):

for pH > 7,

$$\text{Log(Rate)} = 4.69 - \frac{1085}{T} + 0.12\log_{10}[\text{CO}_3]_{\text{Total}} + 0.32\log_{10}[P_{\text{O}_2}] \qquad (3.3.4\text{-}1)$$

and for pH ≤ 7,

$$\text{Log(Rate)} = 7.13 - \frac{1085}{T} + 0.32\log_{10}[P_{\text{O}_2}] - 0.41\text{pH} \qquad (3.3.4\text{-}2)$$

where rate is expressed in mg/m^2 • day, T is the absolute temperature in K, carbonate concentration is in moles/liter, and oxygen partial pressure is in atmospheres. The abstracted equations are derived so the rates from the two equations are equal at pH 7. The burnup in these tests ranged from 0 to 50 MWd/kg [0 to 23 MWd/lb] uranium.

It must be noted that Eq. (3.3.4-1) is an empirical regression model loosely based on irreversible thermodynamic reasoning (Stout and Leider, 1998a,b). The regression coefficient, adjusted R^2, for the high pH equation is 0.5014 (CRWMS M&O, 2000k), indicating the model does not represent a significant portion of the variance in the experimental data. A more elaborate model, with cross terms and a term involving burnup, was proposed by Stout and Leider (1998a,b) and has a much better statistical fit to the data (adjusted $R^2 = 0.8174$).

Equation (3.3.4-2) is derived by assuming that the dependence of dissolution rate on oxygen partial pressure and temperature is the same at pH values below 7 as it is above this pH. The term involving carbonate is neglected based on the reasoning that surface adsorption of carbonate ions is negligible below this pH. Additionally, one experimental data point at pH 3 and the calculated rate at pH 7 from Eq. (3.3.4-1) are used to derive the slope of the pH dependence for pH values between 3 and 7. The statistical significance of the abstraction for the acid environment is difficult to estimate because it is based on only two data points, one of

[15]Schlueter, J. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000)." Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

which is a calculated value. The model is then compared to other rate measurements and found to predict higher rates than the experiments, thus justifying its use as a bounding model. The fuel burnup is also not considered directly in the abstracted model, although a variety of burnups was used in the flow-through tests.

Unsaturated drip tests were performed by DOE during the past 8 years. The tests involved spent nuclear fuel contained in Zircaloy holders exposed to dripping water or a moist environment. The drip rates used, 0.0078–0.078 L/yr [0.0021–0.021 gal/yr], are much lower than those assumed in the in-package calculations, 1.5–150 L/yr [0.40–40 gal/yr]. The drip rates used should scale to the surface area of reacting media exposed. Based on 1 cm^2 [0.155 in^2] of fuel surface, the low end of the drip rate would correspond to approximately 8 cm/yr [3.1 in/yr] of dripping, which is much larger than the seepage rates predicted in the Total System Performance Assessment–Site Recommendation model (CRWMS M&O, 2000t). This scaling relationship remains poorly understood—it may depend on the manner in which dripping water contacts the fuel (Wronkiewicz, et al., 1992). The release rates of various radionuclides were monitored. The release rates of Tc-99 and Sr-90 were used to derive the intrinsic dissolution rate of the spent nuclear fuel (CRWMS M&O, 2000b,k) The dissolution rates measured in the high-drip-rate tests are lower than that predicted by Eq. (3.3.4-1) if a surface roughness factor of three is assumed in the drip tests (CRWMS M&O, 2000k). The low-drip-rate tests exhibited lower dissolution rates. The drip tests showed that Np-237 and Pu-239 are retained in the corrosion products after an initial period of high release (CRWMS M&O, 2000b).

The staff review regarding the DOE abstraction of commercial spent nuclear fuel dissolution follows.

3.3.4.4.4.1 System Description and Model Integration Are Adequate

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the effect of commercial spent nuclear fuel dissolution on radionuclide release rates and solubility limits with respect to system description and model integration.

DOE provided a detailed description of the commercial spent nuclear fuel characteristics, numbers, and design of the waste package internal components (CRWMS M&O, 2000b). An empirical model is used in the Total System Performance Assessment–Site Recommendation that is based on extensive measurements of spent nuclear fuel and unirradiated UO_2 dissolution in flow-through tests (CRWMS M&O, 2000k). DOE also cites measurements of the spent nuclear fuel dissolution rate using other test techniques, notably batch tests for fully immersed conditions and drip tests in partially saturated conditions. These tests and the measurement of mineral assemblages in the natural analog site at Peña Blanca are used appropriately as supporting evidence rather than to derive alternate spent nuclear fuel dissolution models for total system performance assessment. In the Total System Performance Assessment–Site Recommendation, the in-package chemistry calculation is linked to the spent nuclear fuel dissolution model.

The DOE integration of the dissolution rate model in the overall total system performance assessment is sufficient. DOE agreed[16] to provide additional information and analyses on the in-package chemistry critical to determining the commercial spent nuclear fuel dissolution rate. DOE stated that, in a future revision of the analysis and model report (CRWMS M&O, 2000d), specific NRC questions on radiolysis, chemistry of incoming water, localized corrosion, corrosion products, and transient effects will be addressed. A sensitivity study on differing dissolution rates of components will be included, as well as a more detailed calculation of the in-package chemistry effects of radiolysis, the effects of engineered materials on the chemistry of water used for input to in-package abstractions, and the applicability of abstractions for incoming water, taking into account information from a future revision of the analysis and model report (CRWMS M&O, 2000l). DOE stated current planning provides for additional analysis for in-package chemistry model support. This analysis will determine which parts of the model are amenable to additional support by testing and which parts are amenable to sensitivity analysis or use of analogues.

3.3.4.4.4.2 Data Are Sufficient for Model Justification

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the effect of commercial spent nuclear fuel dissolution on radionuclide release rates and solubility limits with respect to data being sufficient for model justification.

Insufficient data have been presented to justify the abstracted model of spent nuclear fuel dissolution in the acid range of the model. Furthermore, the abstracted model eliminated the term related to burnup of fuel, without considering results from high burnup fuels. The DOE model for spent nuclear fuel dissolution evolved from a 12-parameter model (involving burnup, temperature, pH, oxygen, and carbonate and their interaction terms) to a 4-parameter model (involving temperature, pH, carbonate, and oxygen). The effect of burnup is suggested insignificant (Shoesmith, 1999) in comparison to other factors. Tests continue on high burnup fuel, however, which may alter the abstracted model. The linear regression model used with the limited number of parameters explains only a portion of the observed variance in the experimental data (adjusted R^2 = 0.5014), although it is argued that the model represents a bounding case. The reason for going from a more complex model to a simpler model is not clear. Furthermore, the statistical significance of the abstraction of the acid side of the model is difficult to estimate because it is based on only two data points, one of which is a calculated value. In deriving the abstracted model for commercial spent nuclear fuel dissolution, the flow-through corrosion test data for commercial spent nuclear fuel spans the pH range 8–10. Tests on the unirradiated UO_2 test data span the pH range 3.5–11.6 (CRWMS M&O, 2000k),

[16]Schlueter, J. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000)." Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

but the acid test data are used only for confirmation purposes. DOE agreed[17] to address these concerns.

3.3.4.4.4.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the effect of commercial spent nuclear fuel dissolution on radionuclide release rates and solubility limits with respect to the characterization and propagation of data uncertainty through the model abstraction.

DOE has not provided adequate information about how the uncertainties in spent nuclear fuel dissolution rate data and the various parameters used in the calculation of in-package chemistry are propagated through model abstractions and predictions of radionuclide release rates from spent nuclear fuel. The commercial spent nuclear fuel dissolution model is coupled to the calculated in-package chemistry. The in-package chemistry calculation abstraction (CRWMS M&O, 2000e) suggests that the in-package chemistry is likely to be near-neutral or alkaline during the long time period. The in-package chemistry model has data uncertainties related to the spent nuclear fuel dissolution rates, the dissolution rates of other in-package components, and the local chemical changes in crevices between cladding and fuel, between fuels, or between basket material and fuel. Additionally, uncertainties exist regarding incoming water chemistry. Similarly, there are uncertainties in the dissolution rates of spent nuclear fuel, especially in the acid side, where data are sparse. Finally, DOE is currently testing the high burnup fuel, and the data have not been included in the model abstraction. DOE agreed[18] to provide an update on the in-package chemistry effects on dissolution rates.

3.3.4.4.4.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the effect of commercial spent nuclear fuel dissolution on radionuclide release rates and solubility limits with respect to the characterization and propagation of model uncertainty through the model abstraction.

DOE relied primarily on flow-through test data to construct its abstracted model for commercial spent nuclear fuel dissolution rate (CRWMS M&O, 2000b,k). The electrochemical mechanism was used to justify the dissolution rate data derived from flow-through tests (Shoesmith, 1999).

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[17]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration—Features, Events, and Processes (May 15–17, 2001).” Letter (May 30) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[18]Schlueter, J. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000).” Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000

DOE also suggested that the flow-through test results form an upper bound of dissolution rates measured by other techniques. DOE has not considered alternate models derived from the unsaturated drip tests, the immersion tests, or natural analogues. Although the drip test model and natural analog data may provide more realistic assessments of the spent nuclear fuel dissolution rate, the choice of the conservative flow-through test to support commercial spent nuclear fuel dissolution is acceptable.

3.3.4.4.4.5 Model Abstraction Output Is Supported by Objective Comparisons

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5) is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the effect of commercial spent nuclear fuel dissolution on radionuclide release rates and solubility limits with respect to model abstraction output being supported by objective comparisons.

The model abstraction used for the commercial spent nuclear fuel dissolution rate in the Total System Performance Assessment–Site Recommendation (CRWMS M&O, 2000t) is based on experimental measurements. The flow-through experiments used to derive the model are considered bounding because the dissolution process is not limited by transport of species, corrosion products, or back reactions. The flow-through tests, however, do not adequately simulate the geometries and material interactions that can occur in the waste package. The flow-through experiments also do not correspond to natural analogs because of the lack of secondary minerals in the former, which are expected to lower the dissolution rate. Therefore, the model abstraction cannot be verified by long-term experiments or natural analogues but can be shown to be conservative.

3.3.4.4.5 DOE Spent Nuclear Fuel Dissolution

DOE spent nuclear fuel consists of more than 250 distinct spent nuclear fuel types divided into 11 groups. In addition, the process model report (CRWMS M&O, 2000b) considered immobilized ceramic plutonium waste. This waste form will consist of disks of a plutonium-containing, titanium dioxide-based ceramic enclosed in stainless steel cans. The process model report evaluated the following 12 types of fuels and waste forms:

Group 1 — Naval spent nuclear fuel
Group 2 — Plutonium/uranium alloy
Group 3 — Plutonium/uranium carbide
Group 4 — Mixed oxide and plutonium oxide fuels
Group 5 — Thorium/uranium carbide
Group 6 — Thorium/uranium oxides
Group 7 — Uranium metal
Group 8 — Uranium oxide
Group 9 — Aluminum-based spent nuclear fuel
Group 10 — Unknown
Group 11 — Uranium-zirconium-hydride
Group 12 — Immobilized ceramic plutonium waste

The process model report considered three types of degradation models for DOE spent nuclear fuel and waste forms: upper limit, conservative, and best estimate. The upper-limit model predicts release rates that are always well in excess of actual dissolution rates. The conservative degradation model provides an estimate of a dissolution rate that reflects the higher end of available dissolution data for the spent nuclear fuel groups or similar materials. Presently, there are no directly relevant experimental dissolution/degradation data for many DOE spent nuclear fuel waste forms. Only limited test data are available on some DOE spent nuclear fuel waste forms. Because of the lack of available data, various surrogate spent nuclear fuels were evaluated for degradation behavior to develop the conservative and best-estimate models. A full instantaneous release of radionuclides was assumed for the upper-limit model for all waste forms except Group 1. Models for the Group 1 fuel—Naval spent nuclear fuel—will be provided later by the U.S. Navy.

Because of the large effort expected for qualifying the conservative and best-estimate models, DOE conducted total system performance assessment sensitivity analyses for DOE spent nuclear fuel. Initial results indicate the performance of the repository is insensitive to DOE spent nuclear fuel degradation kinetics. That is, use of the upper-limit model, which predicts instantaneous release of radionuclides, for DOE spent nuclear fuel in the total system performance assessment still resulted in a calculated dose to the receptor group well within safety requirements. For its Total System Performance Assessment–Site Recommendation (CRWMS M&O, 2000t) model, DOE conservatively assumed the dissolution rate is a constant value equal to the rate for uranium-metal-based fuel (CRWMS M&O, 2000m). The assumed rate results in the complete dissolution of the fuel in a single timestep and in the release of the entire DOE spent nuclear fuel inventory in the waste package as soon as the package is breached (CRWMS M&O, 2000m).

The staff review regarding the abstraction of DOE spent nuclear fuel dissolution follows.

3.3.4.4.5.1 System Description and Model Integration Are Adequate

Description of the characteristics, dissolution processes, and integration of the dissolution rates for DOE spent nuclear fuel types is limited. Additional information regarding system description and model integration for DOE spent nuclear fuel degradation is not needed, however, because DOE uses the upper-limit model, which predicts instantaneous release of radionuclides for every type of DOE spent nuclear fuel. Thus, the impact of DOE spent nuclear fuel on the performance of the repository would depend only on the total inventory of the radionuclides in DOE spent nuclear fuel (CRWMS M&O, 2000b,m), and that inventory is adequately defined.

3.3.4.4.5.2 Data Are Sufficient for Model Justification

Data on the characteristics of the large number of DOE spent nuclear fuel types presented in the process model report (CRWMS M&O, 2000b) are limited. Additional data to support abstraction of DOE spent nuclear fuel degradation are not needed, however, because DOE uses the upper-limit model, which predicts instantaneous release of radionuclides, in its total system performance assessment analyses for every type of DOE spent nuclear fuel.

3.3.4.4.5.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Use of the upper-limit model by DOE in its total system performance assessment analyses for every type of DOE spent nuclear fuel is reasonable. No additional information is needed regarding the characterization and propagation of data uncertainty through the abstraction of DOE spent nuclear fuel dissolution.

3.3.4.4.5.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

The use of the upper-limit model by DOE in its total system performance assessment analyses for every type of DOE spent nuclear fuel is reasonable. No additional information is needed regarding the characterization and propagation of model uncertainty through the abstraction of DOE spent nuclear fuel dissolution.

3.3.4.4.5.5 Model Abstraction Output Is Supported by Objective Comparisons

The use of the upper-limit model, by DOE in its total system performance assessment analyses for every type of DOE spent nuclear fuel is reasonable. No additional information is needed regarding model support for the abstraction of DOE spent nuclear fuel dissolution.

3.3.4.4.6 High-Level Waste Glass Dissolution

The basic form of the rate expression adopted by DOE (CRWMS M&O, 2000n) to describe the dissolution of waste glass immersed in water is given by a form of transition state rate law as

$$\text{Rate} = S\left\{k_o \bullet 10^{\eta \cdot \text{pH}} \bullet \exp\left(\frac{-E_a}{RT}\right) \bullet \left[1 - \frac{Q}{K}\right] \right. \tag{3.3.4-3}$$

where

S — surface area of glass immersed in water, in units of area
k_o — intrinsic dissolution rate, which depends only on glass composition, in units of mass/(area • time)
η — pH dependence coefficient
E_a — effective activation energy, in units of kJ/mol
R — gas constant, which is 8.314 J/(mol • K) [1.987 cal/(mol • K)]
T — absolute temperature in K
Q — concentration of dissolved silica in the solution, in units of mass/volume
K — a quasi-thermodynamic fitting parameter for glass equal to the apparent silica saturation value for the glass, in units of mass/volume

Equation (3.3.4-3) contains two main factors. The first factor is the forward rate, $k_o \bullet 10^{\eta \bullet \text{pH}} \bullet \exp(-E_a/RT)$, which represents the dissolution rate in the absence of concentration effects of dissolved silica (and other aqueous species), and the other factor is the reaction affinity term $1-(Q/K)$, which quantifies such effects. Because of the complexity in defining parameters and associated uncertainties, a simpler bounding approach was adopted that combined $1-(Q/K)$ with k_o, and the following abstraction was developed for aqueous degradation of high-level waste:

$$\text{Rate} = S\left\{k_{\text{eff}} \bullet 10^{\eta \cdot \text{pH}} \bullet \exp\left(\frac{-E_a}{RT}\right)\right. \qquad (3.3.4\text{-}4)$$

where

$$k_{\text{eff}} = k_o \bullet (1 - \frac{Q}{K}) \qquad (3.3.4\text{-}5)$$

This bounding approach reduces the abstracted model to an equation involving four parameters (η, E_a, S, and k_{eff}) and two variables (pH and T). The forward rate was measured in flow-through experimental conditions where the affinity term can be maintained close to one because of the absence of concentration effects from the products of the glass dissolution. Test results indicated that the rate dependence on pH and temperature was independent of the glass composition, within the range of the glass compositions tested, and, therefore, the same values were used for all waste glasses. The log of the dissolution rate exhibited a V-shaped curve when plotted versus pH. The value of k_{eff} was determined through experimental observations. Several options were evaluated to conservatively bound the three stages of glass corrosion. Based on this evaluation, data from the product consistency test (PCT)-A test were used to obtain bounding values for k_{eff}. The exposed surface area was estimated based on 20 times the surface area of the glass log and assumed that the entire surface corrodes at the same rate when exposed to water. In addition, the DOE model assumes the surface area remains constant during the corrosion process.

Because of the discontinuity in the log of the dissolution rate as a function of pH at intermediate pHs, separate rate expressions were obtained for the acid range and the alkaline range, as shown by Eqs. (3.3.4-6) and (3.3.4-7) (CRWMS M&O, 2000o).

For the low pH range (pH < pH_m)

$$\frac{\text{Rate}}{S}(\text{gm} \bullet \text{m}^2/\text{day}) = 10^{(14 \pm 0.5)} \bullet 10^{(-0.6 \pm 0.1) \bullet \text{pH}} \bullet \exp\left(\frac{-80 \pm 10}{RT}\right. \qquad (3.3.4\text{-}6)$$

For the high pH range (pH $\geq$ pH_m)

$$\frac{\text{Rate}}{S}(\text{gm} \bullet \text{m}^2/\text{day}) = 10^{(6.9 \pm 0.5)} \bullet 10^{(0.4 \pm 0.1) \bullet \text{pH}} \bullet \exp\left(\frac{-80 \pm 10}{RT}\right. \qquad (3.3.4\text{-}7)$$

where pH_m, equal to 7.1, is the pH at which a minimum dissolution rate occurs.

The staff review regarding the DOE abstraction of high-level waste glass dissolution follows.

3.3.4.4.6.1 System Description and Model Integration Are Adequate

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the effect of high-level waste glass dissolution on radionuclide release rates and solubility limits with respect to system description and model integration.

DOE has not accounted for the range of environmental conditions expected inside breached waste packages in its abstraction of high-level waste glass degradation. Many studies and reviews have been reported on the effects of γ- and α-radiations on the dissolution or alteration of glass waste in the moist-air systems (Burns, et al., 1982; Wronkiewicz, et al., 1994, 1997). Wronkiewicz, et al. (1997) reported that although both γ- and α-radiations have no adverse effects on the dissolution of nuclear glass waste form immersed in water in contact with air, the radiation exposure of the glass waste form to humid air resulted in a four-to-tenfold increase of alteration layer thickness relative to samples reacted without radiation exposure. Wronkiewicz, et al. (1994, 1997) suggested increases for the irradiated humid-air experiments appear to result from condensation of radiolytic acids into the thin film of water contacting the glass surface. The radiolytic acids increased the rate of ion exchange between the glass and the thin film of condensate, resulting in accelerated corrosion rates for the glass. DOE should consider this finding in its evaluation of the dissolution of glass waste form because, after the failure of the waste package, the glass waste form may be exposed to a thin film of water in dripping conditions, and the radiation dose rate from the long-lasting alpha-emitters in the glass waste form still may be high enough to produce a significant effect. On the other hand, the radiolysis-induced nitric acid is a stable product with repository conditions and, therefore, may accumulate on the surface of the glass waste form and produce an acidic film of water even if the radiation field is low after failure of the waste package. DOE agreed[19] to provide an update on the in-package chemistry effects on dissolution rates.

DOE conducted limited analyses of high-level waste glass degradation in the presence of corrosion products from the dissolution of waste package internal components, such as FeOOH, $FeCl_2$, and $FeCl_3$, that could influence glass corrosion processes. DOE stated (CRWMS M&O, 1998b) that dissolution rates of glass strongly decrease in the presence of dissolved magnesium, lead, and zinc, but are strongly enhanced in some conditions by dissolved iron. The potential effect of dissolved iron is particularly important because corrosion of the stainless steel inner barrier of the Enhanced Design Alternative-II design could provide significant quantities of iron. DOE agreed[20] to provide an update on the in-package chemistry effects on dissolution rates.

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[19]Schlueter, J. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000)." Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[20]Ibid.

3.3.4.4.6.2 Data Are Sufficient for Model Justification

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the effect of high-level waste glass dissolution on radionuclide release rates and solubility limits with respect to data being sufficient for model justification.

Based on review of the DOE abstraction of high-level waste glass degradation presented in the process model report (CRWMS M&O, 2000b) and in the analysis and model reports (CRWMS M&O, 2000n,o), the data and technical bases used to support the model abstraction are not sufficient.

- The analysis and model report (CRWMS M&O, 2000n, p. 9, Bullet #1) assumes the high-level waste glass dissolution rate based on the boron release rate can be used to provide an upper bound for the radionuclide release rate. The analysis for determining the coefficients for the pH dependence violates this assumption. While the dissolution rate based on the boron release rate was used for calculating k_{eff}, the coefficients for the pH dependence of the dissolution rate were determined using the silica release rate. Silica has limited solubility, and high-level waste glass dissolution rates could be significantly underestimated if based on measured silica release rates. This closed-pending agreement was addressed in the revised analysis and model report (CRWMS M&O, 2000o).

- The pH coefficients for high-level waste glass degradation rates were determined from experiments that used pH buffers to prepare aqueous solutions. DOE conducted limited experiments to determine the possible effect on glass dissolution of corrosion products that could result from dissolution of waste package internal components. Corrosion products, such as FeOOH, $FeCl_2$, and $FeCl_3$, could influence the mechanisms and rates of glass corrosion (Pan, et al., 2001). DOE stated that glass dissolution rates are strongly enhanced in some conditions by dissolved iron. The potential effect of iron is particularly important because corrosion of the stainless steel inner barrier of the Enhanced Design Alternative–II design could provide significant quantities of dissolved iron.

- The work of Advocat, et al. (1991), cited in the analysis and model report for the effect of pH on release rate, indicates the presence of potassium ions on the surface of the corroded glass. Because the glass had no potassium, the presence of potassium ions is attributed to the ion exchange from KOH or KH_2PO_4 used for adjusting the pH of the solutions. The potassium ion, by virtue of its larger size, could lower the release rate from glass by retarding the migration of hydrogen ions in the glass matrix. Such comparisons could lead to erroneous conclusions.

- DOE assumed the release rate is independent of high-level waste glass composition. At best, one can state the intrinsic dissolution rate, k_o, can be represented as an expected value of a distribution based on the expected variation in glass compositions using a risk-informed, performance-based evaluation. In addition, the coefficients for pH and E_a

are assumed to be independent of glass composition. Again, pH and E_a values should bound the variability expected from glass compositions. This analysis is acceptable as long as it captures the expected variability in glass composition.

DOE agreed[21] to provide revised documentation on in-package water chemistry modeling for waste forms. The revised documentation will include an assessment of the chemical form and concentration of iron corrosion products and their effects on glass dissolution rates.

3.3.4.4.6.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the effect of high-level waste glass dissolution on radionuclide release rates and solubility limits with respect to data uncertainty characterized and propagated through the model abstraction.

The DOE model for high-level waste glass dissolution is based on a single set of experiments conducted by Knauss, et al. (1990). This experiment defines the glass dissolution dependence on pH and temperature for a single, simple glass composition. Although DOE bounded the forward reaction-rate term in the model by performing several sets of experiments using various glass compositions, the uncertainties associated with pH and temperature dependence have not been evaluated using anticipated glass compositions. The DOE model lacks evaluation of data and model uncertainties. Because DOE bounded high-level waste glass dissolution rates using a conservative forward reaction rate, no additional information is needed regarding the characterization and propagation of data uncertainty through the abstraction of the high-level waste glass dissolution.

3.3.4.4.6.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the effect of high-level waste glass dissolution on radionuclide release rates and solubility limits with respect to the characterization and propagation of model uncertainty through the model abstraction.

3.3.4.4.6.5 Model Abstraction Output Is Supported by Objective Comparisons

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the effect of high-level waste glass dissolution

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[21]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

on radionuclide release rates and solubility limits with respect to model abstraction output being supported by objective comparisons.

3.3.4.4.7 Radionuclide Solubility

The DOE approach to calculate bounds on the aqueous concentration of radionuclides in water that reacted with the waste form is initially to derive the concentrations from the waste form dissolution model. Subsequently, a comparison is made between the waste form dissolution-based aqueous concentration of the radionuclides and a value for the solubility limit, thermodynamically derived or based on a bounding assumption, for each radionuclide considered. If the solubility-limited value is lower for a given radionuclide than its concentration derived from the waste form dissolution, the aqueous concentration is set to the solubility-limited value, and the difference in mass is assumed to precipitate out of solution. The solubility-limited values place constraints on the aqueous concentration of the particular radionuclide element considered with each isotope of that element present in proportion to its isotopic abundance (CRWMS M&O, 1998b).

The concentration usually is constrained by the solubility limit of the solid phases that contains the radioisotopes (either solid phases with the radioisotope as the dominant element or solid phases with trace amounts of the radionuclide, as in coprecipitated species). The solid phases that form depend on temperature, redox conditions, and chemical composition of the groundwater. Because of uncertainty in the precise values for these variables in the waste package and near-field environment, there is a wide range of possible radionuclide concentration limits.

For the Total System Performance Assessment–Site Recommendation (CRWMS M&O, 2000t) the dissolved concentration limits calculation builds on three primary feeds: (i) estimates of in-package fluid chemistry (pH, Eh, ionic strength, and carbonate concentration), (ii) measured (and estimated) thermodynamic parameters describing the stabilities of aqueous species and solid radioisotope phases, and (iii) determinations of the likely solubility controlling phases for the radionuclides of concern (CRWMS M&O, 2000b). For the Total System Performance Assessment–Site Recommendation analysis, pure phases were chosen because, in general, they yield higher dissolved concentrations compared to coprecipitated phases. The specific phase selected for a particular radionuclide is based on information from geologic and experimental observations or from crystallochemical arguments. Where no information can be gleaned from field or experimental observations, the most amorphous and hydrated form of the radionuclide believed the most soluble was selected. For uranium, schoepite was assumed the solubility-controlling phase. For neptunium, plutonium, americium, and nickel, the solubility-controlling solids chosen were Np_2O_5 [or $Np(OH)_4$(am) for reducing conditions], $Pu(OH)_4$(am), $AmOHCO_3$, and NiO (CRWMS M&O, 2000p).

Thermodynamic data available for the different radionuclides, the sensitivity of solubilities to fluid chemistry, and the importance of the different radionuclides to total system performance assessment are uneven. Thus, DOE used three approaches to implement solubility limits within the Total System Performance Assessment–Site Recommendation analysis (CRWMS M&O, 2000m,b,p): (i) multitermed functions of chemistry for uranium, neptunium, americium, actinium, curium, and samarium; (ii) distributions for plutonium, protactinium, lead, and nickel;

and (iii) constant bounding values for technetium, iodine, thorium, cesium, strontium, chlorine, carbon, niobium, zirconium, radium, and tin. The concentration of uranium for the Total System Performance Assessment–Site Recommendation analysis was calculated using an equation fit to EQ3-derived schoepite solubility as a function of pH, CO_2 fugacity, and temperature. The solubility of neptunium for the Total System Performance Assessment–Site Recommendation analysis was calculated from a pH-dependent equation fit to Np_2O_5 solubilities calculated with EQ3 for a pH range 4.5–8.5. A log-uniform distribution was assigned for plutonium solubility, with a minimum of 1.0×10^{-10} and a maximum of 2.0×10^{-4} M, based on EQ3 calculations of $Pu(OH)_4$ solubility in J–13 Well waters for a range of pH, Eh, and CO_2 fugacity. To calculate americium concentrations for the Total System Performance Assessment–Site Recommendation analysis, an equation with pH and CO_2 fugacity terms was used, and similar equations were used to calculate the solubilities of actinium, curium, and samarium. The solubilities of technetium, carbon, chlorine, iodine, and cesium were set to 1.0 M, which lets the waste inventory control release, because no solubility-limiting solids are predicted to form for these radioelements. The solubility of strontium was also set to 1.0 M to simplify the analysis. A log-uniform distribution was proposed for nickel solubility, assumed controlled by the solubility of NiO with a minimum of 1.4×10^{-6} M and a maximum of 3.1 M. For lead solubility, a log-uniform distribution was recommended for the Total System Performance Assessment–Site Recommendation with a minimum of 1.0×10^{-10} M and a maximum of 1.0×10^{-5} M. In the case of protactinium solubility, a log-uniform distribution was recommended, with a minimum of 1.0×10^{-10} M, a maximum of 1.0×10^{-5} M, and a mean of 3.2×10^{-8} M. Constant values of 1.0×10^{-7} M for the solubilities of niobium, 2.3×10^{-6} M for radium, 5.0×10^{-8} M for tin, 1.0×10^{-5} M for thorium, and 6.8×10^{-10} M for zirconium were recommended (CRWMS M&O, 2000b,p).

The staff review regarding the DOE abstraction of radionuclide concentration limits follows.

3.3.4.4.7.1 System Description and Model Integration Are Adequate

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess radionuclide solubility limits with respect to system description and model integration.

For most of the 21 radionuclides considered in the abstraction, the process model report (CRWMS M&O, 2000b) and analysis and model reports (CRWMS M&O, 2000p), provide sufficient descriptions of the approach and technical bases for estimating the solubility limit of the radionuclides and the integration of the radionuclide concentration limits into the total system performance assessment analyses. The use of 1 M as a conservative upper bound for the solubility limit of technetium, carbon, iodine, chlorine, cesium, and strontium is considered acceptable. The technical basis, however, is inadequate for the solubility limits of some radionuclides. In particular, actinium, curium, and samarium are all assumed to be analogous to americium and use the same pH- and fCO_2-dependent equation and parameter values, except the first one, as americium. No technical basis was provided, however, for the differences in the value of the first parameter in the equation.

DOE recognizes that solubility of an element varies as the environmental conditions within a repository change, and evaluation of solubility limits requires knowledge of changes in this environment and the dependence of radionuclide solubility on the environment. For several radionuclides considered (e.g., zirconium, nickel, tin, and radium), however, the abstraction relied on EQ3 equilibrium modeling that assumed the water had a composition similar to that of J–13 Well water. For those radionuclides, the pH range (6–9) used in the EQ3 calculations does not encompass the reasonable range of pH inside breached waste packages. The analysis and model report (CRWMS M&O, 2000d) indicates that pH inside commercial spent nuclear fuel waste packages can range from 3.6 to 8.1, whereas pH inside DOE spent nuclear fuel/high-level waste glass (codisposal) waste packages can range from 4.8 to 10.0. In addition, although the EQ3 results show that neptunium solubility varies with fCO_2 at pH > 7, DOE selected an equation dependent only on pH to represent the solubility of neptunium. DOE justified its neglect of fCO_2 dependence by claiming in-package chemistry calculations show the maximum pH inside breached waste packages is 8.1, which is true only for the commercial spent nuclear fuel waste packages, not for the codisposal packages. In the analysis and model report (CRWMS M&O, 2000p), DOE stated that analysis of in-package chemistry and analysis of solubility limits were conducted in parallel. The in-package chemistry calculated for the codisposal waste packages, which exhibit higher pH levels and ionic strengths than either J–13 Well water or commercial spent nuclear fuel waste packages, was not considered because commercial spent nuclear fuel is the dominant waste, and the resources for this analysis are constrained (CRWMS M&O, 2000p). Thus, the results of the in-package chemistry calculations were not fully used in evaluating the solubility limits. Also, the temperature dependence of radionuclide solubilities was generally ignored, except for uranium. Thus, DOE has not reasonably accounted for the range of environmental conditions expected inside breached waste packages in its abstraction of radionuclide concentration limits.
DOE agreed[22,23] to provide revised documentation on the effects of in-package water chemistry on radionuclide solubility.

3.3.4.4.7.2 Data Are Sufficient for Model Justification

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess radionuclide solubility limits with respect to sufficient data for model justification.

For radionuclides with adequate experimental data, DOE provided an adequate description how the experimental data and EQ3 modeling results were used, interpreted, and synthesized into the abstraction of radionuclide concentration limits. For radionuclides with inadequate experimental data, the assumption of 1 M as a conservative bounding limit is considered

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[22]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[23]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

acceptable. In several cases, however, inadequate justification is provided for parameters used in the solubility equation. For example, the equation for the solubility of actinium, curium, and samarium has the same form as that of americium, and six of the seven parameters in the equations have the same value. The first parameter in the equations is different for actinium, curium, samarium, and americium, but no technical basis was provided for the different values, and, as stated in the analysis and model report (CRWMS M&O, 2000p), no separate solubility evaluation has been conducted for actinium, curium, and samarium.

Also, data are lacking or inadequate to support the parameters used in the abstraction of concentration limits for several radionuclides. For example, the solubility of zirconium calculated using EQ3 is uncertain because, as stated in the analysis and model report (CRWMS M&O, 2000p), some data for zirconium complexes in the EQ3 database are suspect. There are no thermodynamic data available for the pertinent aqueous niobium species, and published niobium solubility data vary by several orders of magnitude. Furthermore, in several cases, DOE used data supplemented by EQ3 calculations to support its abstraction of solubility limits. The EQ3 calculations, however, did not encompass the potential range of chemical conditions that could be present inside breached waste packages.

In addition, the analysis and model report (CRWMS M&O, 2000q) states that the analysis did not use a uniform EQ3/6 data file because the data file was still being developed. Moreover, CRWMS M&O (2000q) states that unqualified data were used in the analysis. Thus, data DOE used to justify model abstraction of dissolved concentration limits are considered inadequate. DOE agreed[24] to provide documentation of all deviations from the reference EQ3/6 database and justification for those deviations.

3.3.4.4.7.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess radionuclide solubility limits with respect to the characterization and propagation of data uncertainty through the model abstraction.

The bounding values used by DOE for the solubility limits of technetium, carbon, iodine, chlorine, cesium, and strontium reasonably account for uncertainties and variabilities of the solubility of those radionuclides. The parameter values used in the solubility equations for several of the radionuclides do not adequately reflect the range of environmental conditions expected inside breached waste packages. DOE has not adequately considered the uncertainties in the in-package chemical environment in deriving the abstracted equations for

[24]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

the solubility limits of several radionuclides. DOE agreed[25] to provide revised documentation on the effects of in-package water chemistry on radionuclide solubility.

3.3.4.4.7.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess radionuclide solubility limits with respect to the characterization and propagation of model uncertainty through the model abstraction.

The effects of thermal-hydrological-chemical coupled processes that may occur inside waste packages and that may change the in-package chemistry are not appropriately considered in the DOE abstraction of dissolved radionuclide concentration limits. DOE agreed[26] to provide revised documentation on the effects of in-package water chemistry on radionuclide solubility.

3.3.4.4.7.5 Model Abstraction Output Is Supported by Objective Comparisons

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess radionuclide solubility limits with respect to model abstraction output being supported by objective comparisons.

DOE has not adequately supported, by objective comparisons with empirical data, the range of chemical conditions it used in deriving the abstracted equations for the solubility limits for several radionuclides. As noted in Section 3.3.4.4.2.5, in the revised analysis and model report (CRWMS M&O, 2001a), DOE states that planned validation of the in-package chemistry model will involve using EQ3/6 to model some combination of the following processes: (i) alteration observed during drip tests performed at Argonne National Laboratory; (ii) formation of ore deposits that might constitute natural analoges; and (iii) glass, mineral, and steel corrosion measurements taken in the laboratory. The planned DOE validation exercises are expected to address staff concerns about model abstraction of radionuclide solubility limits being supported by objective comparisons.

3.3.4.4.8 Colloidal Release

Colloidal radionuclide release from waste forms is addressed in two analysis and model reports: one, (CRWMS M&O, 2000r) describing the abstraction to be incorporated into the DOE total system performance assessment; the second, in support of the first (CRWMS M&O, 2000s). This colloid release abstraction is limited to defining colloid-associated concentrations of certain radionuclides in water as these leave the waste package. No retardation in the waste package

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[25]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[26]Ibid.

is assumed, and transport outside the waste package is not within the scope of the release model. For high-level waste glass, the abstraction allows reversible and irreversible radionuclide attachment to colloids. For spent nuclear fuel waste forms, irreversible attachment was not included in the abstraction.

The DOE abstraction of colloidal radionuclide release uses empirical data on release and colloid stability to formulate a dependence of colloidal radionuclide release on in-package ionic strength and pH. The abstraction analysis and model report (CRWMS M&O, 2000r) uses literature and Yucca Mountain project data to support the construction of an algorithm for calculating colloid-associated radionuclide concentrations in solutions leaving the waste package. No credit is taken for colloid retardation within the waste package. Direct input for conceptual models and parameters was obtained from Yucca Mountain project laboratory studies and from a few literature sources. The abstraction takes output from in-package geochemical models and uses pH, ionic strength, and dissolved radionuclide concentration to calculate colloid concentrations, irreversibly colloid-bound radionuclide concentrations, and reversible colloid binding of radionuclides. The results are combined to provide a total colloid-associated source term for a given radionuclide. The abstraction classifies colloids as waste form, groundwater (preexisting), or iron oxyhydroxide (from corrosion) colloids. True colloids (i.e., products of radionuclide precipitation) are not included.

The following key input are used in the colloid release abstraction (CRWMS M&O, 2000r):

- Solution ionic strength, pH, and radionuclide concentration from separate total system performance assessment in-package geochemical calculations

- Effect of ionic strength on water concentration of waste form colloidal plutonium, including a maximum colloidal plutonium concentration of 6×10^{-8} M at ionic strength <0.01 M and a minimum of 1×10^{-11} M at ionic strength >0.05 M, from data in an analysis and model report (CRWMS M&O, 2000s)

- Maximum stability limits for waste form colloids as a function of pH, ranging from ionic strength of 0.01 M at pH 2 to ionic strength of 0.05 M at pH $\geq$6, based on montmorillonite data from Tombacz, et al. (1990) and an analysis and model report (CRWMS M&O, 2000s)

- Maximum stability limits for iron oxyhydroxide colloids as a function of pH, ranging from ionic strength of 0.05 M at pH <6 and >11 to a minimum ionic strength of 0.01 M at pH 8–9, from Liang and Morgan (1990)

- Relationship between ionic strength and mass of groundwater colloids, ranging between a minimum of 3×10^{-6} mg/L and a maximum of 3×10^{-2} mg/L (CRWMS M&O, 1998b)

- Range of distribution coefficients for reversible sorption onto colloids using literature and Yucca Mountain project laboratory data

The order of calculation is

- Water concentration of radionuclide irreversibly sorbed to waste form colloids, using ionic strength and pH
- Mass concentration of waste form colloids, using experimental relationship between concentrations of colloids and radionuclide irreversibly sorbed to them
- Radionuclide reversibly sorbed to waste form colloids, using distribution coefficient
- Mass concentration of iron oxyhydroxide colloids, using ionic strength and pH
- Radionuclide reversibly sorbed to iron oxyhydroxide colloids, using distribution coefficient
- Mass concentration of groundwater colloids, using ionic strength
- Radionuclide reversibly sorbed to groundwater colloids, using distribution coefficient
- Summed colloidal radionuclide concentration and summed colloid mass concentration output to exterior of waste package

The analysis and model report (CRWMS M&O, 2000s) contains literature and previous Argonne National Laboratory data from static- and drip-corrosion tests on high-level waste glass and spent nuclear fuel supporting a model of irreversible plutonium colloid attachment used in the analysis and model report (CRWMS M&O, 2000r). The direct input to the adopted abstraction—all based on Argonne National Laboratory work—are (i) a relationship between colloidal plutonium concentration and ionic strength based on static high-level waste glass corrosion tests, (ii) the effect of ionic strength on colloid stability, and (iii) a direct relationship between colloidal plutonium concentration and colloid concentration. The adopted abstraction uses data only from the high-level waste glass tests, however, spent nuclear fuel results were included in the development of a model in the Argonne National Laboratory analysis and model report that was used in the abstraction analysis and model report as an alternative model.

The staff review regarding the DOE abstraction of colloidal release follows.

3.3.4.4.8.1 System Description and Model Integration Are Adequate

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the effect of colloid release on radionuclide solubility limits with respect to system description and model integration.

DOE has not yet assembled the information relating to system description and model integration for colloid release abstraction needed for a potential license application, but has a reasonable approach to do so by the time of license application, based on DOE agreements to provide additional documentation.[27]

The technical basis for selecting radionuclides for release modeling through reversible and irreversible colloidal attachment is not transparent and traceable in all cases. The Evolution of the Near-Field Environment Technical Exchange agreements matrix[28] states this issue is resolved in Section 3.5.6.1 of CRWMS M&O (2000m); the relevant section is actually in the Total System Performance Assessment–Site Recommendation report (CRWMS M&O, 2000t). This discussion does not address the possibility that waste form colloids (irreversible attachment) could significantly transport radioelements other than plutonium and americium, despite observations of other elements, such as uranium and thorium, irreversibly attached on colloids in waste corrosion tests (CRWMS M&O, 2000s). In addition, the argument neglects the potential for a contribution to release by the reversible colloid attachment of less sorbing radioelements such as neptunium and uranium. These issues also are not addressed adequately in the analysis and model report (CRWMS M&O, 2000r). DOE agreed[29] to provide the technical basis for selection of radionuclides released and transported via colloids in the total system performance assessment.

The technical basis for the exclusion of irreversible radionuclide attachment onto spent nuclear fuel colloids is not adequate. It is noted in CRWMS M&O (2000r) that the lack of observed attachment of this type may be an effect of the spent nuclear fuel test configuration, and that fewer data were obtained from the commercial spent nuclear fuel testing than from the high-level waste glass testing. In addition, CRWMS M&O (2000r) discusses the possibility that a plutonium-rich alteration layer on corroded spent nuclear fuel may be released by spallation, though this has not yet been observed. According to NRC (2000a, p. 224), several reports discuss evidence for irreversible plutonium attachment to corrosion product colloids. For other waste types such as DOE spent nuclear fuel, DOE needs to either screen out colloid-associated radionuclide release or develop modeling approaches for them. In CRWMS M&O (2000r), N-Reactor fuel is specifically discussed as requiring an assessment of importance to performance and possible inclusion in the abstraction. DOE agreed[30] to provide the technical basis for selection of waste forms for which irreversible colloidal release is modeled.

According to CRWMS M&O (2000m), only plutonium, americium, thorium, and protactinium are modeled as colloidally released, but other radioelements are included in unsaturated and

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[27]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[28]Ibid.

[29]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7, 2000)." Letter (December 12) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[30]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

saturated zone transport models. This apparent lack of model integration may be justified but should be clarified. In addition, confusion exists among the various reports cited in this section regarding the disposition of specific radioelements in colloid modeling. DOE agreed[31] to provide the technical basis for selection of radionuclides released and transported via colloids in the total system performance assessment.

3.3.4.4.8.2 Data Are Sufficient for Model Justification

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the effect of colloid release on radionuclide solubility limits with respect to sufficient data for model justification.

Site- and waste-specific data on colloid release parameters are not sufficient. For example, DOE needs to more strongly support the maximum concentration of colloidal waste form plutonium (i.e., $CRN_{coll,wf,irrev,max} = 6 \times 10^{-8}$ M) (CRWMS M&O, 2000r), which is constrained by the results of experiments on only one high-level waste glass sample. No basis is provided for the assertion in CRWMS M&O (2000r) that the values for the concentration range of iron oxide corrosion product colloids ($M_{coll,FeOx,max}$ and $M_{coll,FeOx,min}$) are reasonable and conservative. CRWMS M&O (2000r) asserts that plutonium and americium behave similarly enough during high-level waste glass colloid irreversible attachment that a constant concentration ratio may be assumed, however, supporting test data are not described or cited. Similarly, the colloid analysis and model reports do not demonstrate that the samples studied are sufficiently representative of the range of waste types. DOE agreed[32] to demonstrate that colloidal release and transport model parameters are sufficiently supported.

3.3.4.4.8.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the effect of colloid release on radionuclide solubility limits with respect to the characterization and propagation of data uncertainty through the model abstraction.

DOE has not demonstrated that its selection of colloid release model parameters (e.g., K_c) bound the uncertainty associated with data limitations, the special chemical environment in the waste package, and possible coupled thermal-hydrological-chemical processes (NRC, 2000a). For example, laboratory data were obtained for a limited range of chemical and thermal conditions {e.g., at 90 °C [194 °F]} (CRWMS M&O, 2000s), and it is not clear that adopted parameters, such as colloidal plutonium concentration, reflect the associated uncertainties. As

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[31]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7, 2000)." Letter (December 12) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[32]Ibid.

another example, the maximum colloidal plutonium concentration of 6×10^{-8} M, as measured during corrosion tests, is also the maximum value used in the model abstraction (CRWMS M&O, 2000r). No additional uncertainty is reflected in this value. In modeling chemical effects on colloid concentrations and radionuclide attachment, the DOE considers only pH and ionic strength; potential uncertainties associated with neglecting other effects are not explicitly addressed. DOE agreed[33] to demonstrate that colloidal release and transport model parameters are suitably bounding.

3.3.4.4.8.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the effect of colloid release on radionuclide solubility limits with respect to the characterization and propagation of model uncertainty through the model abstraction.

Section 6.3.4.6 of CRWMS M&O (2000m) describes implementation for codisposal packages of the colloid release and invert transport model abstractions in Total System Performance Assessment–Site Recommendation. The following issues concerning the characterization and propagation of uncertainty arising from model abstraction were identified in this review.

- Calculation of the concentration of iron (hydr)oxide colloids available to sorb radionuclides (CRWMS M&O, 2000m, p. 326) depends on pH and ionic strength. Determination of iron (hydr)oxide colloid stability involves comparison of solution pH and ionic strength against a plot of regions of colloid stability and instability (CRWMS M&O, 2000r, Figure 11). As demonstrated in the model verification discussion and in Figure 6-144 of CRWMS M&O (2000m), small changes in pH or ionic strength in the ranges plausible for in-package conditions can result in an abrupt change in modeled iron (hydr)oxide colloid concentration by a factor of 1,000; there are no intermediate values. This result demonstrates marked sensitivity to the adopted stability boundaries and to modeled solution parameters unlikely to be simulated accurately or precisely. In the example case of a codisposal package, the minimum iron (hydr)oxide colloid concentration results for the waste package, that is, far less radionuclide can be mobilized by reversible attachment. DOE should perform analyses to show if this high model sensitivity—which is not reflected in model uncertainty—has implications for modeled dose.

- Similar model sensitivity exists for the pH dependence of stability of waste form colloids. A functional relationship defines the variation of waste form colloid concentration with ionic strength for the range 0.01–0.05 M, but, at ionic strength above 0.05 M, the concentration drops abruptly by three orders of magnitude to 10^{-11} M (CRWMS M&O,

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[33]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7, 2000)." Letter (December 12) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

2000r, Figure 13). In-package and invert chemistry models suggest this sensitivity will not affect results because ionic strength stays below 0.05 M after waste package failure (CRWMS M&O, 2000m). A stability boundary in ionic strength versus pH space, however, introduces an abrupt concentration boundary (CRWMS M&O, 2000r, Figure 12), which is revealed in a model result that shows a one-time step drop by a factor of 6,000 in concentration of irreversibly bound plutonium (CRWMS M&O, 2000a, Figure 6-139). Although the concentration then rebounds to near maximum value for the remainder of the model, high sensitivity to modeled pH (which is uncertain) is evident. DOE should perform analyses showing if this high model sensitivity—which is not reflected in model or parameter uncertainty—has implications for modeled dose.

- The modeled concentration of groundwater colloids (which facilitate reversible attachment) also shows an abrupt pH dependence, using the same stability fields as for waste form colloids (CRWMS M&O, 2000r, Figure 12). In this case, an abrupt change in concentration by a factor of 10^4 can result from a shift in ionic strength versus pH space; behavior similar to that of waste form colloids results (CRWMS M&O, 2000m). DOE should perform analyses showing if this high model sensitivity—which is not reflected in model or parameter uncertainty—has implications for modeled dose.

DOE agreed[34] to demonstrate that colloidal release and transport model parameters are suitably bounding. In addition, DOE agreed[35] to provide a sensitivity analysis of these potentially abrupt changes in modeled colloid concentration.

3.3.4.4.8.5 Model Abstraction Output Is Supported by Objective Comparisons

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the effect of colloid release on radionuclide solubility limits with respect to model abstraction output being supported by objective comparisons.

Section 6.3.4.6 of CRWMS M&O (2000m) discusses model verification for the colloid release and invert transport model abstractions for a codisposal package. In one case—calculation of invert iron (hydr)oxide colloid concentration—an independent analysis did not agree with the reports results. It may be deduced from Figure 6-143 of CRWMS M&O (2000m) that modeled invert water pH after 100,000 years is approximately 7.7. This pH, combined with ionic strength of 0.01 M places this water within the iron (hydr)oxide colloid stability field of Figure 11 of CRWMS M&O (2000r), contradicting the Total System Performance Assessment–Site Recommendation verification result that these colloids will be at their minimum concentration of

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[34]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7, 2000)." Letter (December 12) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[35]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC, 2001.

0.001 mg/L (CRWMS M&O, 2000m, p. 332). That the invert water lies in the stability field is also shown in Figure 6-143 of CRWMS M&O (2000m) by the location of the ionic strength curve below the curve for the stability boundary equation, 0.02 • pH + 0.17 (note that the equation is incorrect in the figure legend). Thus, the iron (hydr)oxide colloid concentration should be at its maximum value of 1 mg/L, three orders of magnitude higher than the conclusion of CRWMS M&O (2000m). That invert ionic strength is above the 0.02 • pH – 0.17 curve is irrelevant because pH is not between 9 and 11.

The inconsistent results could be related to a potential problem noted with the definition of Condition B in CRWMS M&O (2000m). As stated on page 326, Condition B in the abstraction is equal to one if ionic strength in the waste package is greater than *either* of two values calculated to represent portions of the stability boundaries linear with pH. It is more correct to say that the condition is one if ionic strength is greater than the value calculated using the particular equation for the relevant pH range of 6 to 8 or 9 to 11 (CRWMS M&O, 2000r, Figure 11). Ionic strength may be below one calculated value and above another and still be in the region of stability. The way Condition B is described on page 326, a combination of Conditions A and B both being equal to one is not sufficient to conclude the colloids are unstable. If the description of Condition B in CRWMS M&O (2000m) is as intended, DOE should correct the description so that erroneous stability conclusions will not be drawn. Such a conclusion could lead to a three-order-of-magnitude underestimate of iron (hydr)oxide colloid concentration. DOE agreed[36] to provide revised documentation on the effects of in-package water chemistry on colloid release.

3.3.4.4.9 Engineered Barrier Subsystem Flow and Transport

The release of radionuclides from the engineered barrier subsystem can occur primarily through transport either as dissolved constituents in water or as bound to colloids. Both dissolved and colloidal radionuclides can diffuse and advect through the water within the waste package and through the invert below the waste packages. Before radionuclide transport can occur, however, the waste package must be breached, the cladding must fail (for commercial spent nuclear fuel packages), and the waste forms must degrade. Thus, radionuclide transport from the engineered barrier subsystem into the unsaturated zone is dependent on a complex series of events in the potential repository (CRWMS M&O, 2000u). Several factors will affect the mobilization and transport of radionuclides through the engineered barrier subsystem: (i) drip shield performance, (ii) waste package performance, (iii) cladding performance, (iv) waste form dissolution rates, (v) entry and movement of water through the waste package, (vi) solubility limit for each radionuclide, (vii) radionuclide transportation through and out of the waste package, (viii) radionuclide transportation through the invert, and (ix) radionuclide transportation via colloids.

The DOE conceptual model for engineered barrier subsystem flow abstraction relies on several key elements. Flow through the engineered barrier subsystem is abstracted to a

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[36]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

one-dimensional network of flow pathways, and the flow system is assumed to be quasi-steady (i.e., fluid immediately flows through the system and does not accumulate within the engineered barrier subsystem). The abstraction also uses a flow-through model for the waste package (i.e., fluid does not accumulate in the waste package). The type, number, and timing of breaches in the drip shield and waste package are predicted by the WAPDEG code. Separation of the drip shields in response to rock fall, seismic events, or thermal expansion is assumed not to occur.

The DOE conceptual model for engineered barrier subsystem transport abstraction has several key elements. Advective transport of radionuclides may occur through patches and pits created by various corrosion mechanisms in the waste package. Patches can be created by general corrosion, and pits can be created by localized corrosion. Both patches and pits are conceptualized to have a large enough cross-sectional area to provide a pathway for advective flow and transport through the waste package. Radionuclides also can be transported by diffusion through any breach in the waste package (i.e., through stress corrosion cracks, patches, or pits).

DOE recognizes potentially large uncertainties in the response of a complex engineered barrier subsystem through long periods of time. To bound the uncertainties in the model parameters used in its abstraction of flow and radionuclide transport processes in the engineered barrier subsystem, DOE made several assumptions, as discussed in the analysis and model report (CRWMS M&O, 2000u). These assumptions include

- The fluid flux is assumed to pass through any patch or stress corrosion crack on the surface of the waste package, independent of its location on the upper or lower surface of the waste package. DOE states this is a conservative assumption for the patches and pits on the lower half of the waste package, where little inflow is expected to occur, and for flow-through stress corrosion cracks because fluid is unlikely to reach any stress corrosion cracks on the upper half of the lid.

- The fluid flux onto the closure lid of the waste package (where stress corrosion cracks can occur) is reasonably bounded by assuming the waste package is tilted at the maximum angle possible beneath the drip shield.

- All fluid that flows as a film on the closure lid of the waste package flows through a stress corrosion crack, if present.

- Radionuclide transport through a stress corrosion crack is assumed limited to diffusive transport through a thin, continuous film that is always present (meaning radionuclide diffusion out of the waste package is possible as soon as a stress corrosion crack forms on the canister lid). Advective flux through a stress corrosion crack is considered negligible because of the small cross-sectional area of the stress corrosion crack.

- Advective transport occurs only in the vertical direction and is always downward.

- The effects of longitudinal and transverse dispersion are ignored.

- The diffusion coefficient of all relevant radionuclides is bounded by the self-diffusion coefficient for water.
- Sorption of dissolved radionuclides to stationary phases in the waste package and invert is negligible.
- The flux of water into the waste package is equal to the flux out of the waste package and into the invert (flow-through system).

The staff review regarding the DOE abstraction of engineered barrier subsystem flow and transport follows.

3.3.4.4.9.1 System Description and Model Integration Are Adequate

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess engineered barrier subsystem flow and transport with respect to system description and model integration.

Sufficient description is provided on the approach and technical basis for the abstraction of engineered barrier subsystem flow and transport and the integration into total system performance assessment analyses. The assumptions are clearly stated, used consistently, and are technically defensible. In general, important design features, processes, and couplings are incorporated or bounded.

DOE has not provided a satisfactory evaluation of floor buckling and the potential effects on the rates of water flow and radionuclide release through the invert. DOE proposed to screen out floor buckling (CRWMS M&O, 2000u,v) based on results presented in CRWMS M&O (1998c), which indicate floor heave from thermal-mechanical effects would not exceed approximately 10 mm. The NRC staff (2000c),[37] however, have expressed concern about the appropriateness of the thermal-mechanical parameters used in the DOE assessment of rock-mass behavior around the emplacement drifts (CRWMS M&O, 1998c, 2000w). Stress conditions induced by thermal loading would favor reverse-faulting-style slip on subhorizontal fractures beneath the floor of the emplacement drifts (Ofoegbu, 2001). The occurrence of such fracture slip, which depends on thermal-load magnitude and rock-mass, thermal-mechanical properties, would lead to buckling of the emplacement-drift floor and a change in the hydrological characteristics of the invert and underlying rock. DOE agreed[38] to revise its analysis of thermal-mechanical behavior around the emplacement drifts to include site-specific, thermal-mechanical properties and spatial and temporal variations of the property values.

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[37]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Thermal Effects on Flow (January 8–9, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[38]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Repository Design and Thermal-Mechanical Effects (February 6–8, 2001)." Letter (February 28) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

3.3.4.4.9.2 Data Are Sufficient for Model Justification

The DOE abstraction of engineered barrier subsystem flow and transport relies on input from other total system performance assessment abstractions. Thus, staff evaluation with respect to sufficient data for model justification is discussed in Sections 3.3.6 and 3.3.7.

3.3.4.4.9.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

DOE made several assumptions in its abstraction of flow and radionuclide transport processes in the engineered barrier subsystem to bound the uncertainties in the model parameters. The DOE approach to incorporating data uncertainty by making conservative assumptions in its abstraction is reasonable.

3.3.4.4.9.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess engineered barrier subsystem flow and transport with respect to the characterization and propagation of model uncertainty through the model abstraction.

DOE made several assumptions in its abstraction of flow and radionuclide transport processes in the engineered barrier subsystem to bound the uncertainties in the conceptual models. The DOE approach to incorporating model uncertainty by making conservative assumptions in its abstraction is reasonable. DOE, however, has not considered the potential effect of floor buckling on the rates of water flow and radionuclide release through the invert. Stress conditions induced by thermal loading could result in reverse-faulting-style slip on subhorizontal fractures beneath the floor of the emplacement drifts, which would lead to buckling of the emplacement-drift floor and a change in the hydrological characteristics of the invert and underlying rock. DOE agreed[39] to revise its analysis of thermal-mechanical behavior around the emplacement drifts to include site-specific, thermal-mechanical properties and spatial and temporal variations of the property values.

3.3.4.4.9.5 Model Abstraction Output Is Supported by Objective Comparisons

The DOE abstraction of engineered barrier subsystem flow and transport relies on input from other total system performance assessment abstractions. Thus, staff evaluation of DOE information to support its abstraction of engineered barrier subsystem flow and transport is discussed in Sections 3.3.6 and 3.3.7.

[39]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Repository Design and Thermal-Mechanical Effects (February 6–8, 2001)." Letter (February 28) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

3.3.4.4.10 Near-Field Criticality

Using the Total System Performance Assessment–Site Recommendation (based on no waste package breach or failure at any time during the first 10,000 years of postclosure), DOE screened the occurrence of nuclear criticality for commercial spent nuclear fuel for normal conditions and seismic events (CRWMS M&O, 2000r). The NRC concerns regarding the DOE screening argument for nuclear criticality are delineated in Section 3.2.2. As agreed during the DOE and NRC Technical Exchange on Criticality,[40] DOE agreed to perform a what-if analysis, to analyze the probability and consequences of a criticality event given an early waste package failure using the topical report methodology.

This topical report (DOE, 1998) describes the methodology to be used to assess the probability and consequences of criticality events within the repository system for all fuel types, except Naval fuels, in all locations (i.e., in package, near field, and far field). In this topical report, DOE proposed to use a systematic approach for identifying scenarios and configurations that could result in a criticality event. Included are the configurations within which fissile material is precipitated in the vicinity of the waste package inside the drift or within the invert on release from the waste package.

NRC reviewed this topical report and documented the results of its review in a safety evaluation report (NRC, 2000c). This safety evaluation report contains 28 open items on the methodology, which, when closed, will document NRC acceptance of the proposed methodology to address criticality in the repository system. According to an agreement made during the DOE and NRC Technical Exchange on Criticality,[41] DOE provided NRC with Revision 1 of this topical report, which is intended to address 27 of the open items (DOE, 2000). The remaining open item on burnup measurements was discussed at the DOE and NRC Technical Exchange on Pre-Closure Safety.[42] On December 10, 2001, the NRC staff notified DOE that NRC accepted Revision 1 of the topical report for detailed technical review. If found acceptable, NRC will have confidence that DOE will be able to address the effects of criticality on the performance of the repository system in any potential license application, even if DOE is not able to support its arguments for screening criticality from the total system performance assessment.

Criticality in Naval fuel has been addressed in a separate addendum, which, for security reasons, is not discussed in this report.

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[40]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Criticality (October 23–24, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[41]Ibid.

[42]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

3.3.4.4.10.1 System Description and Model Integration Are Adequate

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess near-field criticality with respect to system description and model integration.

In Revision 0 of the topical report (DOE, 2000), identified five near-field (NF) scenarios with potentials for a criticality event: (i) NF–1: solute transport of fissile material from the waste package and accumulation in the invert; (ii) NF–2: slurry transport of fissile material from the waste package and accumulation on the invert; (iii) NF–3: colloidal transport of fissile material from the waste package and accumulation in the invert; (iv) NF–4: collection of water ponds in drift, degradation of waste package and waste form, and accumulation of fissile material in clays at the bottom of the drift; and (v) NF–5: collection of water ponds in drift, degradation of waste package, and settlement of intact waste package in pond. All scenarios require—in addition to release and transport of fissile material—a degree of separation of fissile material from neutron absorbers, and mechanisms for this process are, therefore, included. Each scenario encompasses one or more configuration classes, which further specify the processes and settings that define the potentially critical configuration. For example, scenario NF–3 includes three configuration classes—NF–3a, NF–3b, and NF–3c—that specify whether colloids accumulate in waste package corrosion products, invert fractures, or degraded concrete.

In its evaluation of the near-field criticality scenarios, except for the igneous-activity-induced criticality scenario, the staff found that—contingent on the topical report revisions promised in the DOE responses to the request for additional information—DOE comprehensively identified generic and site-specific near-field criticality scenarios. With respect to igneous-activity criticality, DOE provided an approach in the topical report (DOE, 2000, Revision 01, Section 3.3.4) for identifying potential critical configurations following a volcanic event. The staff will evaluate the DOE approach and document the results in the amendment to the safety evaluation report. On the other hand, DOE screened the occurrence of igneous-induced criticality based on a low probability of formation of a critical configuration. The basis for this screening was documented in the probability of criticality within 10,000 years calculation report (CRWMS M&O, 2000h). Staff concerns with this report are discussed in Section 3.2.2.

For identifying near-field criticality configuration within each scenario, DOE proposed to quantify parameter ranges for each configuration class. Formulation of a configuration is based on parameters consistent with repository features, taking into consideration current design and site characterization. Examples include drift floor materials and host rock fracture density. DOE proposed a six-step determination for formulating a configuration: (i) fissile material source term using information generated by waste package internal configuration; (ii) water flow rates and patterns; (iii) sorption along flow paths; (iv) mineral precipitates along flow paths; (v) alternate paths when primary rock fractures are filled, including possible coalescence of contaminant plumes from several waste packages; and (vi) reaction products resulting from the plume encountering a reducing zone.

In performing the previous steps, DOE proposed to use a geochemistry transport and a geochemistry computer code to calculate fissile material accumulation external to the waste package. These models will include relevant geochemical processes and will incorporate transport. DOE indicated that the geochemical transport code PHREEQC, supplemented by a modification of EQ3/6, will be used.

Another component of near-field configuration modeling will be the configuration generator code. This code will provide bookkeeping for the transport between sites of application of a detailed geochemistry code and, in some situations, provide more rapid calculation when the detailed geochemistry code results can be used to develop heuristic models for the most significant ions for a few solution parameters.

The NRC staff accepted the DOE use of a geochemistry-transport code, a geochemistry code used in a mode that simulates transport, or both, to calculate fissile material accumulation external to the waste package, provided these are properly applied and validated.

3.3.4.4.10.2 Data Are Sufficient for Model Justification

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess near-field criticality with respect to sufficient data for model justification.

Data sufficiency for calculation of k_{eff} for near-field configurations is addressed in the safety evaluation report (NRC, 2000c). For example, one open item states that DOE needs to use the cross-sectional data corresponding to the temperature for the waste package or critical benchmarks. If a critical configuration is credible while near-field temperatures are elevated, this item would also apply to external accumulations. Revision 1 of the topical report is intended to address the open items (DOE, 2000). NRC review of this revision of the topical report will be documented in a revision to the safety evaluation report on the topical report. Additionally, DOE indicated at the DOE and NRC Technical Exchange on Criticality that additional data would be located in the validation reports for computer codes that will be used in the criticality modeling. DOE agreed[43] to provide these validation reports to the NRC before submission of a license application for the proposed Yucca Mountain repository.

With respect to geochemical models for external near-field critical configurations, NRC (2000c) did not address data sufficiency because the review was focused on methodology. DOE stated that geochemical transport modeling for external near-field critical configurations will be consistent with models supporting total system performance assessment. Therefore, data sufficiency for models of near-field release and transport for total system performance assessment (see elsewhere in this section, as well as Section 3.3.9) will, in general, ensure data sufficiency for near-field criticality models. Some exceptions to this correspondence exist. For example, the DOE model validation report on external accumulation relies on satisfactory

[43]Schlueter, J.R. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Criticality (October 23–24, 2000).” Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

characterization of fracture and lithophysae distributions immediately below the drift (Bechtel SAIC Company, LLC, 2001a). The aspects of this report have not yet been reviewed.

3.3.4.4.10.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess near-field criticality with respect to the characterization and propagation data uncertainty through the model abstraction.

Data uncertainty issues with respect to calculation of k_{eff} for near-field configurations are addressed in the safety evaluation report (NRC, 2000c). For example, two relevant open items are include (i) DOE must include the cross dependency of configuration parameters for k_{eff} regression equations and (ii) DOE must include the isotopic bias and uncertainty in developing the critical limit. Revision 1 of the topical report (DOE, 2000) is intended to address the open items including those related to data uncertainty. The NRC review of this revision of the topical report will be documented in a revision to the safety evaluation report. In the DOE and NRC Technical Exchange on Criticality, DOE indicated[44] that quantification of data uncertainty would be located in the validation reports for computer codes that will be used in the criticality modeling. DOE agreed to provide these validation reports to NRC before submission of any license application for the proposed Yucca Mountain repository (NRC, 2000a).

With respect to geochemical models for external near-field critical configurations, the topical report safety evaluation report (NRC, 2000c) did not address data uncertainty because the review was focused on methodology. It is expected that data uncertainty issues not covered in other geochemical modeling applications in total system performance assessment will be addressed in future criticality reports. Reports currently available (Bechtel SAIC Company, LLC, 2001a,b) have not yet been reviewed with respect to this issue.

3.3.4.4.10.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess near-field criticality with respect to the characterization and propagation of model uncertainty through the model abstraction.

Model uncertainty issues with respect to calculation of k_{eff} for near-field configurations are addressed in the safety evaluation report (NRC, 2000c). The relevant open item states: DOE must demonstrate the adequacy of using one-dimensional calculations to capture three-dimensional neutron spectrum effects in their point-depletion calculation or use

[44]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Criticality (October 23–24, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

two/three-dimensional calculations for determining the neutron spectra during the depletion cycles to be used in the depletion analyses. This open item is intended to be addressed in Revision 1 of the topical report (DOE, 2000). NRC review of this revision of the topical report will be documented in a revision to the safety evaluation report. In the DOE and NRC Technical Exchange on Criticality, DOE indicated the validation reports will support use of the inventory computer code that will be used in the criticality modeling. DOE[45] agreed to provide these validation reports to NRC prior to submission of any license application for the proposed Yucca Mountain repository.

With respect to geochemical models for external near-field critical configurations, the topical report safety evaluation report (NRC, 2000c) did not explicitly evaluate model uncertainty issues because the review was focused on methodology. It is expected that model uncertainty issues not covered in other geochemical modeling applications in total system performance assessment will be addressed in future criticality reports. Reports currently available (Bechtel SAIC Company, LLC, 2001a,b) have not yet been reviewed with respect to this issue.

3.3.4.4.10.5 Model Abstraction Output Is Supported by Objective Comparisons

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.4.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess near-field criticality with respect to model abstraction output being supported by objective comparisons.

Model verification issues with respect to calculation of k_{eff} for near-field configurations are addressed in the safety evaluation report (NRC, 2000c). Open items include: (i) DOE must present a validation methodology or work scope for external criticality models, and (ii) DOE must verify the regression equation or look-up table for all ranges of configuration and waste form parameters affecting $k_{eff,}$. Revision 1 of the topical report is intended to address the open items (DOE, 2000), including those related to model support. NRC review of Revision 1 of the topical report will be documented in an amendment to the NRC safety evaluation report. In the DOE and NRC Technical Exchange on Criticality, DOE indicated justification of the models used in the criticality analyses would be located in the validation reports for the inventory and neutronics computer codes. DOE agreed[46] to provide these validation reports to NRC prior to submission of a license application for the proposed Yucca Mountain repository (NRC, 2000a).

DOE provided two of these reports—the Geochemistry Model Validation Report: Material Degradation and Release Model (Bechtel SAIC Company, LLC, 2001b) and Geochemistry Model Validation Report: External Accumulation Model (Bechtel SAIC Company, LLC, 2001a). These two report are currently being reviewed by staff.

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[45]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Criticality (October 23–24, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[46]Ibid.

3.3.4.5 Status and Path Forward

Table 3.3.4-1 provides the status of all key technical issue subissues referenced in Section 3.3.4.2 and the related DOE and NRC agreements for the Radionuclide Release Rates and Solubility Limits Integrated Subissue. The agreements listed in the table are associated with one or all five generic acceptance criteria discussed in Section 3.3.4.4. Note the status and detailed agreements (or path forward) pertaining to all the key technical issue subissues are provided in Table 1.1-3 and Appendix A.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (through specified testing, analyses, and the like), acceptably addresses the NRC questions so that no information beyond that provided, or agreed to, will likely be required at the time of a potential license application.

Table 3.3.4-1. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreements*
Container Life and Source Term	Subissue 3—The Rate at Which Radionuclides in Spent Nuclear Fuel Are Released from the Engineered Barrier Subsystem Through the Oxidation and Dissolution of Spent Nuclear Fuel	Closed-Pending	CLST.3.01 through CLST.3.10
	Subissue 4—The Rate at Which Radionuclides in High-level Waste Glass Are Leached and Released from the Engineered Barrier Subsystem	Closed-Pending	CLST.4.01 through CLST.4.11
	Subissue 5—The Effect of In-package Criticality on Waste Package and Engineered Barrier Subsystem Performance	Closed-Pending	CLST.5.01 CLST.5.04 CLST.5.05 CLST.5.07
	Subissue 6—The Effects of Alternate Engineered Barrier Subsystem Design Features on Container Lifetime and Radionuclide Release from the Engineered Barrier Subsystem	Closed-Pending	None
Evolution of the Near-Field Environmental	Subissue 3—Effect of Coupled Thermal-Hydrologic-Chemical Processes on the Chemical Environment for Radionuclide Release	Closed-Pending	ENFE.3.03 ENFE.3.04 ENFE.3.05

Table 3.3.4-1. Related Key Technical Issue Subissues and Agreements (continued)			
Key Technical Issue	Subissue	Status	Related Agreements*
Evolution of the Near-Field Environmental	Subissue 4—Effects of Coupled Thermal-Hydrologic-Chemical Processes on Radionuclide Transport Through Engineered and Natural Barriers	Closed-Pending	ENFE.4.06
	Subissue 5—Effects of Coupled Thermal-Hydrologic-Chemical Processes on Potential Nuclear Criticality in the Near Field	Closed-Pending	None
Total System Performance Assessment and Integration	Subissue 1—System Description and Demonstration of Multiple Barriers	Closed-Pending	None
	Subissue 2—Scenario Analysis and Event Probability	Closed-Pending	TSPAI.2.01 TSPAI.2.02
	Subissue 3—Model Abstraction	Closed-Pending	TSPAI.3.14 through TSPAI.3.17 TSPAI.3.42
	Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards	Closed-Pending	None

*Related DOE and NRC agreements are associated with one or all five generic acceptance criteria.

Note: Key Technical Issue Agreement GEN.1.01 pertains to multiple integrated subissues, as well as some specific issues related to this integrated subissue.

3.3.4.6 References

Advocat, T., J.L. Crovisier, E. Vernaz, G. Ehret, and H. Charpentier. "Hydrolysis of R717 Nuclear Waste Glass in Dilute Media: Mechanisms and Rate: A Function of pH." Scientific Basis for Nuclear Waste Management XIV. Symposium Proceedings 212. T.J. Abrajano and L.H. Johnson, eds. Pittsburgh, Pennsylvania: Materials Research Society. pp. 57–64. 1991.

Bechtel SAIC Company, LLC. "Geochemistry Model Validation Report: External Accumulation Model." ANL–EBS–GS–00002. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2001a.

———. "Geochemistry Model Validation Report: Material Degradation and Release Model." ANL–EBS–GS–00001. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2001b.

Burns, W.G., A.E. Hughes, J.A.C. Marples, R.S. Nelson, and A.M. Stoneham. "Effects of Radiation on the Leach Rates of Vitrified Radioactive Waste." *Journal of Nuclear Materials*. Vol. 107. pp. 245–270. 1982.

Cavanaugh, M.A., J.A. Kargol, J. Nickerson, and N.F. Fiore. “The Anodic Dissolution of a Ni-base Superalloy.” *Corrosion.* Vol. 39, No.4. pp. 144–150. 1983.

Chan, K.S. “A Micromechanical Model for Predicting Hydride Embrittlement in Nuclear Fuel Cladding Material.” *Journal of Nuclear Materials*. Vol. 227. pp. 220–236. 1996.

CRWMS M&O. “U.S. Department of Energy Spent Nuclear Fuel Information in Support of TSPA-VA.” DOE/SNF/REP–027. Idaho Falls, Idaho: Idaho National Engineering and Environmental Laboratory. 1998a.

———. “Total System Performance Assessment–Viability Assessment (TSPA-VA) Analyses Technical Basis Document.” B00000000–01717–4301. Revision 01. Las Vegas, Nevada: CRWMS M&O. 1998b.

———. “Repository Ground Support Analysis for Viability Assessment.” BCAA00000–01717–0200–00004. Revision 01. Las Vegas, Nevada: CRWMS M&O. 1998c.

———. “Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations.” TDR–WIS–RL–000001. Revision 4 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000a.

———. “Waste Form Degradation Process Model Report.” TDR–WIS–MD–000001. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000b.

———. “Inventory Abstraction.” ANL–WIS–MD–000006. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000c.

———. “Summary of In-Package Chemistry for Waste Forms.” ANL–EBS–MD–000050. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000d.

———. “In-Package Chemistry Abstraction.” ANL–EBS–MD–00037. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000e.

———. “Clad Degradation—Summary and Abstraction.” ANL–WIS–MD–000007. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000f.

———. “Initial Cladding Condition.” ANL–EBS–MD–000048. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000g.

———. “Clad Degradation—FEPs Screening Arguments.” ANL–WIS–MD–000008. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000h.

———. “Clad Degradation—Local Corrosion of Zirconium and Its Alloys Under Repository Conditions.” ANL–EBS–MD–000012. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000i.

———. “Hydride Related Degradation of SNF Cladding Under Repository Conditions.” ANL–EBS–MD–000011. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000j.

———. “CSNF Waste Form Degradation Summary Abstraction.” ANL–EBS–MD–000015. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000k.

———. “Environment on the Surfaces of the Drip Shield and Waste Package Outer Barrier.” ANL–EBS–MD–000001. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000l.

———. “Total System Performance Assessment (TSPA) Model for Site Recommendation.” MDL–WIS–PA–000002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000m.

———. “Defense High-Level Waste Glass Degradation.” ANL–EBS–MD–0016. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000n.

———. “Defense High-Level Waste Glass Degradation.” ANL–EBS–MD–0016. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000o.

———. “Summary of Dissolved Concentration Limits.” ANL–WIS–MD–000010. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000p.

———. “Pure Phase Solubility Limits—LANL.” ANL–EBS–MD–000017. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000q.

———. “Waste Form Colloid-Associated Concentration Limits: Abstraction and Summary.” ANL–WIS–MD–000012. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000r.

———. “Colloid-Associated Radionuclide Concentration Limits.” ANL–EBS–MD–000020. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000s.

———. “Total System Performance Assessment (TSPA) for the Site Recommendation.” MDL–WIS–PA–000002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000t.

———. “EBS Radionuclide Transport Abstraction.” ANL–WIS–PA–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000u.

———. “EBS FEPs/Degradation Modes Abstraction.” ANL–WIS–PA–000002. Revision 00 ICN 1. Las Vegas, Nevada: CRWMS M&O. 2000v.

———. “Ground Control for Emplacement Drifts for SR.” ANL–EBS–GE–000002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000w.

———. “In-Package Chemistry for Waste Forms.” ANL–EBS–MD–000056. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2001a.

———. “In-Package Chemistry Abstraction.” ANL–EBS–MD–00037. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2001b.

DOE. “Viability Assessment of a Repository at Yucca Mountain. Vol. 3: Total System Performance Assessment.” DOE/RW–0508/3. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 1998.

———. “Draft Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca Mountain, Nye County, Nevada.” DOE/EIS–0250D. Las Vegas, Nevada: DOE, CRWMS M&O. 1999.

———. “Disposal Criticality Analysis Methodology Topical Report.” YMP/TR–004Q. Revision 01. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 2000.

Electric Power Research Institute. “Yucca Mountain Performance Assessment, Phase 4.” Palo Alto, California: Electric Power Research Institute. 1998.

Kirchheim, R., B. Heine, H. Fischmeister, S. Hofmann, H. Knote, and U. Stoltz. “The Passivity of Iron-Chromium Alloys.” *Corrosion Science*. Vol. 30, No. 7. pp. 899–917. 1989.

Knauss, K.G., W.L. Bourcier, K.D. McKeegan, C.I. Merzbacher, S.N. Nguyen, F.J. Ryerson, D.K. Smith, and H.C. Weed. “Dissolution Kinetics of a Simple Analogue Nuclear Waste Glass As a Function of pH, Time, and Temperature.” Scientific Basis for Nuclear Waste Management XIII. V.M. Oversby and P.W. Brown, eds. Symposium Proceedings. Warrendale, Pennsylvania: Materials Research Society. pp. 176, 371–381. 1990.

Liang, L. and J.J. Morgan. “Chemical Aspects of Iron Oxide Coagulation in Water: Laboratory Studies and Implications for Natural Systems.” *Aquatic Sciences*. Vol. 52. pp. 32–55. 1990.

Matsuo, Y. “Thermal Creep of Zircaloy-4 Cladding Under Internal Pressure.” *Journal of Nuclear Science and Technology*. Vol. 24. pp. 111–119. 1987.

Mohanty, S., R. Codell, R.W. Rice, J. Weldy, Y. Lu, R.M. Byrne, T.J. McCartin, M.S. Jarzemba, and G.W. Wittmeyer. “System-Level Repository Sensitivity Analyses Using TPA Version 3.2 Code.” CNWRA 99-002. San Antonio, Texas: CNWRA. 1999.

Mohanty, S., T.J. McCartin, and D.W. Esh. “Total-system Performance Assessment (TPA) Version 4.0 Code: Module Descriptions and User’s Guide.” San Antonio, Texas: CNWRA. 2000.

Northwood, D.O. and U. Kosasih. “Hydrides and Delayed Hydrogen Cracking in Zirconium and Its Alloys.” *International Metals Review*. Vol. 28, No. 2. pp. 92–121. 1983.

NRC. NUREG–1688, “NRC Sensitivity and Uncertainty Analyses for a Proposed High-Level Waste Repository at Yucca Mountain, Nevada, Using TPA 3.1.” Vol. 2. Washington, DC: NRC. March 1999.

———. “Issue Resolution Status Report, Key Technical Issue: Evolution of the Near-Field Environment.” Revision 3. Washington, DC: 2000a.

———. “Issue Resolution Status Report, Key Technical Issue: Total System Performance Assessment and Integration.” Revision 3. Washington, DC: NRC. 2000b.

———. “Safety Evaluation Report for Disposal Criticality Analysis Methodology Topical Report.” Revision 0. Washington, DC: NRC. 2000c.

———. “Issue Resolution Status Report, Key Technical Issue: Container Life and Source Term.” Revision 3. Washington, DC: NRC. 2001.

———. NUREG–1804, “Yucca Mountain Review Plan—Draft Report for Comment.” Revision 2. Washington, DC: NRC. March 2002.

Oak Ridge National Laboratory. “ORIGEN2.1: Isotope Generation and Depletion Code—Matrix Exponential Method.” C00371. Oak Ridge, Tennessee: Oak Ridge National Laboratory. 1983.

Ofoegbu, G.I. “Hydrological Implications of Thermally Induced Geomechanical Response at Yucca Mountain, Nevada.” Proceedings of the 38th U.S. Rock Mechanics Symposium. San Antonio, Texas: CNWRA. 2001.

Pan, Y.M., V. Jain, M. Bogart, and P. Deshpande. “Effect of Iron Chlorides on the Dissolution Behavior of Simulated High-Level Waste Glasses.” *Ceramic Transactions of the Environmental Issues and Waste Management Technologies in the Ceramic and Nuclear Industries*. Vol. 119. D. Spearing, et al., eds. Westerville, Ohio: American Ceramic Society. 2001.

Puls, M.P. “The Influence of Hydride Size and Matrix Strength on Fracture Initiation at Hydrides in Zirconium Alloys.” *Metallurgical Transactions A*. 19A. pp. 1,507–1,522. 1988.

Shoesmith, D.W. “Fuel Corrosion Processes Under Waste Disposal Conditions.” AECL–12034. Pinawa Manitoba, Canada: AECL Technologies, Inc. 1999.

Stout, R.B. and H.R. Leider. “Waste Form Characteristics Report.” UCRL–ID–108314. Version 1.3. Livermore, California: Lawrence Livermore National Laboratory. 1998a.

———. “Waste Form Characteristics Report.” UCRL–ID–132375, CD-ROM Version. Livermore, California: Lawrence Livermore National Laboratory. 1998b.

Tombacz, E., I. Abraham, M. Gilde, and F. Szanto. “The pH-Dependent Colloidal Stability of Aqueous Montmorillonite Suspensions.” *Colloids and Surfaces*. Vol. 49. pp. 71–80. 1990.

Wronkiewicz, D.J., J.K. Bates, T.J. Gerding, and E. Veleckis. “Uranium Release and Secondary Phase Formation During Unsaturated Testing of UO_2 at 90 °C.” *Journal of Nuclear Materials.* Vol. 190. pp. 107–126. 1992.

Wronkiewicz, D.J., C.R. Bradley, J.K. Bates, and L.M. Wang. "Effect of Radiation Exposure on SRL 131 Composition Glass in a Steam Environment." Scientific Basis for Nuclear Waste Management XVII. Symposium Proceedings 333. A. Barkatt and R.A. Van Konynenburg, eds. Pittsburgh, Pennsylvania: Materials Research Society. pp. 259–267. 1994.

Wronkiewicz, D.J., J.K. Bates, E.C. Buck, J.C. Hoh, J.W. Emery, and L.M. Wang. "Radiation Effects in Moist-Air Systems and the Influence of Radiolytic Products on Formation of Nuclear Waste Glass Corrosion." ANL–97/15. Argonne, Illinois: Argonne National Laboratory. 1997.

Yau, T.L. and R.T. Webster. "Corrosion of Zirconium and Hafnium." *Metals Handbook: Vol. 13—Corrosion Materials*. Park, Ohio: ASM International. 1987.

3.3.5 Climate and Infiltration

3.3.5.1 Description of Issue

The Climate and Infiltration Integrated Subissue addresses the near-surface hydrologic processes, such as precipitation, temperature, climate change, and rates of infiltration. Climate strongly influences the rates of shallow infiltration, which in turn, correlates with the amount of water entering the waste emplacement drifts. Relationship of this integrated subissue to other integrated subissues is depicted in Figure 3.3.5-1. The overall organization and identification of all the integrated subissues are depicted in Figure 1.1-2. The DOE description and technical basis for abstractions of climate and infiltration are documented in CRWMS M&O (2000a) and numerous supporting analysis and model reports. This section reviews the abstractions of climate and infiltration incorporated by DOE in its total system performance assessment. Portions of additional analysis and model reports are reviewed to the extent that they contain data or analyses that support the total system performance assessment abstractions for climate and infiltration.

3.3.5.2 Relationship to Key Technical Issue Subissues

The Climate and Infiltration Integrated Subissue incorporates subject matter previously captured in the following key technical issue subissues:

- Unsaturated and Saturated Flow Under Isothermal Conditions: Subissue 1—Climate Change (NRC, 1999)
- Unsaturated and Saturated Flow Under Isothermal Conditions: Subissue 2—Hydrologic Effects of Climate Change (NRC, 1999)
- Unsaturated and Saturated Flow Under Isothermal Conditions: Subissue 3—Present-Day Shallow Infiltration (NRC, 1999)
- Structural Deformation and Seismicity: Subissue 3—Fracturing and Structural Framework of the Geological Setting (NRC, 2000a)
- Total System Performance Assessment and Integration: Subissue 1—System Description and Demonstration of Multiple Barriers (NRC, 2000b)
- Total System Performance Assessment and Integration: Subissue 2—Scenario Analysis and Event Probability (NRC, 2000b)
- Total System Performance Assessment and Integration: Subissue 3—Model Abstraction (NRC, 2000b)
- Total System Performance Assessment and Integration: Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards (NRC, 2000b)

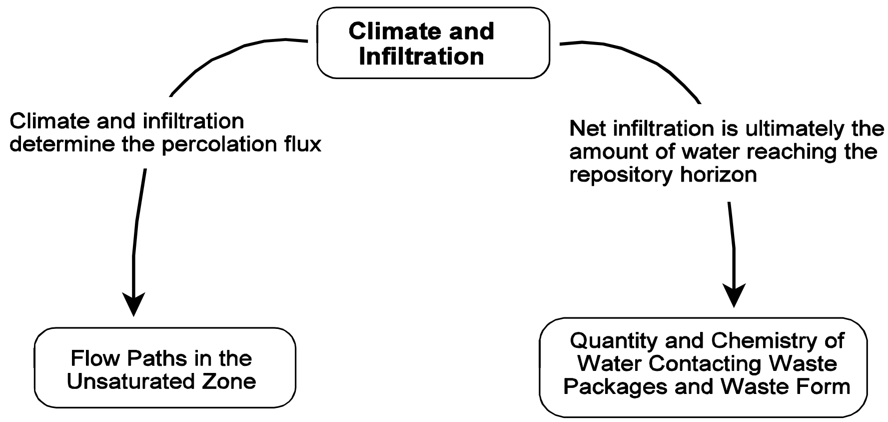

Figure 3.3.5-1. Diagram Illustrating the Relationship Between Climate and Infiltration and Other Integrated Subissues

The key technical issue subissues formed the bases for the previous versions of the issue resolution status reports and also were the bases for technical exchanges with DOE where agreements were reached for potential future resolution of subissues. The resolution status of this integrated subissue is based on the resolution status of each of the contributing key technical issue subissues. The subsequent sections incorporate applicable portions of these key technical issue subissues, however, no effort was made to explicitly identify each subissue.

3.3.5.3 Importance to Postclosure Performance

One aspect of risk-informing NRC review was to determine how this integrated subissue is related to the DOE repository safety strategy. The importance of shallow infiltration to repository performance at Yucca Mountain is recognized by DOE by identifying seepage into emplacement drifts as one of the eight principal factors in the repository safety strategy for the postclosure safety case (CRWMS M&O, 2000b) for the 10,000-year performance period. Because there is little evidence to support lateral movement of water between the ground surface and the repository, shallow infiltration determines the percolation flux at the repository horizon. The total system performance assessment abstraction for the seepage into emplacement drifts relies on seepage as a function of percolation rate. Because infiltration is the primary source of water in the unsaturated zone at Yucca Mountain, it is the first in a series of natural-system processes that must be considered to evaluate the quantity of water that could seep into emplacement drifts and to evaluate the flow paths and rates for transporting radionuclides below the repository to the water table.

Climate changes must also be considered in total system performance assessment because long-term changes in precipitation and temperature will significantly affect shallow infiltration rates (CRWMS M&O, 1999). Hence, during the 10,000-year compliance period, climate changes in the Yucca Mountain region are expected to produce (i) changes in precipitation and temperature that will affect the amount of deep percolation at the proposed repository horizon, (ii) increases in water table elevation that will reduce the distance from the repository horizon to the water table, and (iii) changes in saturated zone groundwater fluxes and flow paths from beneath the repository to the compliance boundary.

3.3.5.4 Technical Basis

The NRC has developed a Yucca Mountain Review Plan (NRC, 2002) that is consistent with the acceptance criteria and review methods found in previous issue resolution status reports. A review of DOE approaches for including climate and infiltration in total system performance assessment abstractions is provided in the following subsections. The review is organized according to the five acceptance criteria identified in Section 1.5 as follows: (i) System Description and Model Integration Are Adequate, (ii) Data Are Sufficient for Model Justification, (iii) Data Uncertainty Is Characterized and Propagated Through the Model Abstraction, (iv) Model Uncertainty Is Characterized and Propagated Through the Model Abstraction, and (v) Model Abstraction Output Is Supported by Objective Comparisons.

3.3.5.4.1 System Description and Model Integration Are Adequate

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.5.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess climate and infiltration with respect to system description and model integration.

The DOE technical bases for including or excluding the features, events, and processes related to climate and infiltration are provided primarily in CRWMS M&O (2000c). A list of features, events, and processes, for which screening arguments by DOE are not adequate or require verification, is provided in Section 3.2.1 of this report. The following paragraphs provide a brief description of the conceptual and modeling approach developed by DOE to integrate features, events, and processes that affect climate and infiltration into the total system performance assessment abstraction.

The approach and technical basis for the abstraction of climate change are documented by DOE in CRWMS M&O (2000d), herein referred to as the climate analysis and model report. Key assumptions are that (i) climate is cyclical, (ii) climate change cycles can be timed with an orbital clock (i.e., Milankovitch forcing) calibrated with the Devils Hole chronology, and (iii) past climate cycles repeat themselves in sequential order. The DOE features, events, and processes database[1] was reviewed in NRC (2000b). Based on these assumptions, a 10,000-year climate history, beginning from approximately 400,000 years before the present, was selected as the most probable analog for the next 10,000 years. During this period, three different climate states have been identified: (i) present-day climate for the first 600 years; (ii) a monsoon climate that is warmer and wetter than present day for the following 1,400 years; and (iii) a glacial transition climate that is cooler and wetter than present for the balance of the 10,000-year period (CRWMS M&O, 2000a,d).

Changes in precipitation rates and temperature from one climate state to the next are estimated and integrated as boundary conditions for the shallow infiltration process model. For each climate state, time-varying precipitation and temperature boundary conditions were derived from measurements at local and climate analog sites. The basis for the choices of analog sites in Washington, Utah, Nevada, Arizona, and New Mexico is the relationship between climate change and the movement of the jet stream across the western United States (CRWMS M&O, 2000d). For the shallow infiltration abstraction, DOE has also added consideration of climate-induced changes in vegetation during future climates (CRWMS M&O, 2000e). The DOE abstraction of climate in total system performance assessment also includes an assumed climate-induced water table rise of 120 m [394 ft], which reduces transport path lengths from the proposed repository level to the water table during the monsoon and glacial-transition climate states.

The scope of the DOE shallow-infiltration process model is limited to surficial hydrological processes, with estimates of net infiltration limited to depth of the root zone only. As described in CRWMS M&O (2000a), the infiltration model covers a domain of 123.7 km^2 [47.8 mi^2] with 30 × 30 m [98 × 98 ft] computational cells. The important portions of the infiltration model domain are the 4.7-km^2 [1.8-mi^2] area of the repository footprint, which is dominated by Tiva Canyon bedrock covered by a thin layer of soil or no soil and the 38.7-km^2 [14.9-mi^2] area of the three-dimensional unsaturated zone site-scale model domain that uses the shallow infiltration estimates as steady-state boundary conditions. The shallow infiltration model is documented in CRWMS M&O (2000e).

Processes considered in the shallow infiltration model are precipitation, infiltration, evapotranspiration, snow accumulation and snowmelt, and surface water run-on. These processes are incorporated into a watershed-scale, volume-balanced model using a one-dimensional (vertical), root-zone infiltration submodel; an evaporation and net radiation submodel; a snowpack submodel; and a two-dimensional (horizontal) surface-water flow-routing submodel. Depending on the climate state, synthetic or measured meteorological data from local or climate analog sites are used as the boundary conditions for the shallow infiltration model. Combinations of a 15-year precipitation and temperature record developed from multiple local meteorological stations and two 100-year stochastically generated records are used to simulate mean, lower-, and upper-bound modern climate net infiltration. Measured meteorological data from the future climate analog sites described in CRWMS M&O (2000d) are used for lower- and upper-bound monsoon and glacial transition climate net infiltration. The

[1]DOE. “Yucca Mountain Features, Events, and Processes Database.” Revision 00b. Washington, DC: DOE. Preliminary version. September 1999.

meteorological boundary conditions are spatially distributed based on empirical correlations to elevation. In the infiltration model, water that exceeds the infiltration capacity of a soil column is routed to lower elevation nodes for subsequent infiltration or further downgradient routing. Potential evapotranspiration is determined by an energy balance that depends on net radiation, air temperature, ground heat flux, a saturation-specific humidity curve, and wind.

Calibration of the shallow infiltration is accomplished on a subwatershed basis using two storm events with concurrent stream gage measurements. Important parameters, fixed before calibration, include the soil thickness and equivalent bedrock permeability. Where soil is thin {0.5 m [1.6 ft]}, which is particularly true for the repository footprint, bedrock permeability becomes a sensitive parameter. Important parameters adjusted during the calibration process are root zone depth and percent area contributing to runoff. DOE agreed[2] to demonstrate that effects of near surface lateral flow on the spatial variability of net infiltration are appropriately considered.

It is reasonable to assume that vegetation density will increase and vegetation types will change during wetter and colder future climates. However, the infiltration analysis and model report indicates that these changes in vegetation are only considered for the upper-bound climate scenarios (CRWMS M&O, 2000e, Section 6.9.4). For the upper-bound monsoon climate, the root-zone weighting parameters were adjusted to approximate a 40-percent vegetation cover (compared with 20 percent for modern climate) and the maximum thickness of the bedrock root zone layer was increased from 2 to 2.5 m [6.5 to 8.2 ft]. For the upper-bound glacial-transition climate, the root-zone weighting parameters were adjusted to approximate a 60-percent vegetation cover and the maximum thickness of the bedrock root-zone layer was increased to 3 m [9.84 ft]. These increases in vegetation cover and root-zone depth increase evapotranspiration and, hence, decrease net infiltration. Increases in root-zone depth also increase the water-holding capacity of the soil and bedrock, which decreases shallow infiltration. It is reasonable to assume that the large increases in precipitation assumed for the upper-bound future climate scenarios would support increased vegetation cover and vegetation types with greater root-zone depth. No basis nor sensitivity analysis, however, was presented for the magnitude of the changes that account for increased vegetation and root-zone depth, hence, it is difficult to access the reasonableness of the magnitude of these changes. DOE agreed[3] to provide justification for use of the evapotranspiration model, and justify the use of the analog site temperature data.

Output from the DOE infiltration model is used to define spatially distributed, time-averaged estimates of net infiltration, which provide the necessary steady-state flux boundary conditions for the site-scale unsaturated zone flow model. Nine boundary conditions for the unsaturated zone flow model are developed, including low-, medium-, and high-infiltration scenarios for each of the three climate states. This integration of the infiltration model with the site-scale unsaturated zone flow model requires spatial averaging because the unsaturated zone flow

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[2]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001).” Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[3]Ibid.

model grid is coarser than that of the infiltration model. Temporal averaging is also used to convert the time-varying infiltration model output into an equivalent steady-state flux. DOE justifies spatial averaging and use of a steady-state flux boundary because the sparsely fractured, highly sorptive Paintbrush nonwelded tuff layer beneath the surface at Yucca Mountain is postulated to attenuate episodic surface infiltration pulses and spatially smooth localized zones of high infiltration. As discussed in Section 3.3.6, the assumption of steady-state flow caused by the Paintbrush nonwelded tuff requires further basis, which DOE has agreed to provide.

In summary, the unsaturated zone process model report, supporting analysis and model reports, and DOE and NRC agreements generally provide sufficient descriptions of the conceptual models, model formulations, and methods of integrating the models into total system performance assessment analyses. The climate and infiltration abstractions are generally consistent with the available data, and important physical phenomena and couplings are adequately incorporated or bounded. Assumptions are clearly stated and used consistently. The unsaturated zone process model report and supporting analysis and model reports provide sufficient descriptions of (i) the technical basis for estimating climate conditions during the compliance period, (ii) integration of the future climate conditions with the shallow-infiltration process, (iii) the approach and technical basis for the shallow-infiltration model, and (iv) integration of the shallow-infiltration process model into total system performance assessment analyses.

3.3.5.4.2 Data Are Sufficient for Model Justification

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.5.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess climate and infiltration with respect to data being sufficient for model justification.

Detailed descriptions of the climate data sets and how they can be used to justify the abstraction approach are provided in CRWMS M&O (2000d). Three data sets are crucial to development of the DOE approach: (i) Devils Hole calcite deposits, (ii) Owens Lake microfossil records, and (iii) meteorologic records from climate analog sites.

Devils Hole is located approximately 90 km [56 mi] south of Yucca Mountain in the Paleozoic limestone that comprises the regional aquifer. Calcite has precipitated on the walls of Devils Hole during the last 500,000 or more years, leaving a record of $\delta^{18}O$ that provides insights about long-term changes in average annual groundwater temperatures (i.e., climate change) (CRWMS M&O, 2000d). Because the calcites in Devils Hole have been dated, they provide a chronology of climate that reflects a cyclic change from interglacial to glacial climates. A relation between Devils Hole data and orbital precession is evident where maximal values of precession mark the ends of the Devils Hole interglacials and other warm periods (CRWMS M&O, 2000d). This relation was developed to provide a rationale for timing future climate change in terms of the Devils Hole chronology of climate change in the Yucca Mountain region. Thus, the Devils Hole data set provides a reasonable basis for forecasting the cyclical timing of climate change.

To reconstruct the climatological conditions that existed in the Yucca Mountain region for each climate state, microfossil records of diatoms and ostracodes from cores drilled at Owens Lake were used (CRWMS M&O, 2000d). Owens Lake is located on the eastern side of the Sierra Nevada Mountains, east of Los Angeles. The known environmental tolerances of ostracode and diatom species provide a way to interpret the relative total dissolved solids of the Owens paleolake, and the relative temperature of its water. The total dissolved solids and water-temperature information are then used to qualitatively infer a range of likely climate conditions—namely precipitation and temperature—during the Owens Lake stage 11 (interglacial period about 400,000 years ago) to stage 10 (glacial period) transition. In this manner, monsoon and glacial-transition climate states were identified as the sequence of climate states most likely to follow present-day climate in the Yucca Mountain region during the 10,000-year compliance period.

Once qualitative descriptions of future climate states were obtained from the Owens Lake record, it was necessary to identify analog sites where present-day climate conditions were qualitatively consistent with those inferred for the monsoon and glacial-transition climates (CRWMS M&O, 2000d). Meteorological stations within these analog areas were then selected to obtain precipitation and temperature data to be used as analog input to the infiltration process model. For the monsoon climate, meteorological stations from two analog sites (Nogales, Arizona, and Hobbs, New Mexico) were chosen to represent an upper bound; the modern climate meteorological record was used as a lower bound. For the glacial transition climate, lower- and upper-bound analog sites (Beowawe, Nevada; Delta, Utah; Rosalia, Washington; Spokane, Washington; and St. John, Washington) were chosen. Shallow infiltration simulation results using lower- and upper-bound meteorological records as inputs were averaged to create a mean net infiltration estimate for the future climates. The meteorological inputs for estimating mean shallow infiltration for the modern climate, however, were a synthetic 15-year record developed from local Yucca Mountain stations and a stochastically developed 100-year precipitation and temperature record developed from Yucca Mountain and Nevada Test Site weather stations.

There are no direct measurements of shallow infiltration at Yucca Mountain. The infiltration model relies on matrix pore water geochemical data to support and constrain the long-term shallow infiltration results. The infiltration model uses a plug-flow, or bucket, approach to model one-dimensional movement of water vertically into the soil and bedrock (CRWMS M&O, 2000e). Two-dimensional runoff routing is incorporated by tracking the amount of water flux that cannot be stored or transmitted vertically downward by the top layer. The plug-flow approximation for vertical flow ignores the effect of capillarity in the unsaturated zone, though this may be offset by the coarse vertical grids of the one-dimensional infiltration model. DOE agreed[4] to provide a technical basis that the water-balance plug-flow model adequately represents the nonlinear flow processes represented by Richard's equation, particularly over the repository where there is thin soil.

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[4]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

Data collected at Yucca Mountain to support infiltration modeling include soil and bedrock hydrological properties, meteorological data, soil and bedrock water-content profiles, soil and bedrock water chemistry and temperature, and streamflow measurements. These data reveal the episodic nature of precipitation events at Yucca Mountain. Short periods of heavy precipitation (including an occasional snowmelt) may produce fleeting surface run-on and stream flow events. The data also indicate that areas with thin soils and highly fractured bedrock permit rapid infiltration of water below the root zone. Meteorological measurements indicate that the average annual potential evapotranspiration rate is approximately six times greater than the average annual precipitation rate for the current climate, resulting in the arid condition of Yucca Mountain between episodic precipitation events (CRWMS M&O, 2000e). These data and observations are generally consistent with the conceptual model for infiltration at Yucca Mountain on which the process model is based and show the importance of considering processes such as surface runoff and evapotranspiration.

Estimation of equivalent bedrock permeability for a fractured tuff for the one-dimensional bucket model is highly uncertain. Bedrock permeability is a sensitive parameter for net infiltration estimates where soils are thin or nonexistent (NRC, 1999). The approach described in CRWMS M&O (2000e) to estimate equivalent bedrock properties for the one-dimensional infiltration model did not change from that used for the viability assessment. Rough estimates of properties were developed from laboratory measurements of cores and assumed fracture properties. A range of estimates for each lithologic unit was developed because matrix scaling and unknown fracture properties led to a large uncertainty in hydraulic property estimates. Six different assumptions about fracture characteristics were used to estimate six different columns of equivalent permeability. Geometric averaging of core permeability values was used to upscale the matrix permeability to the infiltration grid scale. The column of values chosen to be used in the infiltration model was determined by modeling changes in water content profiles over time as measured by neutron probes installed in shallow boreholes. Because neutron probes measure water content in the rock matrix, equilibration of bedrock matrix and fractures must be assumed. Near the ground surface, however, this equilibration is unlikely for moderate to densely welded tuffs because of preferential or focused flow in fracture networks. Alcove 1 is the only area where large-scale infiltration measurements into the soil and bedrock have been made at Yucca Mountain. Steady-state influx rates at the ground surface in the Alcove 1 tests can be used to approximate the equivalent bedrock permeability. The influx rates are an aspect of the test that have not been formally documented. However, informal communication of the rates indicate that the equivalent permeability of the fractured bedrock is 35 times greater than the bedrock permeability value used in the model.

Though fracture properties were not directly used in the final choice of equivalent hydraulic properties, the available fracture data from surface exposures could be used to support the assumed fracture characteristics used to develop the columns of potential values. Analysis in the Tiva Canyon upper lithophysal unit indicates that normalized fracture area is 30 to 50 times greater, and fracture porosity is 2 to 10 times greater[5] than assumed for the

[5]Fedors, R.W., D.A. Ferrill, and A.P. Morris. "Integration of Fracture Data into Shallow Infiltration Models." *Presentation to Geological Society of America, November 13–17, 2000*. Reno, Nevada. 2000.

infiltration model. This fracture porosity is relevant to surface infiltration at the ground surface (i.e., two-dimensional porosity along the ground surface plane).

To calibrate the model, streamflow measurements have been collected for selected subwatersheds as calibration targets; data from two storms were used (CRWMS M&O, 2000e). As part of this calibration, geochemical data were used to constrain estimates of net infiltration. Although this approach could lead to a well-calibrated model, it may lack the ability to accurately estimate net infiltration because the data are not sufficient to derive a unique best set of model parameters. For example, important calibrated parameters such as root zone depth, porosity, and area of watershed contributing to runoff may simply compensate for errors in fixed parameters such as bedrock permeability and soil depth.

The DOE infiltration model does not consider variations in bedrock saturation. Bedrock dryout zones beneath areas of thin or no soil cover, however, would tend to lessen rates of shallow infiltration. Thus, the predicted high net infiltration rates in areas of thin soil cover may be partly the result of neglecting variability in bedrock saturation. Another factor to consider is that water potential, saturation, and chloride content data from the Exploratory Studies Facility and East-West Cross Drift suggest that the runoff/run-on component of shallow infiltration is underpredicted beneath stream channels over the repository footprint.[6] So, there are indications that the DOE infiltration model may tend to overpredict net infiltration on ridges with thin soils and underpredict it in stream channels. The overall effect may be that net infiltration is more variable spatially than is predicted by the model.

For the concerns discussed previously, DOE agreed[7] to provide justification and documentation of Monte Carlo analyses. This would include the uncertain input parameters from the Analysis of Infiltration Uncertainty analysis and model report (e.g., reconciling the Alcove 1 test results with the bedrock permeability in the infiltration model).

Net infiltration is highly sensitive to soil depth, particularly when the soil layer covering the bedrock is thin. The repository footprint is dominated by thin soils. Measurements of soil thickness for a 30-m [98-ft] pixel—the grid size for the net infiltration model—are extremely difficult on the highly irregular bedrock surface. On steep slopes, point measurement of soil thickness can vary from 0 to 1 m [3.3 ft] in a 1-m^2 [11-ft^2] area. In small wash channels alone, the soil thickness can vary from 0 to 2 m [6.5 ft] for a 30-m [98-ft] distance. The approach described in CRWMS M&O (2000e) for estimating soil thickness values for the net infiltration grid is based on empirical equations for different geomorphic categories and different depth classes. Each equation assumes a slope angle-soil depth correlation. Although equations for thicker soils are constrained by information from borehole logs, thin soil thicknesses can only be constrained by qualitative visual observations in the field because of the highly irregular bedrock surface. Although the DOE approach leads to qualitatively reasonable results,

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[6]Flint, L. “Measuring Flow and Transport in Unsaturated Fractured Rocks: A Large-Scale Unsaturated Flow Experiment.” *Presentation to Geological Society of America, November 13–17, 2000*. Reno, Nevada. 2000.

[7]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Unsaturated and Saturated Flow Under Isothermal Conditions (October 31–November 2, 2000).” Letter (November 17) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

uncertainty in soil thickness estimates should not be ignored over the repository footprint where the soils are dominantly thin. This uncertainty, combined with the uncertainty in the constraints on the model results described in Section 3.3.5.4.5, leads to uncertain model results, particularly for future climate conditions. Instead of choosing to establish a better basis for the parameter values and constraints, DOE agreed[8] to propagate uncertainty through the abstraction in the total system performance assessment as described in Sections 3.3.5.4.3 and 3.3.5.4.4.

In summary, much of the available data at Yucca Mountain has been collected using acceptable techniques, and the conceptual models for climate and infiltration are generally consistent with the available site-specific data. The review of the paleoclimate data for the Yucca Mountain region and meteorological data from climate analog sites indicate that they have been collected using acceptable techniques. Although the DOE shallow-infiltration model adequately includes important features and processes, direct measurements of shallow infiltration are lacking, and a basis for the parameter values lacks supporting data. The missing data needed to fully support the shallow infiltration estimates, however, can be compensated for by propagating data uncertainty through the model, which is discussed in the following section. Thus, with the caveat that data uncertainty must be propagated through the shallow infiltration abstraction (see Section 3.3.5.4.3), adequate DOE and NRC agreements and sufficient data exist to support development of the shallow-infiltration process model for Yucca Mountain.

3.3.5.4.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.5.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess climate and infiltration with respect to data uncertainty being characterized and propagated through the model abstraction.

CRWMS M&O (2000a) identifies several sources of data uncertainty. First, there is uncertainty in knowing whether changes in $\delta^{18}O$ values are directly correlated with changes in mean annual precipitation and mean annual temperature or if there is a lead or a lag time between changes in regional climate. Second, each Devils Hole sample integrates a particular thickness of carbonate in a continuous sample series and represents about 1,000 years. Consequently, the data would not reveal changes in regional climate with durations much less than 1,000 years. Third, there is uncertainty in the sediment accumulation rate that was used to infer relative ages of the microfossils obtained from cores in Owens Lake. There is no simple nor objective way of assessing the nature of any of these three sources of uncertainty. A fourth source of uncertainty is the standard deviation associated with age estimates of Devils Hole calcite samples. Although the standard deviation of Devils Hole ages is itself an estimate of uncertainty, that estimate was not incorporated into the abstraction because the other sources of uncertainty cannot be estimated, and hence, their relation to standard deviation is unknown. A final source of uncertainty is the choice of a starting point, at 400,000 years before the

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[8]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001).” Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

present, assumed equivalent to modern climate for purposes of projecting forward. Though possible, the choice is somewhat arbitrary, considering the lack of data from Devils Hole over the last 8,000 years.

To address data uncertainty in the shallow infiltration model, DOE developed distributions for values of 12 input parameters to the infiltration process model (CRWMS M&O, 2000f, Table 4-1). These input parameters were stochastically sampled using a Latin hypercube sampling algorithm in a 100-realization Monte Carlo analysis of infiltration for a glacial-transition climate state. CRWMS M&O (2000f) did not, however, provide tangible evidence that 100 realizations would adequately represent the uncertainty distribution. The parameters chosen for development of uncertainty distributions were effective bedrock porosity, bedrock root zone thickness, soil depth, precipitation, potential evapotranspiration, bulk bedrock saturated hydraulic conductivity, soil saturated hydraulic conductivity, two parameters associated with bare soil evaporation, and effective surface-water flow area. Two additional parameters are related to sublimation and melting of snow cover.

Upper and lower bounds for the 12 infiltration model parameters were estimated partly by using physical limits and partly by judgment based on existing bounds within the available data. The logic and the data used to deduce reasonable limits, however, are not clearly described in CRWMS M&O (2000f), and the methods used to deduce these parameter distributions are not transparent to NRC staff. In fact, some of the parameter ranges listed in the infiltration uncertainty analysis and model report are physically impossible (e.g., a value of −10 for the lower bound of the precipitation multiplier). DOE indicated there are typographic errors that will be corrected in a future revision of the infiltration analysis and model report. DOE also agreed[9] to provide additional justification for the 12 stochastic parameters identified in CRWMS M&O (2000f, Table 4-1).

The range and distribution of net infiltration rates obtained from these Monte Carlo analyses of parameter uncertainty were used as the basis for estimating probability weighting factors of 0.17, 0.48, and 0.35 for low-, medium-, and high-infiltration scenarios, respectively (CRWMS M&O, 2000f, Table 6-2). For example, for a total system performance assessment realization with stochastically sampled inputs, there is a 48-percent chance that the unsaturated zone flow fields obtained from the medium-infiltration case will be selected. In this manner, data uncertainty is propagated through the total system performance assessment abstraction. It should be noted that values of the probability weighting factors are expected to change as a result of an NRC concern that the DOE upper-bound net infiltration estimates for the three climate states do not incorporate parameter uncertainty. DOE agreed[10] to provide the documentation sources and schedule for the Monte Carlo method for analyzing infiltration.

In summary, there are several concerns related to the propagation of data uncertainties in the abstraction of climate and infiltration. In each case, however, either the current DOE approach

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[9]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Unsaturated and Saturated Flow Under Isothermal Conditions (October 31–November 2, 2000)." Letter (November 17) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[10]Ibid.

is reasonably bounding, the uncertainty is not expected to be of significant importance to performance predictions, or DOE agreed to provide additional information or analyses to support the abstraction approach. Uncertainty is not incorporated into the deterministic approach used to estimate magnitude, type, and duration of climate change. Although CRWMS M&O (2000g) relies on robust canisters (no failures over 10,000 years) to justify that climate uncertainty is not important in total system performance assessment analyses, the duration of the glacial transition climate (i.e., covering 80 percent of the 10,000-year performance period) is, nonetheless, a reasonable conservative bound. DOE agreed that parameter uncertainty should be reflected in the lower- and upper-bound infiltration scenarios. The DOE approach to incorporating data uncertainty into the infiltration process model and total system performance assessment abstraction through Monte Carlo analysis will provide sufficient information for review.

3.3.5.4.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.5.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess climate and infiltration with respect to model uncertainty being characterized and propagated through the model abstraction.

Perhaps the most significant model uncertainty lies in not knowing what the magnitude will be of changes in precipitation and temperature for each climate state. This uncertainty is addressed in the climate model abstraction by using several analog sites for each climate state. The locations of these analog sites are described in CRWMS M&O (2000d, Table 2). Upper- and lower-bound values for precipitation and temperature are quantified by selecting meteorological stations at locations in areas with some or all of the common ostracodes and diatoms found in Owens Lake, thus integrating the biology, hydrology, and climate linkages that were expressed in the past at Owens Lake. Mean (expected) values of precipitation and temperature are determined by averaging the upper- and lower-bounding values obtained from the analog sites. DOE estimates of annualized mean, lower-, and upper-bound values of precipitation and temperature for the three climate states are listed in Table 3.3.5-1. These annualized values are for comparison only; actual inputs to the infiltration process model are time varying on a daily basis (CRWMS M&O 2000e). Model uncertainty that DOE did not directly consider is the variation of climate, on the scale of decades to centuries, that could lead to greater estimates of net infiltration.

It can be seen in Table 3.3.5-1 that the ranges of precipitation between lower- and upper-bounds for all climate states is quite large; hence, a large range of model uncertainty is incorporated into the abstraction. Note also that the increase in precipitation from modern to the monsoon and glacial transition climates is also quite large. These precipitation estimates for future climates are consistent with those previously estimated by DOE for the viability assessment (DOE, 1998) and found to be acceptable by NRC (1999) but have a more rigorous technical basis linking the approach to Devils Hole calcite and Owens Lake microfossil data.

Infiltration process model uncertainty results from the combined model parameter uncertainty, uncertainty in boundary conditions defined by the climate abstraction, and general uncertainty in the validity of various conceptual model assumptions. It is thus important that the ranges of infiltration estimates—the low, medium, and high cases—for each postulated climate state are sufficient to reasonably bound this combined uncertainty. The approach described in CRWMS M&O (2000a), however, falls short of this goal because the estimated low-, medium-, and high-infiltration scenarios are based only on consideration of climate uncertainty. That is, the low-, medium-, and high-infiltration estimates for each climate scenario are determined by setting model parameters to their expected values and simply running the model with the mean, lower-bound, and upper-bound climate boundary conditions (see Table 3.3.5-1). The DOE approach yields a set of nine infiltration scenarios used as constant-flux boundary inputs to the site-scale unsaturated zone flow model (CRWMS M&O, 2000a). The nine unsaturated zone flow model net infiltration scenarios are summarized in Table 3.3.5-2. Note that net infiltration flux to the unsaturated zone flow model is spatially variable; the values in Table 3.3.5-2 are averaged over the unsaturated zone flow model domain and are used for comparison only. A specific concern with the DOE approach is that model parameter uncertainty is not propagated into the range of net infiltration estimates, which should reflect both model and data uncertainties. Additionally, the current estimates for the upper-bound net infiltration scenarios are significantly lower than those the NRC staff considers acceptable for the viability

Table 3.3.5-1. Annualized Precipitation and Temperature Estimates Used in the Climate Abstraction for the Three Climate States*

Climate	Mean Annual Precipitation and Temperature		
	Lower Bound	**Mean**	**Upper Bound**
Modern (Note: temperature not provided for modern)	186.8 mm/yr [7.35 in/yr]	190.6 mm/yr [7.50 in/yr]	268.4 mm/yr [10.57 in/yr]
Monsoon	190.6 mm/yr [7.50 in/yr] 17.3 °C [63.1 °F]	302.7 mm/yr [11.92 in/yr] 17.2 °C [63.0 °F]	414.8 mm/yr [16.33 in/yr] 17.0 °C [62.6 °F]
Glacial Transition	202.2 mm/yr [7.96 in/yr] 10.2 °C [50.4 °F]	317.8 mm/yr [12.51 in/yr] 9.8 °C [49.6 °F]	433.5 mm/yr [17.07 in/yr] 9.4 °C [48.9 °F]

*CRWMS M&O. "Unsaturated Zone Flow and Transport Model PMR." Section 3.5.1.8. TDP–NBS–HS–000002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000.

Table 3.3.5-2. Area-Averaged Mean Annual Infiltration Estimates for the Unsaturated Zone Site-Scale Flow Model Area*

Climate	Low-Infiltration Case (mm/yr)	Medium-Infiltration Case (mm/yr)	High-Infiltration Case (mm/yr)
Modern Climate	1.3 [0.051 in/yr]	4.6 [0.18 in/yr]	11.1 [0.44 in/yr]
Monsoon Climate	4.6 [0.18 in/yr]	12.2 [0.48 in/yr]	19.8 [0.78 in/yr]
Glacial-Transition Climate	2.5 [0.10 in/yr]	17.8 [0.70 in/yr]	33.0 [1.30 in/yr]

*CRWMS M&O. "Unsaturated Zone Flow and Transport Model PMR." Table 3.5-4. TDP–NBS–HS–000002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000.

assessment (DOE, 1998). The DOE plan to address this NRC concern includes three elements: (i) develop an upper-bound infiltration case based on the 90th percentile from the Monte Carlo analysis of the glacial-transition climate documented in CRWMS M&O (2000f), (ii) develop upper-bound infiltration cases for the monsoon and modern climates by proportional scaling based on the ratio between upper-bound and mean cases for the glacial-transition climate, and (iii) calculate new probability weighting factors into the total system performance assessment analyses using the same methodology developed in CRWMS M&O (2000f).

At a technical exchange,[11] DOE staff conveyed preliminary estimates for the revised high-infiltration scenarios for the glacial-transition and monsoon climates as being 53 and 30 mm/yr [2.1 and 1.2 in/yr]; the estimate for modern climate is not expected to change. Probability weighting factors also need to be recalculated, DOE staff explained, because selecting the high-infiltration scenario from the end of the Monte Carlo distribution translates to a decreased probability that this scenario would occur. It was stated that the revised probability weighting factor for the high-infiltration scenario will be about 20 percent. Although the weighting factor is lower, total system performance assessment simulations would still sample a reasonably large proportion of high-infiltration scenarios. NRC staff agreed that this concern regarding infiltration model uncertainty is resolved, pending incorporation of these proposed changes into total system performance assessment calculations used to support the license application.

In summary, the use of multiple analog sites results in a wide range of mean annual precipitation estimates for the monsoon and glacial-transition climate states. The estimated climate conditions are consistent with those previously found acceptable by NRC staff (NRC, 1999) and are considered acceptable for the current abstraction. Staff are concerned that the range of net infiltration estimates used for the abstraction does not adequately bound the model and parameter uncertainty in the shallow infiltration process model. In response, DOE agreed to use Monte Carlo analyses of model parameters to revise the upper-bound infiltration scenario for the total system performance assessment abstraction.

[11]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Unsaturated and Saturated Flow Under Isothermal Conditions (October 31–November 2, 2000)." Letter (November 17) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

3.3.5.4.5 Model Abstraction Output Is Supported by Objective Comparisons

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.5.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess climate and infiltration with respect to model abstraction being supported by objective comparisons.

Predictions of future climate are derived from meteorological conditions recorded at analog sites across the western United States. The sites were chosen based on their consistency with the Owens Lake record. In the climate analysis and model report, it is reasoned that climate conditions at Owens Lake are similar to those at the top of Yucca Mountain and subject to the same climate cycles. Regional changes to climate are driven by shifts in the jet stream pattern. Thus, an objective comparison exists between modern climate conditions at Yucca Mountain and Owens Lake. Although the comparisons are subjective between future climate conditions (based on the Owens Lake record) and those climate conditions that may occur at Yucca Mountain, confidence is gained because uncertainty is incorporated through the use of upper-bound precipitation and temperature estimates for the climate abstraction.

Estimates of precipitation and temperature during past glacial climates in the Yucca Mountain region have been derived from a study of the plant macrofossils found in packrat middens (Thompson, et al., 1999). These observations were interpreted to show that, during the last full-glacial climate at Yucca Mountain, mean annual precipitation was approximately 266–321 mm [10.5–12.6 in.], and mean annual temperature was about 7.9–8.5 °C [46.2–47.3 °F]. Although these estimates are uncertain, they provide an independent and objective basis for comparison showing that a precipitation estimate for the last full glacial climate at Yucca Mountain is consistent with the mean estimated for the glacial-transition climate (Table 3.3.5-1). In addition, the uncertainty in the estimates from packrat middens is conservatively bounded by upper-bound glacial-transition estimates (Table 3.3.5-1).

For validation of the shallow-infiltration abstraction, CRWMS M&O (2000e) cites a 7–14-mm/yr [0.28–0.55-in/yr] estimate of recharge to the saturated zone beneath Yucca Mountain, based on measurements of chloride from saturated zone boreholes (CRWMS M&O, 2000h) and an assumed long-term average annual precipitation rate of 170 mm/yr [6.7 in/yr]. Using a chloride mass balance approach, net infiltration has also been estimated from matrix pore water samples in the Exploratory Studies Facility; samples obtained from the North Ramp, Main Drift, and Cross Drift correspond to infiltration rates of 5–14 mm/yr [0.20–0.55 in/yr]; samples from the South Ramp yielded estimates of 1–2 mm/yr [0.04–0.08 in/yr] (CRWMS M&O, 2000h). These estimates are broadly consistent with the DOE estimates for spatial distributions of infiltration for the modern climate (CRWMS M&O, 2000e). It should be noted, however, these values were revised downward by approximately 50 percent from previously reported values (CRWMS M&O, 1998) because of a reinterpretation of the chloride input from precipitation and wind-blown processes. The reduction was accomplished by a reinterpretation of the chloride input from precipitation and wind-blown processes. The previously assumed chloride concentration of precipitation and wind-blown soil particles (0.62 mg/L) [3.58×10^{-7} 07/in^3] was revised downward (0.30 mg/L) [1.73×10^{-7} 07/in^3] based on historical interpretation of Cl-36 data. Temporal aspects, both in the precipitation and in the dating of bedrock matrix water and its geochemical composition, clearly are important.

There are uncertainties and potential biases associated with recharge estimates obtained from the chloride mass balance method. For example, the chloride mass balance applies to one-dimensional plug flow in a homogeneous porous medium. Chloride measurements are obtained from matrix pore water, yet the conceptual model for flow in the unsaturated zone at Yucca Mountain is that flow occurs predominantly in fractures; fracture-matrix interactions are not taken into account in the chloride mass balance method. Based on the assumptions for the method, chloride mass balance should lead to an estimate of the lower bound on percolation, not the mean value. Thus, to gain additional confidence in chloride-based infiltration estimates, the site-scale unsaturated zone flow and transport model, which includes fracture-matrix interactions, used pore water chloride concentrations in the Exploratory Studies Facility and East-West Cross Drift as calibration targets. Model results indicate a range of net infiltration rates from 3–10 mm/yr [0.12–0.39 in/yr] (CRWMS M&O, 2000a, Figure 3.8-4). Though this range of infiltration estimates is generally consistent with infiltration model calculations, the meaning of the results are not clear. The results may demonstrate (i) that the model is self-consistent with its calibration to those same infiltration rates, (ii) that the assumed chloride fluxes at the ground surface can be matched with the matrix chloride concentrations, and (iii) that a deficiency in using a simple mixing model approach exists. Chloride content in the subsurface depends on the flux at the ground surface and also on the spatially variant evaporation history in the subsurface, particularly in the Tiva Canyon where barometric pumping is likely prominent.

Neutron probe profiles collected during a 4-year period were used to estimate shallow infiltration at approximately 98 locations covering a range of geomorphic sites. The range of shallow infiltration estimates is 0–80 mm/yr [0–3.1 in/yr] for all geomorphic areas (CRWMS M&O, 2000e); an approximate average of 33 mm/yr [1.3 in/yr] is estimated for ridges and slideslopes only, which dominate the repository footprint (CRWMS M&O, 2000e, Figure 6-5). The high value of shallow infiltration may reflect the correspondence with wetter than average climatic conditions during the short period of measurements collected in the 1990s. Conversely, neutron probe data reflect minimum estimates because the probes estimate bedrock matrix water content; flow bypassing in fractures may be missed by the probe.

In an independent analysis, Winterle, et al. (1999) estimated an infiltration rate of 6.7 mm/yr [0.26 in/yr], for an area comparable to the unsaturated zone flow model area, based on a fit of infiltration estimates obtained from borehole temperature profiles to a lognormal statistical distribution. Uncertainty in shallow infiltration estimates based on temperature profiles is reflected in (i) the bias of geomorphic locations of boreholes, (ii) the bias created by elimination of boreholes with high values of percolation because they must be affected by a fault system, and (iii) the bias caused by the small number of point estimates.

The uncertainty in the parameter values and the uncertainty in the constraints on the model results described in this section, lead to uncertain model results, particularly for future climate conditions. DOE agreed[12] to propagate uncertainty through the abstraction in the total system

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[12]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

performance assessment. In addition, DOE agreed[13] to provide justification and documentation of Monte Carlo analyses. This would include the uncertain input parameters from the analysis of infiltration uncertainty (CRWMS M&O, 2000f).

In summary, the climate and infiltration abstractions of Yucca Mountain are generally consistent with the DOE interpretations of empirical observations. Interpretation of past climate conditions based on plant macrofossils in packrat middens is used to verify DOE climate forecasts for Yucca Mountain. For the shallow infiltration model, there is generally good agreement—well within one order of magnitude—between the infiltration model estimates and those obtained from geochemical data, flow and transport modeling, and borehole thermal profiles. Considering the manifold uncertainties in model boundary conditions, parameter values, and conceptual model assumptions, however, it is important for DOE to assess repository performance using ranges of future climate conditions and net infiltration estimates that reasonably bound those uncertainties. The agreements reached between DOE and NRC (discussed in the preceding section), when implemented, will ensure that the range of uncertainty in climate change and in the spatial and temporal distributions of infiltration at Yucca Mountain will be adequate for inclusion in a potential license application.

3.3.5.5 Status and Path Forward

Table 3.3.5-3 provides the status of all key technical issue subissues, referenced in Section 3.3.5.2, for the Climate and Infiltration Integrated Subissue. The table also provides the related DOE and NRC agreements pertaining to the Climate and Infiltration Integrated Subissue. The agreements listed in the table are associated with one or all five generic acceptance criteria discussed in Section 3.3.5.4. Note that the status and the detailed agreements (or path forward) pertaining to all the key technical issue subissues, are provided in Table 1.1-3 and Appendix A.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (through specified testing, analyses, and the like), acceptably addresses the NRC questions so that no information beyond that provided, or agreed to, will likely be required at the time of a potential license application.

Table 3.3.5-3. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreements*
Unsaturated and Saturated Flow Under Isothermal Conditions	Subissue 1—Climate Change	Closed	None
	Subissue 2—Hydrologic Effects of Climate Change	Closed	None
	Subissue 3—Present-Day Shallow Infiltration	Closed-Pending	USFIC.3.01 USFIC.3.02

[13]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Unsaturated and Saturated Flow Under Isothermal Conditions (October 31–November 2, 2000)." Letter (November 17) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

Table 3.3.5-3. Related Key Technical Issue Subissues and Agreements (continued)

Key Technical Issue	Subissue	Status	Related Agreements*
Structural Deformation and Seismicity	Subissue 3—Fracturing and Structural Framework of the Geologic Setting	Closed-Pending	None
Total System Performance Assessment and Integration	Subissue 1—System Description and Demonstration of Multiple Barriers	Closed-Pending	None
	Subissue 2—Scenario Analysis and Event Probability	Closed-Pending	TSPAI.2.01 TSPAI.2.02
	Subissue 3—Model Abstraction	Closed-Pending	TSPAI.3.18 TSPAI.3.19 TSPAI.3.20 TSPAI.3.21
	Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards	Closed-Pending	None

*Related DOE and NRC agreements are associated with one or all five generic acceptance criteria.

3.3.5.6 References

CRWMS M&O. "Chapter 2: Unsaturated Zone Hydrology Model. Total System Performance Assessment—Viability Assessment (TSPA-VA) Analyses Technical Basis Document." B00000000–01717–4301–0002. Revision 01. Las Vegas, Nevada: TRW Environmental Safety Systems, Inc. 1998.

———. "Total System Performance Assessment—Site Recommendation Methods and Assumptions." TDR–MGR–MD–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 1999.

———. "Unsaturated Zone Flow and Transport Model PMR." TDP–NBS–HS–000002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000a

———. "Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations." TDRL–WIS–RL–000001. Revision 04. Las Vegas, Nevada: CRWMS M&O. 2000b.

———. "Features, Events, and Processes in UZ Flow and Transport." ANL–NBS–MD–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000c.

———. "Future Climate Analysis." ANL–NBS–GS–000008. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000d.

———. "Simulation of Net Infiltration for Modern and Potential Future Climates." ANL–NBS–GS–000032. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000e.

———. "Analysis of Infiltration Uncertainty." ANL–NBS–HS–000027. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000f.

———. "Total System Performance Assessment for the Site Recommendation." TDR–WIS–PA–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000g.

———. "Analysis of Geochemical Data for the Unsaturated Zone." ANL–NBS–GS–000004. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000h.

DOE. "Viability Assessment of a Repository at Yucca Mountain." Overview and all five volumes. DOE/RW–0508. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 1998.

NRC. "Issue Resolution Status Report, Key Technical Issue: Unsaturated and Saturated Flow Under Isothermal Conditions." Revision 2. Washington, DC: NRC. 1999.

———. "Issue Resolution Status Report, Key Technical Issue: Structural Deformation and Seismicity." Revision 3. Washington, DC: NRC. 2000a.

———. "Issue Resolution Status Report, Key Technical Issue: Total-System Performance Assessment and Integration." Revision 3. Washington, DC: NRC. 2000b.

———. NUREG–1804, "Yucca Mountain Review Plan—Draft Report for Comment." Revision 2. Washington, DC: NRC. March 2002.

Thompson, R.S., K.H. Anderson, and P.J. Bartlein. "Quantitative Paleoclimatic Reconstructions from Late Pleistocene Plant Macrofossils of the Yucca Mountain Region." U.S. Geological Survey Open-File Report 99-338. 1999.

Winterle, J.R., R.W. Fedors, D.L. Hughson, and S. Stothoff. "Review of the Unsaturated Zone Models Used to Support the Viability Assessment of a Repository at Yucca Mountain." San Antonio, Texas: CNWRA. 1999.

3.3.6 Flow Paths in the Unsaturated Zone

3.3.6.1 Description of Issue

The Flow Paths in the Unsaturated Zone Integrated Subissue addresses effects of subsurface geology and hydrologic processes on the distribution and velocity of flow between the shallow subsurface and the water table at Yucca Mountain. Relationship of this integrated subissue to other integrated subissues is depicted in Figure 3.3.6-1. The overall organization and identification of all the integrated subissues are depicted in Figure 1.1-2. The DOE description and technical basis for abstractions of flow paths in the unsaturated zone are documented in CRWMS M&O (2000a) and numerous supporting analysis and model reports. This section reviews the abstractions of flow paths in the unsaturated zone incorporated by DOE in its total system performance assessment.

3.3.6.2 Relationship to Key Technical Issue Subissues

The Flow Paths in the Unsaturated Zone Integrated Subissue incorporates subject matter previously captured in the following key technical issue subissues:

- Unsaturated and Saturated Flow Under Isothermal Conditions: Subissue 4—Deep Percolation (NRC, 1999)
- Radionuclide Transport: Subissue 1—Radionuclide Transport Through Porous Rock (NRC, 2000a)
- Radionuclide Transport: Subissue 3—Radionuclide Transport Through Fractured Rock (NRC, 2000a)
- Structural Deformation and Seismicity: Subissue 3—Fracturing and Structural Framework of the Geologic Setting (NRC, 2000b)
- Thermal Effects on Flow: Subissue 1—Features, Events, and Processes Related to Thermal Effects on Flow (NRC, 2000c)
- Thermal Effects on Flow: Subissue 2—Thermal Effects on Temperature, Humidity, Saturation, and Flux (NRC, 2000c)
- Repository Design and Thermal-Mechanical Effects: Subissue 2—Design of the Geologic Repository Operations Area for the Effects of Seismic Events and Direct Fault Disruption (NRC, 2000d)
- Repository Design and Thermal-Mechanical Effects: Subissue 3—Thermal-Mechanical Effects on Underground Facility Design and Performance (NRC, 2000d)
- Evolution of the Near-Field Environment: Subissue 1—Importance to Performance of Coupled Thermal-Hydrological-Chemical Effects on Seepage and Flow (NRC, 2000e)

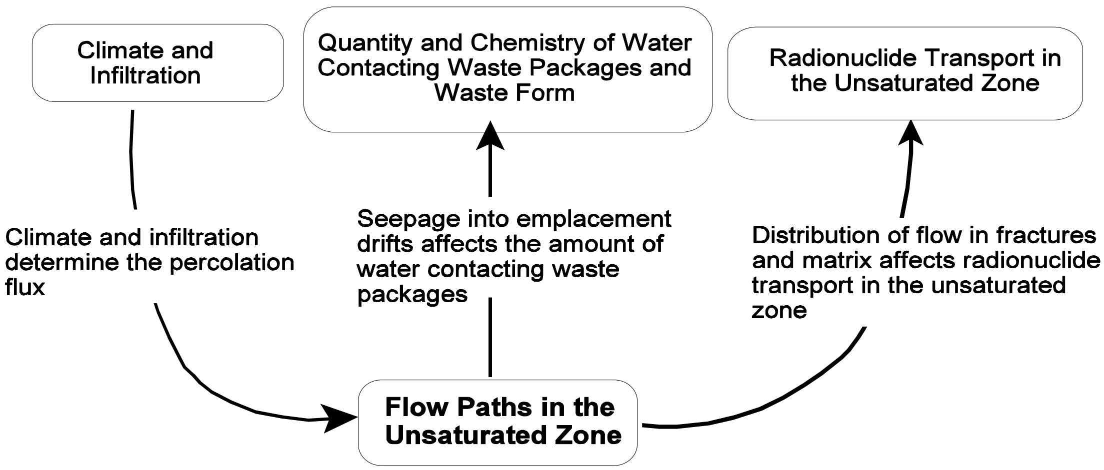

Figure 3.3.6-1. Diagram Illustrating the Relationship Between Flow Paths in the Unsaturated Zone and Other Integrated Subissues

- Total System Performance Assessment and Integration: Subissue 1—System Description and Demonstration of Multiple Barriers (NRC, 2000f)
- Total System Performance Assessment and Integration: Subissue 2—Scenario Analysis and Event Probability (NRC, 2000f)
- Total System Performance Assessment and Integration: Subissue 3—Model Abstraction (NRC, 2000f)
- Total System Performance Assessment and Integration: Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards (NRC, 2000f)

The key technical issue subissues formed the bases for the previous versions of the issue resolution status reports and also were the bases for technical exchanges with DOE where agreements were reached on what additional information DOE needed to provide to resolve the subissue. The resolution status of this integrated subissue is based on the resolution status of each of the contributing key technical issue subissues. The subsequent sections incorporate applicable portions of these key technical issue subissues, however, no effort was made to explicitly identify each subissue.

3.3.6.3 Importance to Postclosure Performance

One aspect regarding risk-informing the NRC review was to determine how this integrated subissue is related to the DOE repository safety strategy. The importance of considering flow paths in the unsaturated zone at Yucca Mountain is directly related to two of the principal factors in the current postclosure safety case identified by DOE in the repository safety strategy (CRWMS M&O, 2000b)—seepage into emplacement drifts and radionuclide delay through the unsaturated zone. Above the proposed repository horizon, the spatial distribution of hydrologic properties in the unsaturated zone can affect the spatial and temporal distribution of flow intersecting repository drifts. For example, flow of a given volume of water uniformly distributed in space and time is less likely to drip into an underground opening than if the same volume of water was channeled or focused into a small area above a drift or if the water was to arrive as a transient pulse. Within the proposed repository horizon, host-rock properties and engineering design features will affect the quantity of water that may contact drip shields or waste packages, which may affect waste package corrosion and mobilize radionuclides in the event of a waste package failure. Below the repository horizon, it is necessary to understand how the spatial distribution of hydrologic properties may affect the flow paths from the proposed repository horizon to the water table. For example, flow diverted into fast pathways along faults will have short travel times to the water table, and less mineral surface area will be available for sorption of radionuclides. Conversely, flow through sparsely fractured, vitric, nonwelded tuffs will occur mainly in rock matrix with much slower transport velocity and greater exposure of the surface area of mineral grains for radionuclide sorption.

Sensitivity analyses DOE conducted for the site recommendation (CRWMS M&O, 2000c) show that proposed repository performance at Yucca Mountain can be affected by flow focusing in fracture networks and seepage into drifts. Because of the assumed high diffusive releases from the waste packages, however, neither of these two processes had a significant effect on performance, particularly at simulation times prior to 40,000 years when the drip shield is mostly intact and Tc-99 dominates the dose estimate.

3.3.6.4 Technical Basis

NRC has developed a Yucca Mountain Review Plan (NRC, 2002) that is consistent with the acceptance criteria and review methods found in previous issue resolution status reports. A review of DOE approaches for including flow paths in the unsaturated zone in total system performance assessment abstractions is provided in the following subsections. The review is organized according to the five acceptance criteria identified in Section 1.5: (i) System Description and Model Integration Are Adequate, (ii) Data Are Sufficient for Model Justification, (iii) Data Uncertainty Is Characterized and Propagated Through the Model Abstraction, (iv) Model Uncertainty Is Characterized and Propagated Through the Model Abstraction, and (v) Model Abstraction Output Is Supported by Objective Comparisons.

3.3.6.4.1 System Description and Model Integration Are Adequate

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.6.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess flow paths in the unsaturated zone with respect to system description and model integration.

The site-scale unsaturated zone flow model is a three-dimensional, dual-continuum, unsaturated flow model used to estimate the flow rates and spatial distribution of flow reaching the proposed repository horizon and to evaluate potential contaminant transport pathways to the water table. For the mountain-scale unsaturated zone flow model, outputs from nine infiltration process model scenarios were used to develop an equal number of steady-state flux boundaries for discrete flow model realizations corresponding to the low, medium, and high net-infiltration scenarios for each of the three climate states. The numerical model grid represents the complex geology and stratigraphy using 32 layers with differing hydrologic properties. These layers dip to the east and are offset by numerous faults that are explicitly considered in the model. The area of the proposed repository transects three different model layers of the Topopah Spring welded tuff unit: about 10 percent is in the middle nonlithophysal layer, 78 percent in the lower lithophysal layer, and 12 percent in the lower nonlithophysal layer (CRWMS M&O, 2000d).

Each layer in the site-scale unsaturated zone flow model is assigned homogenous hydrologic properties, with the exception of the layers in the Calico Hills nonwelded unit, which are assigned hydrologic properties for either vitric or zeolitically altered rock types. The intralayer variability of hydrologic properties for the Calico Hills nonwelded unit is necessary to reproduce observations of perched water bodies found primarily in the northern part of the proposed repository area where lower-permeability, sparsely fractured zeolitic rock units predominate. The presence of the perched water bodies creates potential for the lateral flow of water to nearby high-permeability faults. Three-dimensional simulations of flow and radionuclide transport in the northern part indicate that flow in faults increases with depth below the repository horizon so that, over the unsaturated zone model domain, 35 percent of the deep percolation reaches the water table through faults (CRWMS M&O, 2000a). The percentage of flow from the repository horizon that reaches the water table through faults is not clear in the related process model report and analysis and model report. However, radionuclide transport studies using unsaturated zone flow fields from the mean modern infiltration scenario clearly show that rapid flow in fault zones contributes substantially to the calculated arrival of nonsorbing species at the water table (e.g., CRWMS M&O, 2000e, Section 6.12). DOE agreed[1] to provide the analysis of geochemical data used for support of the flow field below the repository.

Output from the site-scale unsaturated zone flow model is integrated into total system performance assessment analyses in two ways. First, estimates of flow reaching the proposed repository horizon in fractures are used to develop maps of percolation flux that are input to the drift seepage abstraction, which calculates the fraction of waste canisters that receive drips and the fraction of water that seeps into repository drifts. Second, calculated flow vectors in both fracture and matrix continua are used to delineate nine sets of unsaturated zone flow fields

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[1]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

used as input for the abstraction of radionuclide transport in the unsaturated zone. The drift seepage abstraction is discussed in the following paragraphs; the abstraction of radionuclide transport in the unsaturated zone is discussed in Section 3.3.7 of this report.

DOE acknowledges that accurate prediction of seepage from fractures into underground openings is an extremely difficult endeavor, and many of the physical processes that may affect seepage rates are poorly understood. Hence, DOE does not expect to accurately predict either individual seepage events or the precise spatial distribution along the emplacement-drift axis or the drift ceiling. Rather, the approach taken is aimed at yielding robust, conservative seepage estimates for a wide range of hydrologic conditions (CRWMS M&O, 2000a). The seepage abstraction begins with the Seepage Calibration Model (CRWMS M&O, 2000f), which incorporates results from air-permeability and liquid-release tests from Niche 3650 of the Exploratory Studies Facility to develop a methodology for the subsequent development of seepage process models. The Seepage Calibration Model is used to develop methodology and provide some confidence in the conceptual model for the performance assessment abstraction. The calibrated properties estimated from this model, however, are not used directly in the seepage abstraction. Rather, the seepage model for performance assessment (CRWMS M&O, 2000g) was developed as a stochastic approach to provide seepage estimates for a variety of hydrologic properties, percolation fluxes, and drift shapes. These stochastic results are then used in the seepage abstraction (CRWMS M&O, 2000h) to develop a simplified transfer function approach to include drift seepage in total system performance assessment simulations.

The seepage model for total system performance assessment is a three-dimensional, single-continuum, drift-scale unsaturated flow model used to develop transfer functions to estimate the fraction of wetted waste packages and the rate of seepage onto the wetted packages as functions of percolation flux at the repository horizon. This drift seepage process model represents a 5.23-m [17.2-ft] drift segment and is used to develop two transfer functions for use in the seepage abstraction. The first transfer function is a relationship between percolation flux and the fraction of waste package locations onto which seepage occurs (seepage fraction). The second transfer function describes a relationship between percolation flux and the seepage flux that enters those drift segments that receive seepage (seepage flux). An adjustment to the seepage flux transfer function was made to account for the effects of changes in the drift shape caused by rockfall. DOE simulations using the seepage model for total system performance assessment suggested a moderate increase of drift seepage as a result of partial drift degradation. Accordingly, seepage flow rates were increased by a factor of 1.55 to account for the effects from partial drift degradation and rock bolts. Seepage flow rates were further increased by 10 percent to account for potential correlation between fracture network permeability and the van Genuchten α parameter (related to capillary retention). These adjustment factors are based on results obtained from alternative scenario modeling (CRWMS M&O, 2000g). For example, seepage estimates from alternative models with correlated permeability and α parameters were 0–10 percent higher than the uncorrelated model; thus, rather than incorporate the correlated seepage model into the abstraction, DOE simply increased the current abstraction estimates by 10 percent, the upper end of this range.

The seepage abstraction for total system performance assessment makes use of the transfer functions for seepage fraction and seepage flux using maps of percolation flux estimates from the site-scale unsaturated zone flow model that are divided into six subregions. DOE recognizes that flow within the hundreds-of-meters scale of the six subregions may occur as localized weeps that focus flow from scales of several tens of meters into the scales smaller than the 15-m × 5.23-m [49-ft × 17.2-ft] scale of the drift seepage model. To account for this potential focusing, the area-averaged flux to each subregion is modified in the seepage abstraction using a flow-focusing factor (CRWMS M&O, 2000h, Section 6.3.3). The adjusted percolation fluxes are then used to obtain seepage fraction and seepage flux estimates for each subregion from the aforementioned transfer functions. Seepage fraction estimates are then reduced by dividing by the focusing factor to account for the fact that focusing of flow in one area needs to be balanced by a reduction of flow to other areas (CRWMS M&O, 2000h, Section 6.3.3).

Thermal-hydrological effects on seepage are accounted for by using the flux time histories from the thermal-hydrology abstraction (CRWMS M&O 2000i, Section 6.3) as input to the seepage abstraction. During the thermal pulse, increased percolation flux estimated from the drift-scale thermal-hydrological model is used as input to the seepage fraction and seepage flow rate transfer functions.

Depending on stress states and fracture orientations, various changes to fracture aperture could occur as a result of waste-generated thermal effects. DOE presently assumes that thermal-mechanical effects can be neglected in the drift seepage abstraction (CRWMS M&O, 2000j). To justify this assumption, DOE evaluated thermal-mechanical effects on hydrological properties through analyses of localized thermally induced rock response near a heated drift (CRWMS M&O, 2000k). However, an important case of a potential increase in the aperture of subhorizontal fractures in pillars between drifts was not considered. Such aperture increases may result from thermal-mechanical effects and could be important to cross-repository water flow because of the potential diversion of water flux from pillars to adjacent drifts, thereby focusing flux toward the drift (Ofoegbu, et al., 2001). To address this concern, DOE agreed[2] to provide (i) sensitivity analyses of thermal-mechanical effects on fracture permeability, including the effects of boundary conditions, coefficient of thermal expansion, fracture distributions, rock mass and fracture properties, and drift degradation, consistent with site-specific data and integrated with appropriate models; and (ii) the results of additional validation analysis of field tests related to the thermal-mechanical effects on fracture permeability.

DOE proposes to neglect thermal-hydrological-chemical-induced changes to hydrological properties based on numerical simulations of the Topopah Spring welded tuff that show that any such changes will have a negligible effect on seepage and flow paths (CRWMS M&O, 2000a,c). However, seepage and flow paths also can be affected by thermal-hydrological-chemical-induced changes to hydrological properties of the nonwelded Paintbrush Tuff and Calico Hills formations, for which no numerical simulations or analyses have been provided. The technical basis has not been provided for neglecting thermal alteration of the nonwelded

[2]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Thermal Effects on Flow (January 8–9, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

Paintbrush Tuff and Calico Hills hydrogeologic units. To address this concern, DOE agreed[3] to provide (i) additional documentation of results of thermal-hydrological-chemical simulations showing negligible porosity and permeability changes in the nonwelded Paintbrush Tuff and Calico Hills hydrogeological units; and (ii) additional technical bases for the treatment of the effects of cementitious materials on hydrologic properties, including an evaluation of the potential effects on hydrologic properties and radionuclide transport characteristics of the unsaturated zone.

The identification and screening of features, events, and processes are discussed in Section 3.2 of this report. Features, events, and processes for which DOE screening arguments were not adequate or required verification are discussed, as are their associated path forward. Several features, events, and processes are excluded from the Total System Performance Assessment–Site Recommendation abstraction of unsaturated zone flow based on screening arguments that the features, events, and processes are of low probability or low consequence to performance predictions. The screening arguments pertaining to the abstraction of flow paths in the unsaturated zone are outlined by CRWMS M&O (2000l). The adequacy of features, events, and processes integration into the total system performance assessment abstractions is discussed in Section 3.2.1.

In summary, the unsaturated zone process model report, supporting analysis and model reports, and DOE and NRC Agreements generally provide sufficient descriptions of the conceptual models, model formulations, and methods of integrating the unsaturated zone flow and drift seepage models into total system performance assessment analyses. Important design features, physical phenomena, and couplings are adequately incorporated or bounded for inclusion in a potential license application. Assumptions are clearly stated and used consistently throughout the abstraction of flow paths in the unsaturated zone.

3.3.6.4.2 Data Are Sufficient for Model Justification

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.6.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess flow paths in the unsaturated zone with respect to data being sufficient for model justification.

An extensive database is available for rock matrix properties at Yucca Mountain. These properties include moisture retention characteristics, permeability, porosity, and rock density, which are all measured in the laboratory on samples and cores collected from bedrock transects, surface-based boreholes, and alcove, drift, and niche boreholes in the Exploratory Studies Facility (e.g., Flint, 1998).

Pneumatic pressure signals between boreholes, core saturation data from laboratory measurements, and *in-situ* moisture potential profiles from boreholes were used to calibrate

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[3]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001).” Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

the unsaturated zone flow model (CRWMS M&O, 2000m). Observations of perched water also are used for unsaturated zone flow model calibration. Perched water bodies exist in the north below the potential repository horizon and in the south in the vicinity of Ghost Dance fault. Perched water bodies have been encountered in boreholes at both the vitrophyre between the Topopah Spring welded tuff and Calico Hills nonwelded units and at the vitric-zeolitic interface within the Calico Hills nonwelded unit. Data from pumping tests were collected to evaluate the spatial extent of the perched water bodies, and water samples were collected for age dating.

Subsurface studies in the underground Exploratory Studies Facility include data from four alcoves in the North Ramp: Alcove 1 provides access to the upper Tiva Canyon welded tuff unit, Alcove 2 to the Bow Ridge fault, Alcove 3 to the upper Paintbrush nonwelded tuff contact, and Alcove 4 to the lower Paintbrush nonwelded tuff contact. These alcoves were largely used to collect cores, measure air permeability, and sample gases. Alcoves 6 and 7, along the Main Drift, were designed to measure the properties of the Ghost Dance fault. Alcoves 4 and 6 were used to conduct fracture-matrix and fault-matrix interaction tests. Alcove 1 was instrumented with seepage collectors and wall sensors for a large-scale infiltration and seepage test. Bomb-pulse Cl-36 data have verified the existence of fast flow from the land surface to the potential repository horizon. A majority of the bomb-pulse signal locations in the Exploratory Studies Facility and East-West Cross Drift can be linked with locations where faults cross the Paintbrush nonwelded tuff, though several of these locations have no clear association with faults. It should be noted that investigators at Lawrence Livermore National Laboratory and Los Alamos National Laboratory appear to have collected conflicting data regarding the presence of bomb-pulse Cl-36 in the Exploratory Studies Facility. The U.S. Nuclear Waste Technical Review Board suggested that high priority be given to resolving this conflict.[4] DOE agreed[5] to reconcile the differences between the Cl-36 studies. Until the conflict is resolved, however, it is conservative to continue conceptual model development assuming the earlier findings that bomb-pulse Cl-36 has penetrated to repository depths.

Geochemical data such as total chloride, nonbomb-pulse Cl-36, and calcite fillings in fractures are used to build confidence in the conceptual and numerical models of flow and transport processes occurring in the mountain and to constrain the predictions of local and global percolation fluxes. This type of model validation is discussed further in Section 3.3.5.

Data from Niche 3650 seepage tests help to evaluate the capillary barrier and seepage threshold (zero seepage below a threshold percolation flux) conceptual models and provide estimates of fracture-network, moisture-retention properties. These data include air permeability and measurements of injected aqueous dye tracers released as pulses above the ceiling of Niche 3650 (CRWMS M&O, 2000n). The observed distribution of tracers arriving at the ceiling of the niche was sampled to evaluate spatial distributions of flow paths

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[4]Cohon, J.L. Letter (June 16) to Dr. Ivan Itkin, Director, Office of Civilian Radioactive Waste Management, DOE. Arlington, Virginia: U.S. Nuclear Waste Technical Review Board. <www.nwtrb.gov/corr/jlc076.pdf> 2000.

[5]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Unsaturated and Saturated Flow Under Isothermal Conditions (August 16–17, 2000)." Letter (September 8) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

associated with the wetting-front movement through the fractures. These data are used in inverse models to estimate hydrologic properties for fracture networks surrounding drifts.

DOE researchers interpret the seepage test data to indicate that seepage thresholds may be much larger than the percolation fluxes predicted by the unsaturated zone flow model (CRWMS M&O, 2000f). NRC previously commented, however, that conclusions drawn from Niche 3650 seepage tests potentially could be biased by ventilation dryout, the close proximity of the injection boreholes to the Niche ceiling, and by injection rates much greater than ambient percolation flux (e.g., NRC, 1999). Several ongoing tests at Yucca Mountain, if conducted carefully, may address these concerns. These tests include the Alcove 8–Niche 3 test and the East-West Cross Drift passive monitoring test. In the Alcove 8–Niche 3 test, tracer-bearing water is to be applied to areas in Alcove 8 of the East-West Cross Drift, about 10 m [33 ft] directly above Niche 3 of the Exploratory Studies Facility. This test encompasses a relatively large volume (compared to previous tests) and is sealed off from ventilation. Of perhaps greater interest are ongoing passive monitoring tests in an approximately 1-km [0.62-mi] section of the East-West Cross Drift and in Alcove 7, which have been sealed off from ventilation (except for periodic entry to maintain equipment) and are continuously monitored to evaluate when ambient conditions have returned. DOE agreed[6] to complete the planned and ongoing testing in the underground at Yucca Mountain to address this issue.

Ongoing seepage and transport tests in the drifts, niches, and alcoves at Yucca Mountain are being used to evaluate seepage and solute transport properties at Yucca Mountain. If and when repository construction occurs, it may not be feasible to conduct new seepage and transport studies for each repository drift. Rather, performance confirmation of seepage and transport properties may be based largely on examination of fracture patterns that intersect drift walls to evaluate whether they are consistent with fracture patterns in the drifts, niches, and alcoves used to develop and validate the total system performance assessment abstraction. Therefore, an approach needs to be in place to relate observed fracture patterns to possible drift seepage and transport properties. Although such an approach may be largely qualitative, it would nonetheless provide a useful basis for performance confirmation. DOE agreed[7] that observations of seepage need to be related to observed fracture patterns. Accordingly, observations of seepage in the passive test in the East-West Cross Drift will be related to full periphery fracture maps and other fracture data; fracture characterization data from the Alcove 8–Niche 3 test will also be provided.

Seepage into drifts also may be affected by thermally driven redistribution of water caused by waste-generated heat. An objective of the current design (Enhanced Design Alternative II) is to maintain temperatures below boiling in the pillars between drifts to allow condensate drainage between drifts. The ability to achieve this design objective depends, in part, on the efficacy of

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[6]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[7]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Structural Deformation and Seismicity (October 11–12, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

the ventilation system. The CRWMS M&O (2000o) ventilation model shows 70-percent heat removal by drift ventilation flow rates between 10 and 15 m^3/s [353 and 530 ft^3/s]. Several simplifying assumptions used in this model are not supported by experimental data, however. To address this concern, a quarter-scale ventilation test is being conducted at the Engineered Barrier Subsystem Test Facility in North Las Vegas, Nevada (CRWMS M&O, 2000p). DOE agreed[8] to provide results of the ventilation test in an update to the ventilation model.

Another concern related to thermal effects on flow is the lack of data to support modeling of fracture saturations, extent of dryout, formation of heat pipes, liquid fluxes in heat pipes, and, ultimately, the fate of thermally mobilized water in the drift-scale heater test. This concern is important because a key aspect of the proposed repository (Enhanced Design Alternative II) is the intention for thermally mobilized water to condense and drain through the pillars between drifts. Given uncertainties of the drift-scale heater test, such as in the losses of moisture through the bulkhead, and the lack of quantitative measurements of condensation and drainage in fractures, it is not clear whether the results of the drift-scale heater test can be used to determine the fate of thermally mobilized water. Measurements of mass losses through the drift-scale heater test bulkhead may help to reduce this uncertainty somewhat, but, if significant losses have occurred through the bulkhead during the past 3 years, it may be too late to assess those losses. To address this concern, DOE agreed[9] to provide a white paper on the technical basis for the current DOE understanding of heat and mass losses through the drift-scale heater test bulkhead and the effects of such losses on the test results. The white paper will include the technical basis for the decision to not monitor heat and mass losses through the drift-scale heater test bulkhead. The white paper will also address uncertainty in the fate of thermally mobilized water in the drift-scale heater test and the effect this uncertainty has on conclusions drawn from the drift-scale heater test results.

In summary, much of the available data on geology, hydrology, and geochemistry at Yucca Mountain have been collected using acceptable techniques, and the conceptual models for unsaturated zone flow and drift seepage are generally consistent with the available site-specific data. DOE has agreed to provide additional information and results from several ongoing and planned tests to validate conceptual models for relationships between seepage into drifts and fracture patterns, thermal and thermal-mechanical effects on flow and seepage, and effects of ventilation on the distribution of heat and water in pillars between drifts.

3.3.6.4.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.6.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess flow paths in the unsaturated zone with respect to data uncertainty being characterized and propagated through the model abstraction.

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[8]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Thermal Effects on Flow (January 8–9, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[9]Ibid.

Uncertainties generally exist in the estimated rock and fracture hydrologic properties because of sparse data and limitations of the estimation procedures used. This is particularly true for fracture and fault properties, such as moisture retention parameters and porosity. Because these properties cannot be readily measured, they were indirectly estimated from other measurements such as air permeability and fracture spacing. Site data are used for initial estimates of most matrix and fracture properties (CRWMS M&O, 2000q). Matrix porosity, fracture porosity, and residual saturation were fixed before calibration, whereas the remaining properties were further adjusted during the model calibration process. Thus, many of the parameter values used in the flow model are more a product of calibration than of site data analysis (CRWMS M&O, 2000m). DOE agreed[10] to use the field test data to provide additional confidence in the seepage abstraction and associated parameter values.

A concern with the treatment of data uncertainty in the abstraction of flow paths in the unsaturated zone is that measurement error, bias, and scale dependence in the saturation, water potential, and pneumatic pressure test data are not adequately accounted for in the process model used to predict flow paths in the unsaturated zone for total system performance assessment. For example, standard deviations of saturation data from cores were used to estimate weights for the weighted-least-squares inverse algorithm (CRWMS M&O, 2000m), but the effect of measurement errors on the resulting calibrated properties was not evaluated. Three types of data (matrix saturation from cores, water potential from boreholes, and pneumatic pressures) were obtained on different scales ranging from a few centimeters for cores to several tens of meters or more for pneumatic pressures. Matrix saturations from core data were upscaled by arithmetic averaging, a process that may tend to smooth out variability. It is not clear how the scale dependence of the water potentials and pneumatic pressure data were treated. Pneumatic pressure data are known to be scale-dependent because fracture permeabilities estimated from barometric pumping responses tend to be about two orders of magnitude greater than those determined from air-injection testing (CRWMS M&O, 2000b). The nonlinear least-squares maximum likelihood inverse method implemented in ITOUGH2 is essentially used only to obtain single parameter values and fails to properly account for all sources of variability and uncertainty and to propagate those sources through the calibrated model. Thus, the measurement error must be generalized to include such things as scale-dependence and modeling errors, because there is no other way to account for uncertainty in the least-squares inverse approach (e.g., McLaughlin and Townley, 1996). To address this concern, DOE agreed[11] to represent the full variability and uncertainty of data in the results of the thermal effects on flow simulations used for the abstraction of thermodynamic variables for other models or to provide technical bases that a reduced representation is appropriate, considering risk significance.

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[10]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[11]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Thermal Effects on Flow (January 8–9, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

The conceptual model used to develop the calibrated property sets for the site-scale unsaturated zone flow model is described in CRWMS M&O (2000m), which treats each geological layer in the model as homogeneous. The resulting average layer-calibrated drift-scale property sets for the basecase show fracture permeability in the Tsw34 unit to be 2.76×10^{-13} m^2 and in the Tsw35 unit to be 1.29×10^{-12} m^2. For the upper bound infiltration map these change to 4.63×10^{-13} m^2 and 5.09×10^{-12} m^2 and for the lower bound to 4.99×10^{-13} m^2 and 1.82×10^{-12} m^2 for the Tsw34 and Tsw35 units, respectively. Thus, variability and uncertainty in model layer fracture permeability for these two units range within approximately one order of magnitude. A statistical analysis of air-injection data collected from the niches in the Exploratory Studies Facility, however, found fracture permeabilities ranging from 1.53×10^{-15} m^2 to 7.15×10^{-10} m^2. These data, all collected in the Tsw34 unit, indicate that heterogeneity of fracture permeability can span at least four orders of magnitude within a single geological layer. It is not clear how using homogeneous layer properties in a model, with variability spanning only one order of magnitude, can adequately represent variability and uncertainty that may range several orders of magnitude within a single geological layer. CRWMS M&O (2000m) recommends that future studies consider the use of Monte Carlo simulations to evaluate the appropriateness of the prior information uncertainty for the calibrated properties. Such exercises would be useful for evaluating the propagation of uncertainty through the least-squares inverse approach as discussed previously. This would not, however, address the uncertainty inherent in spatial heterogeneity nor would it adequately address the uncertainty in the equally valid but significantly different models and property sets of the Thermal Tests Thermal-Hydrological Model (CRWMS M&O, 2000r). Additional studies applying generally accepted methods of stochastic subsurface hydrology, sensitivity, and bounding analyses would be required to address the data and model uncertainty. DOE agreed[12] to provide documentation of analyses of spatially heterogeneous fracture permeability using refinement of the grid for the heterogeneous fields in three dimensions and to evaluate the effect of high-permeability features (e.g., faults) crossing the drifts. DOE will also provide an update to CRWMS M&O (2000m) to incorporate uncertainties from all significant sources.

Data to support the values of assigned hydrologic properties of faults are also lacking. Because data from Borehole USW UZ–7a, used to characterize Ghost Dance fault, represent the most complete data set from within a fault zone at Yucca Mountain, these data are applied to all faults in the Unsaturated Zone Model (CRWMS M&O, 2000a). Additional data on the hydrologic and transport properties are presently being collected from the Alcove 8–Niche 3 test, which is intersected by a fault. One Alcove8–Niche 3 test objective is to characterize the fault and fractures across the lithophysal-nonlithophysal interface. DOE agreed[13] to provide the documentation for the Alcove 8–Niche 3 testing.

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[12]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Thermal Effects on Flow (January 8–9, 2001).” Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[13]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7,2000).” Letter (December 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

Another potentially important source of data uncertainty is the measurement of *in-situ* rock matrix saturations and water potentials used as calibration targets. Saturation data used in the calibration were obtained from rock cores collected *in situ* but analyzed *ex situ*. Corresponding field-based measurements of water content and water potential indicate that laboratory-derived estimates of the water retention relations underpredict saturations. Preliminary monitoring results from the East-West Cross Drift indicate the rock mass in the proposed repository horizon is wetter (i.e., water potentials are higher) and that moisture is more uniformly distributed than was expected based on earlier rock-core analyses.[14] Also, measurements of water potential taken in surface-based boreholes have gradually reequilibrated to ambient conditions that are much wetter than the data used to calibrate the three-dimensional unsaturated zone model. Of concern is that if the more recent measurements are validated, the calibrated unsaturated zone site-scale model should be consistent with these findings. Previous difficulties in matching saturations and water potentials may be alleviated by use of the ambient data in the calibration. Because of the complexity of the model and the large number of hydraulic parameters (matrix, fracture, or matrix/fracture parameter values) whose values could change during calibration to match the ambient, wetter conditions, it is not clear what the effect will be on the calibrated property data sets and predicted distributions of flow between fractures and matrix. DOE agreed[15] to use recent data on saturations and water potentials when calibrating the unsaturated zone flow model; thus, this uncertainty is expected to be reduced in future model iterations.

Input data from CRWMS M&O (2000s) are used to develop the unsaturated zone flow model grid. The unsaturated zone model numerical grids attempt to closely match the Geologic Framework Model 3.1 layers. However, because borehole data used to construct Geologic Framework Model 3.1 are limited, there is uncertainty in the assumptions regarding lateral continuity and thickness trends of layers at Yucca Mountain. Although layers in Geologic Framework Model 3.1 represent a valid interpretation, the effect of greater lateral discontinuity resulting from the inclusion of small faults on flow could be significant, especially in areas where little or no information has been collected. Areas of sparse data are generally outside the proposed repository area, however, so the effect of this data uncertainty is mitigated. Numerous fault zones and associated layer offsets within the proposed repository area are explicitly included in the unsaturated zone model grid. Hence, although considerable uncertainty exists in the accuracy of unsaturated zone model grids at any particular location, the model grid sufficiently allows for consideration of important effects on flow of faults and layer discontinuities at the scale and location of the proposed repository.

For the drift seepage model, spatial variability of air permeability data and the inability to directly measure moisture-retention properties of fracture networks produce uncertainty in the parameters k and α used in the seepage model for total system performance assessment, where k is fracture network permeability and α is a moisture-retention parameter (called

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[14]Craig, R. "Progress Report, March 1999." Letter (April 14, 1999) to W. Kozai, Yucca Mountain Site Characterization Office. Denver, CO: U.S. Geological Survey, Yucca Mountain Project Branch. 1999.

[15]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

van Genuchten's alpha) related inversely to air-entry pressure. In addition to the uncertainty in the appropriate range of values for these parameters is the uncertainty in their spatial distribution. Accordingly, uncertainty in two additional model parameters is considered: the standard deviation, σ, of the logarithm of fracture network permeability; and the spatial correlation length, λ, for fracture permeability. These two parameters are used to generate random spatial heterogeneity for permeabilities assigned to the seepage model grid cells.

The range of fracture permeability considered for *k* in the seepage model is from 0.9×10^{-14} to 0.9×10^{-11} m^2. This range is based on data from air permeability tests at Niche 3650, which indicate a mean permeability of 2.2×10^{-12} m^2. The low end of the range is consistent with host permeability measurements measured elsewhere in the Exploratory Studies Facility not affected by drift excavation; the high end of the range accounts for uncertainty in the degree of enhanced permeability from excavation effects (CRWMS M&O 2000g, Section 6.3.2). This range of *k* values is also consistent with the range of permeability measurements reported by LeCain (1997) for the Topopah Spring welded tuff middle nonlithophysal layer and, thus, seems reasonably to bound uncertainty in this parameter for the seepage process model. It is not yet established, however, if this range also includes or appropriately bounds variability in the lower lithophysal unit. DOE agreed[16] to use the field test data to provide additional confidence in, or a basis for, revising the total system performance assessment seepage abstraction and associated parameter values or provide a technical basis for not using it.

To incorporate uncertainty in the α parameter, four values were used: 1/α = 30, 100, 300, and 1,000 Pa. This range of values is somewhat arbitrary, but as discussed in the analysis and model report, it brackets values used in previous modeling studies (CRWMS M&O, 2000g, Section 6.3.4). Spatial variability of α is not considered for total system performance assessment abstraction. That is, for any particular process model realization used to develop the total system performance assessment abstraction, α was assumed constant throughout the entire model domain. DOE researchers did, however, investigate the sensitivity to spatial variability of α by evaluating a limited number of cases with α correlated to permeability. It is interesting to note that the correlated α condition yielded higher seepage by 0–10 percent (CRWMS M&O, 2000g). For this reason, seepage values used for the total system performance assessment abstraction are increased by 10 percent to allow for possible spatial correlation. One factor that should be considered is that the value of α at the drift-fracture interface is a function of fracture aperture and, hence, can vary considerably within scales of only a few centimeters. Because dripping is more likely to occur where water encounters an increased fracture aperture, DOE should demonstrate that the values of α used to develop the abstraction are consistent with the largest apertures typical for the grid-block scale. From the information presented by DOE thus far, it is not clear that the uncertainty in this important parameter has been incorporated adequately into the total system performance assessment abstraction. Test results from the Alcove 8–Niche 3 test and the East-West Cross Drift passive

[16]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

test should help resolve this concern. DOE agreed[17] to use the field test data to provide additional confidence in the seepage abstraction and associated parameter values.

Three alternatives, σ = 1.66, 1.93, and 2.5, were used to account for uncertainty in the standard deviation in fracture permeability used to incorporate random heterogeneity. The low value is based on data from Niche 3650 tests (CRWMS M&O, 2000f, Table 5); the two higher values span a value of 2.1 estimated in a modeling study by Birkholzer, et al. (1999). Note that higher values of σ represent stronger heterogeneity that would produce greater opportunity for local seepage. The values of σ seem reasonable to bound uncertainty. For example, the σ value of 2.5 would produce a distribution of permeability values that could vary spatially by 10 orders of magnitude (i.e., approximately 95 percent of assigned permeability values will be within a range of ± 2σ from the mean log-*k* value). Niche 3650 air permeabilities ranged from 1.53×10^{-15} m^2 to 1.27×10^{-10} m^2—about 5 orders of magnitude (CRWMS M&O, 2000f, Table 5).

Uncertainty in the correlation length scale, λ, for heterogeneity in fracture network permeability is not propagated through drift seepage model abstraction for total system performance assessment. In CRWMS M&O (2000f, Section 6.3.2), DOE investigators suggest that permeability is essentially random without a noticeable spatial correlation. Thus, to develop the total system performance assessment abstraction, heterogeneous fields for the seepage model were developed with λ equal to a grid size of 0.5 m [1.6 ft] (CRWMS M&O, 2000g). To further support this approach, process-level sensitivity studies were conducted with values of λ = 1 and 4 m [3.3 and 13 ft]. Results suggest that seepage increases with increased λ; hence, the DOE approach of neglecting spatial correlation of permeability may bias seepage predictions to be too low. Although DOE researchers cite data suggesting no spatial correlation beyond the grid-block scale, those data represent only one small niche and, owing to the data uncertainty, also have been interpreted to show a correlation scale of nearly 4 m [3.3 ft] (CRWMS M&O, 2000f). Another potentially important uncertainty is the presence of spatial correlation anisotropy caused by the presence of subvertical high-permeability fractures. The presence of subvertical high-permeability fractures could provide conduits for preferential flow toward drifts with a potentially reduced capacity for lateral capillary diversion—not considered in the DOE abstraction. Here also, test results from the Alcove 8–Niche 3 test and the East-West Cross Drift passive test should help resolve this concern. DOE agreed[18] to use the field test data to provide additional confidence in the seepage abstraction and associated parameter values.

A total of 576 seepage model scenarios was developed to represent the range of parameter uncertainty in the drift seepage model. These scenarios correspond to four values of α, four average-*k* values, three σ values, three realizations of random heterogeneity, and four percolation fluxes. The results of these numerous model scenarios were used to define transfer functions for seepage fraction and seepage flux as functions of percolation flux (CRWMS M&O, 2000h). It should be noted that only three realizations of random heterogeneity

[17]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[18]Ibid.

may not give a statistically meaningful range of results. DOE agreed[19] to evaluate spatial heterogeneity of hydrologic properties within hydrostratigraphic units and the effect this heterogeneity has on model results of unsaturated flow, seepage into the drifts, and transport.

Thermal-chemical effects on seepage are also neglected in the current abstraction approach (CRWMS M&O, 2000j), based on numerical simulations that show that any such changes will have a negligible effect on seepage and flow (CRWMS M&O, 2000m,q). However, uncertainties in the hydrological, thermal, and geochemical parameter values used in these simulations have not been adequately addressed in the drift-scale coupled processes model (CRWMS M&O, 2000t). DOE agreed[20] to evaluate the various sources of uncertainty in the thermal-hydrological-chemical process model, including details regarding how the propagation of various sources of uncertainty is calculated in a systematic uncertainty analysis; this evaluation will be documented in a revision to CRWMS M&O (2000t) or in another future document.

In summary, there are several concerns related to the propagation of data uncertainties in the abstraction of flow paths in the unsaturated zone. In each case, however, either the current DOE approach is reasonably bounding, the uncertainty is not expected to be of significant importance to performance predictions, or DOE agreed to provide additional information or analyses to support the abstraction approach.

3.3.6.4.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.6.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess flow paths in the unsaturated zone with respect to model uncertainty being characterized and propagated through the model abstraction.

To account for combined data and model uncertainty in the site-scale unsaturated zone flow model, 18 flow fields were originally defined for the basecase Total System Performance Assessment–Site Recommendation calculations (CRWMS M&O, 2000a). These flow fields consisted of three infiltration cases (lower, mean, and upper) within each of the three climate states (present-day, monsoon, and glacial transition), along with two different perched-water conceptual model: (i) a permeability-barrier model with reduced permeability in both fracture and matrix elements in the vicinity of the perched water and (ii) an unfractured zeolite model that eliminated fractures in all zeolitic units. Preliminary DOE calculations showed the difference between the two perched-water models was not significant (CRWMS M&O, 2000a, Figure 3.7-17), with the first model being slightly more conservative in predicting early arrival of

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[19]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001).” Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[20]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001).” Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

contaminants. Hence, only the nine flow fields based on the first perched-water model are carried forward to the Total System Performance Assessment–Site Recommendation.

Other sources of site-scale unsaturated zone flow model uncertainty are associated with the many assumptions and simplifications that must be made to model such a complex environment. For example, the assumption of homogenous layers implies that the model grid-block scale is larger than the scale of variability in hydrologic properties (heterogeneity). It is thus assumed that all grid blocks within any layer capture a comparable range of heterogeneity and, therefore, have the same average properties. DOE contends that the calibration process upscales the core-based measurements to the grid scale, thus accounting for intralayer heterogeneity at the subgrid scale. Based on the sparse data available, heterogeneity is not indicated in the Paintbrush nonwelded tuff at scales larger than the grid scale near the repository. Except for the Calico Hills nonwelded unit, the only heterogeneities considered in the model occur at layer interfaces and where layers are offset by faults. Within the Calico Hills nonwelded unit, layers are divided into either vitric or zeolitic rock types—which have significantly different hydrologic properties—based on borehole data and observations of perched water.

Staff are presently evaluating potential effects of lateral heterogeneity in the Paintbrush nonwelded tuff layer on the distribution of flow into the Topopah Spring welded tuff. Work by Ofoegbu, et al. (2001) indicated heterogeneity in the Paintbrush nonwelded tuff properties, caused by either depositional or secondary overprinting processes (e.g., small fault or slumping), could lead to increases in localized fluxes at the repository horizon. Currently, however, no field evidence exists that such effects dominate flow patterns at Yucca Mountain. The present DOE model indicates considerable lateral variability in the percolation flux reaching the proposed repository horizon (CRWMS M&O, 2000a, Figures 3.7-11 and 3.7-12), mainly as a result of the predicted spatial variability in net surface infiltration. DOE agreed[21] to evaluate spatial heterogeneity of hydrologic properties within hydrostratigraphic units and the effect this heterogeneity has on model results of unsaturated flow, seepage into the drifts, and transport.

Another important model uncertainty lies in the use of a steady-state infiltration boundary, which rests on the assumption that the Paintbrush nonwelded tuff layer acts to completely attenuate the infrequent pulses of infiltration predicted by the infiltration model. Indeed, DOE researchers have conducted modeling to demonstrate the validity of this assumption (e.g., CRWMS M&O, 1998, Section 2.4.2.8). Although these transient-flux models support the steady-state assumption, those presented to date have not used infiltration pulses that average more than 5 mm/yr [0.2 in/yr] during the long-term; yet infiltration during future climates may exceed 30 mm/yr [1.2 in/yr] over the proposed repository (CRWMS M&O, 2000a, Figure 3.7-11). Preliminary results of modeling conducted at the CNWRA indicate that, although the Paintbrush nonwelded tuff layer greatly attenuates episodic infiltration, transient percolation flux may occur at repository depth for infiltration pulses that occur every 5 years and average 10 mm/yr

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[21]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[0.4 in/yr] for the long term. To address this concern, DOE agreed[22] to provide additional documentation for the steady-state infiltration assumption.

A potential concern related to the grid scale of the site-scale unsaturated zone flow model is that the vertical length of model grid blocks at layer interfaces is typically much greater than the capillary-rise length scale (approximately the inverse of the van Genuchten α parameter). As a result, the numerical model may not be able to represent adequately lateral capillary diversion at layer interfaces. This concern pertains to the Paintbrush nonwelded tuff–Topopah Spring welded tuff interface, where capillary retention in the Paintbrush nonwelded tuff matrix may be greater than that of the Topopah Spring welded tuff fracture network. Preliminary modeling by Lawrence Berkeley National Laboratory staff using refined vertical grid discretization has simulated lateral capillary diversion in the Paintbrush nonwelded tuff.[23] However, there is little objective evidence that this phenomenon is occurring at the site (e.g., high matrix saturation or perched water above the Paintbrush nonwelded tuff–Topopah Spring welded tuff interface has not been observed). In fact, elevated matrix saturations occur in the uppermost welded unit of the Topopah Spring welded tuff. The difference noted between the highly discretized Lawrence Berkeley National Laboratory preliminary model and on-site observations may be that the model does not incorporate intralayer heterogeneity in the Paintbrush nonwelded tuff and Topopah Spring welded tuff that could interrupt lateral diversion or that the model does not represent adequately the gradational contact between the Paintbrush nonwelded tuff and the Topopah Spring welded tuff. Alternatively, the difference may be caused by the lack of direct flow connections in the model between the Paintbrush nonwelded tuff matrix and the underlying Topopah Spring welded tuff fractures. Thus, with present conditions, it is not expected that capillary lateral diversion in the lowermost Paintbrush nonwelded tuff layer would occur for scales larger than the model grid-block scale. If large-scale lateral diversion was to occur, possibly during future periods of greater infiltration, the likely effect would be to focus the flow into faulted zones. Such an effect could benefit performance if DOE could identify faulted zones at depth and avoid placement of waste packages in those areas. Both DOE and CNWRA researchers continue to investigate the potential for and possible effects of lateral capillary diversion in the Paintbrush nonwelded tuff. The permeability barrier at the contact between the Tiva Canyon welded unit and the Paintbrush nonwelded tuff is also being analyzed to assess the potential for lateral diversion above the Paintbrush nonwelded tuff where core data from surface-based boreholes indicate significantly elevated matrix saturations, including local saturation, in the lowermost Tiva Canyon welded unit layer. At present, however, it does not appear that exclusion of this process will result in overly optimistic performance estimates.

There are many model uncertainties in the drift seepage process model and drift seepage abstraction for total system performance assessment. The process model consists of uniformly sized grid cells of 0.5 m × 0.5 m × 0.5 m [1.6 ft × 1.6 ft × 1.6 ft], which implies an assumption that this volume contains a sufficient number of interconnected fractures to treat the fracture

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[22]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Unsaturated and Saturated Flow Under Isothermal Conditions (August 16–17, 2000)." Letter (September 8) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[23]Bodvarsson, G.S. Presentation at the DOE and NRC Technical Exchange and Management Meeting on Radionuclide Transport, December 5–7, 2000. Berkeley, California. 2000.

network as a three-dimensional continuum. The validity of this assumption is diminished in areas where spacings between water-bearing fractures are greater than a few tens of centimeters or where fractures tend to be near parallel with few intersections. As a result, the model may not calculate dripping that would occur in areas where seepage may be controlled more by fracture geometry than by fracture hydraulic properties. It is thus necessary to develop an improved understanding of the role of fracture characteristics in predicting drift seepage. Toward that goal, DOE agreed to relate any observed seepage in the passive East-West Cross Drift tests to full periphery maps of fractures and to provide a three-dimensional representation of fracture characterization in documentation of ongoing Alcove 8–Niche 3 seepage testing.[24] A desirable outcome of this effort is that drift seepage studies at Yucca Mountain will be fracture- informed so the rates and spatial distributions of drift seepage can be related, at least qualitatively, to observed fracture characteristics (e.g., aperture variability, trace length, density, interconnectedness, orientation, and location of intersection with drifts). Thus, if construction of a repository at Yucca Mountain proceeds, a qualitative basis would exist for evaluating whether fracture patterns in drifts are consistent with those used in the seepage studies used to validate the drift seepage abstraction for total system performance assessment.

Another important model uncertainty in the drift seepage process model is whether the use of the van Genuchten–Mualem model for moisture retention and relative permeability is adequate to model unsaturated flow in a fracture network. For the rather low unsaturated zone percolation fluxes predicted for Yucca Mountain, film flow may be the dominant flow regime. Film flow is a term used to describe flow on fracture surfaces that does not bridge the fracture aperture. Conditions that affect capillary diversion and dripping may be quite different for film flow than are currently modeled. One reason for concern is that parameter estimates obtained from the relatively high flow rate injection tests in Niche 3650 may not be applicable to ambient repository conditions. To address this concern, DOE agreed to either consider film flow processes in the seepage abstraction or to provide justification that the current model approach is adequate to bound this uncertainty.[25] Results from the Alcove 8–Niche 3 test and the East-West Cross Drift passive test may also help resolve this concern.

Modeling assumptions used to evaluate potential effects on seepage flux of rock bolts and changes in drift-geometry represent another source of model uncertainty. The DOE simulations (CRWMS M&O, 2000g) suggested a moderate increase of drift seepage as a result of partial drift degradation. Accordingly, seepage flow rates are increased by a factor of 1.55 in the seepage abstraction to account for the combined effects from partial drift degradation and rock bolts (CRWMS M&O, 2000d). A concern is that the grid scale of the process model used to estimate this adjustment factor is not sufficiently small to account for the scale of asperities in drift geometry caused by rockfall. Scales comparable to the inverse of the van Genuchten α parameter are appropriate, so seepage is not underpredicted for small-scale asperities. To

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[24]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Structural Deformation and Seismicity (October 11–12, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[25]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Unsaturated and Saturated Flow Under Isothermal Conditions (August 16–17, 2000)." Letter (September 8) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

address NRC concerns related to the scale of the model grid used to assess the effects of drift collapse on seepage, DOE agreed[26] to consider smaller-scale tunnel irregularities or, alternatively, to provide justification that the current approach is adequate. Recent observations in the passive seepage test in the East-West Cross-Drift[27] suggest that rock bolts (and other foreign objects such as ventilation ducts and utility lines) appear to attract moisture, and several plate-sized puddles were observed beneath rock bolts.[28] It does not appear that these objects need to be in direct contact with the rock, thus, the observed moisture may be caused by vapor condensation. DOE agreed[29] to provide a technical basis for representation of or the neglect of dripping from rockbolts in performance assessment models.

DOE explicitly considers uncertainty regarding the conceptual model for flow focusing within unsaturated zone subregions in the drift seepage abstraction for total system performance assessment. Two different conceptual models for flow focusing are used to estimate upper- and lower-bounds weep spacings. The upper-bound weep spacing is based on an assumption that actively flowing fractures are saturated (CRWMS M&O, 2000h, Section 6.3.3.1); the lower-bound weep spacing is based on partially saturated fractures using the active-fracture conceptual model of Liu, et al. (1998). Based on analyses of potential weep spacings for the two different conceptual models, DOE developed statistical distributions from which the values of the flow focusing factors are sampled for individual total system performance assessment simulations (CRWMS M&O, 2000h, Section 6.3.3.2). Three different distributions for the focusing factor were developed, corresponding to total system performance assessment analyses for the low-, medium-, and high-infiltration scenarios. All three distributions are log-uniform with a lower bound of 1.0. The upper-bound values for the focusing factor distributions are 47, 22, and 9.7 for the low-, medium-, and high-infiltration scenarios, respectively. One concern with this approach is that the focusing factor distributions are based purely on theoretical considerations, and no consideration is given to how flow focusing may be affected by fracture patterns in the proposed repository host horizon. The DOE agreement to relate analyses of ongoing seepage studies to observed fracture parameters[30] should

[26]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Unsaturated and Saturated Flow Under Isothermal Conditions (August 16–17, 2000)." Letter (September 8) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[27]Observations from Cross-Drift Bulkheads Opening January 22–25, 2001, compiled by David Hudson, U.S. Geological Survey.

[28]Glenn, C. Personal communication (February 2001) to N. Coleman, NRC Project Manager. Las Vegas, Nevada: NRC. 2001.

[29]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[30]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Structural Deformation and Seismicity (October 11–12, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

address this concern. In addition, DOE agreed[31] to use the field test data to provide additional confidence in the seepage abstraction and associated parameter values.

Based on measurements of air permeability, DOE suggests the process of seepage into drifts may be influenced by a 1-m [3.3-ft] thick excavation-induced disturbed zone with increased permeability around drifts (CRWMS M&O, 2000f). This zone of enhanced permeability is postulated to be the effect of dilation of existing fractures rather than the formation of new fractures. No technical basis has been presented, however, to indicate the presence or extent of excavation-induced fractures or new fracture connections. To address this concern, DOE agreed[32] to document data and interpretations regarding excavation-induced fractures in the Exploratory Studies Facility and in the Enhanced Characterization of the Repository Block Cross Drift.

DOE process modeling predicts seepage fractions to be higher when percolation flux is episodic (CRWMS M&O, 2000g, Section 6.6.7), but the unsaturated zone process model report suggests high-frequency fluctuations of infiltration will not reach the potential repository because the Paintbrush nonwelded tuff layers attenuate transient flow. As discussed in Section 3.3.4 of this report, however, the process models used to support this suggestion use average infiltration rates much lower than those expected for future climates. Thus, the validity of the steady-state flow assumption in seepage process models remains an important source of uncertainty that is not propagated through total system performance assessment abstraction. As previously mentioned, DOE agreed[33] to provide additional justification for the steady-state flow assumption, and the effectiveness of the Paintbrush nonwelded tuff to dampen episodic flow. DOE described an approach by which consideration of episodic flow can be considered in the seepage abstraction if the necessary additional justification cannot be provided (CRWMS M&O, 2000h, Section 6.3.4).

Below the proposed repository, where perched water occurs above and within the Calico Hills nonwelded unit, the unsaturated zone model predicts significant lateral diversion of water toward faults where flow to the water table is relatively rapid. The model predicts 35 percent of flow within the entire unsaturated zone model domain reaching the water table via fast flow in faults (CRWMS M&O, 2000a). If a similar percentage is applicable to the proposed repository footprint, it would be reasonable to conclude the total system performance assessment model abstraction does not benefit from undue credit for matrix flow below the proposed repository.

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[31]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[32] Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Structural Deformation and Seismicity (October 11–12, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[33]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Unsaturated and Saturated Flow Under Isothermal Conditions (August 16–17, 2000)." Letter (September 8) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

To further reduce this source of uncertainty, DOE agreed[34] to provide an analysis of data used to support model predictions of the flow field below the repository, particularly in the nonwelded vitric portions of the Calico Hills, Prow Pass, and Bullfrog hydrostratigraphic units.

The DOE multiscale thermohydrologic model (CRWMS M&O, 2000u) uses only the drift-scale property sets to calculate thermohydrologic variables, and it is not clear how this captures the variability and uncertainty seen in predictions using other property sets or the uncertainty in comparisons to actual test results. Note that all thermal tests to date at Yucca Mountain have been conducted in the Tsw34 unit so that all conclusions from the thermal tests thermal-hydrological model (CRWMS M&O, 2000r) apply only to that unit. If the analyses were performed on the remaining geological units, the predicted variability would be greater. To address this concern, DOE agreed[35] to represent the full variability/uncertainty in results of the thermal effects on flow simulations in the abstraction of thermodynamic variables to other models or provide technical basis that a reduced representation is appropriate. DOE also agreed[36] to provide a revision to the unsaturated zone flow and transport process model report that includes consideration of model uncertainties: (i) types of model uncertainty, (ii) flow conceptualization for ambient conditions, (iii) flow conceptualization for thermal conditions, (iv) fracture flow for ambient and thermal conditions, (v) fracture matrix interaction model evolution, (vi) discrete fracture description, and (vii) reduction of model uncertainty.

As previously mentioned, the DOE abstractions of unsaturated zone flow and drift seepage neglect thermal-hydrological-chemical-induced changes to hydrological properties based on numerical simulations that show such changes will have a negligible effect on seepage and flow (CRWMS M&O, 2000a,c). Conceptual model uncertainties in these simulations have not been adequately addressed in the Drift-Scale Coupled Processes Model (CRWMS M&O, 2000t). To address this concern, DOE agreed[37] to provide an evaluation of the various sources of uncertainty in the thermal-hydrological-chemical process model, including details how the propagation of various sources of uncertainty are calculated in a systematic uncertainty analysis. In addition, DOE agreed[38] to provide additional information about the treatment of fully dry conditions in the reactive transport simulations, including information about the amount

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[34]Schlueter, J.R. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Unsaturated and Saturated Flow Under Isothermal Conditions (August 16–17, 2000).” Letter (September 8) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[35]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Thermal Effects on Flow (January 8–9, 2001).” Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[36]Ibid.

[37]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001).” Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[38]Ibid.

of unreacted solute mass trapped in the dryout zone, as well as how this would affect precipitation of solutes and the resulting change in hydrological properties.

In summary, there are several concerns related to the consideration of model uncertainties in the abstraction of flow paths in the unsaturated zone. To address these concerns, DOE agreed to provide additional information or analyses to support the abstraction approach. This additional information includes justification for using a steady-state infiltration boundary; an evaluation of data to support the flow fields below the repository; consideration of fracture patterns, low flow-regime processes, and small-scale tunnel irregularities in the seepage abstraction; and consideration of parameter and model uncertainty in the multiscale thermohydrologic model and in the thermal-hydrological-chemical process model. NRC continues to evaluate the potential effects of heterogeneity in the unsaturated zone, which will be of greater importance if DOE used its refined-grid model with enhanced capillary diversion in the Paintbrush Tuff in the performance assessment abstraction.

3.3.6.4.5 Model Abstraction Output Is Supported by Objective Comparisons

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.6.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess flow paths in the unsaturated zone with respect to model abstraction output being supported by objective comparisons.

The low-, medium-, and high-infiltration scenarios for the unsaturated zone flow model are calibrated using one- and two-dimensional inverse methods to match observations of pneumatic signals between boreholes, core saturation data from laboratory measurements, and *in-situ* moisture potential profiles (CRWMS M&O, 2000m). Additional fine-tuning of the model was performed to match observations of perched water associated with the Calico Hills nonwelded unit layer. Thus, the flow model scenarios are reasonably consistent with those observations. However, supporting data for the predicted flow vectors within, adjacent to, and below the perched water were not presented in the process model report of analysis and model reports. DOE agreed[39] to provide documentation of the analysis of available data to validate the predicted three-dimensional unsaturated zone model flow fields below the repository footprint, particularly below the perched water or through the vitric Calico Hills nonwelded unit, Prow Pass, and Bullfrog hydrostratigraphic units.

DOE obtained additional model validation from two modeling exercises to show the unsaturated zone flow model is broadly consistent with the observed distribution of calcite minerals in Well WT–24 and with chloride concentrations in the subsurface. Geochemical modeling of calcite precipitation was conducted to provide validation of deep percolation rates simulated in the unsaturated zone flow model (CRWMS M&O, 2000v). The result for a range of infiltration rates 2–20 mm/yr [0.08–0.8 in/yr] was that simulated calcite distributions agree reasonably well with measured data from Well WT–24 cuttings. The DOE modelers assume the amount of

[39]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Unsaturated and Saturated Flow Under Isothermal Conditions (August 16–17, 2000)." Letter (September 8) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

calcite precipitation generally increases as percolation increases. The simulated calcite abundances also are sensitive to the assumed water and gas chemistry, vapor movement, reaction kinetics, and mineralogy. The analysis provides some constraints on hydrological parameters, percolation flux, and additional evidence for validation of the flow and transport model. This analysis cannot give a definite value or a narrow range of values, however, because of the dependence of calcite deposition on the other factors. DOE agreed[40] to document the results of the calcite filling observations.

Another simulation was conducted to compare the basecase unsaturated zone flow model with observed chloride data from the Exploratory Studies Facility and East-West Cross Drift. Chloride concentrations from the steady-state transport simulation were compared with measured pore water chloride concentration data (CRWMS M&O, 2000v). The results indicate that measured chloride concentrations show a smaller range than predicted by the modern infiltration rates during steady-state conditions (CRWMS M&O, 2000a, Figure 3.8-3). However, because many measured chloride concentrations are fit closely by the model results, it appears the mean infiltration rate is approximately correct. Differences between measured and modeled chloride concentrations in the high- and low-infiltration regions suggest the time-averaged infiltration rates may be more uniform than predicted by the unsaturated zone flow model. Conversely, L. Flint[41] correlated the systematic measurements of water potential in the East-West Cross Drift and the chloride concentration of matrix pore water to shallow infiltration estimates. It was found percolation estimates from water potential data and from the chloride mass balance method both matched the magnitude and heterogeneity of the highly discretized shallow infiltration model results, except under washes where the model underpredicted percolation estimates from the East-West Cross Drift data.

A rigorous demonstration that the seepage model for total system performance assessment abstraction is valid for its intended purpose would require testing model results against relevant data not used in the original development of the model. For the seepage model for total system performance assessment, these data should include percolation flux at low flow rates for periods of years, even hundreds of years, in many locations in the repository. Unfortunately, such data are not available. Further, data for adequate validation would need to include the wide range of conditions such as drift degradation and collapse with time; those data are not available either. As previously mentioned, DOE agreed[42] to conduct and provide results from several ongoing field studies and modeling studies to increase confidence for the abstraction approach. Of particular importance is an ongoing field test in the East-West Cross Drift in which an approximately 1-km [0.62-mi] section of the tunnel has been sealed off from

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[40]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[41]Flint, L. "Measuring Flow and Transport in Unsaturated Fractured Rocks: A Large-Scale Unsaturated Flow Experiment." *Presentation to Geological Society of America November 13–17, 2000*. Reno, Nevada. 2000.

[42]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Unsaturated and Saturated Flow Under Isothermal Conditions (August 16–17, 2000)." Letter (September 8) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

ventilation and is being allowed to return to ambient conditions. DOE agreed[43] to consider smaller scale tunnel irregularities in drift collapse or justify that the current approach is adequate. DOE also agreed[44] to consider the NRC suggestion of comparing the numerical model results with the Phillips (1996) analytical solution as a means of model validation.

In summary, the site-scale unsaturated zone flow model of Yucca Mountain is broadly consistent with DOE interpretations of empirical observations. Because of model complexity, however, alternate interpretations of these observations are possible and model parameters can be adjusted to match a wide range of possible results. Consequently, DOE agreed to propagate data and model uncertainty through the abstraction, as discussed in the preceding sections.

3.3.6.5 Status and Path Forward

Table 3.3.6-1 provides the status of all key technical issue subissues, referenced in Section 3.3.6.2, for the Flow Paths in the Unsaturated Zone Integrated Subissue. The table also provides the related DOE and NRC agreements pertaining to the Flow Paths in the Unsaturated Zone Integrated Subissue. The agreements listed in the table are associated with one or all five generic acceptance criteria discussed in Section 3.3.6.4. Note that the status and the detailed agreements (or path forward) pertaining to all the key technical issue subissues are provided in Table 1.1-3 and Appendix A.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (through specified testing, analyses, and the like), acceptably addresses the NRC questions so that no information beyond that provided, or agreed to, will likely be required at the time of a potential license application.

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[43]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Unsaturated and Saturated Flow Under Isothermal Conditions (August 16–17, 2000)." Letter (September 8) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[44]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Thermal Effects on Flow (January 8–9, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

Table 3.3.6-1. Related Key Technical Issue Subissues and Agreements			
Key Technical Issue	**Subissue**	**Status**	**Related Agreements***
Unsaturated and Saturated Flow Under Isothermal Conditions	Subissue 4—Deep Percolation	Closed-Pending	USFIC.4.01 through USFIC.4.07
Radionuclide Transport	Subissue 1—Radionuclide Transport Through Porous Rock	Closed-Pending	RT.1.01
	Subissue 3—Radionuclide Transport Through Fractured Rock	Closed-Pending	RT.3.02 RT.3.05 RT.3.06
Structural Deformation and Seismicity	Subissue 3—Fracturing and Structural Framework of the Geologic Setting	Closed-Pending	SDS.3.01 SDS.3.02 SDS.3.04
Thermal Effects on Flow	Subissue 1—Features, Events, and Processes Related to Thermal Effects on Flow	Closed-Pending	None
	Subissue 2—Thermal Effects on Temperature, Humidity, Saturation, and Flux	Closed-Pending	TEF.2.01 TEF.2.06 TEF.2.07 TEF.2.08 TEF.2.10 TEF.2.11 TEF.2.12 TEF.2.13
Evolution of the Near-Field Environment	Subissue 1—Effects of Coupled Thermal-Hydrological-Chemical Processes on Seepage and Flow	Closed-Pending	ENFE.1.03 ENFE.1.04 ENFE.1.05
Repository Design and Thermal-Mechanical Effects	Subissue 2—Design of the Geologic Repository Operations Area for the Effects of Seismic Events and Direct Fault Disruption	Closed-Pending	None
	Subissue 3—Thermal-Mechanical Effects on Underground Facility Design and Performance	Closed-Pending	RDTME.3.14 RDTME.3.20 RDTME.3.21
Total System Performance Assessment and Integration	Subissue 1—System Description and Demonstration of Multiple Barriers	Closed-Pending	None
	Subissue 2—Scenario Analysis and Event Probability	Closed-Pending	TSPAI.2.01 TSPAI.2.02

Table 3.3.6-1. Related Key Technical Issue Subissues and Agreements (continued)

Key Technical Issue	Subissue	Status	Related Agreements*
Total System Performance Assessment and Integration	Subissue 3—Model Abstraction	Closed-Pending	TSPAI.3.07 TSPAI.3.11 TSPAI.3.22 through TSPAI.3.27
	Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards	Closed-Pending	None

*Related DOE and NRC agreements are associated with one or all five generic acceptance criteria.

Note: Key Technical Issue Agreement GEN.1.01 pertains to multiple integrated subissues, as well as to some specific issues related to this integrated subissue.

3.3.6.6 References

Birkholzer, J., G. Li, C.F. Tsang, and Y. Tsang. “Modeling Studies and Analysis of Seepage into Drifts at Yucca Mountain.” *Journal of Contaminant Hydrology.* Vol. 38. pp. 349–384. 1999.

CRWMS M&O. “Chapter 2: Unsaturated Zone Hydrology Model. Total System Performance Assessment—Viability Assessment (TSPA-VA) Analyses Technical Basis Document.” B00000000–01717–4301–00011. Revision 00. Las Vegas, Nevada: TRW Environmental Safety Systems, Inc. 1998.

———. “Unsaturated Zone Flow and Transport Model PMR.” TDR–NBS–HS–000002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000a.

———. “Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations.” TDR–WIS–RL–000001. Revision 04. Las Vegas, Nevada: CRWMS M&O. 2000b.

———. “Total System Performance Assessment (TSPA) for the Site Recommendation.” TDR–WIS–PA–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000c.

———. “Development of Numerical Grids for UZ Flow and Transport Modeling.” ANL–NBS–HS–000015. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000d.

———. “Radionuclide Transport Models Under Ambient Conditions.” MDL–NBS–HS–000008. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000e.

———. “Seepage Calibration Model and Seepage Test Data.” MDL–NBS–HS–000004. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000f.

———. “Seepage Model for PA Including Drift Collapse.” MDL–NBS–HS–000004. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000g.

———. “Abstraction of Drift Seepage.” ANL–NBS–HS–000005. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000h.

———. “Abstraction of NFE Drift Thermodynamic Environment and Percolation Flux.” ANL–EBS–HS–000003. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000i

———. “Total System Performance Assessment (TSPA) Model for Site Recommendation.” MDL–WIS–PA–000002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000j.

———. “Calculation of Permeability Change Due to Coupled Thermal-Hydrological-Mechanical Effects.” CAL–NBS–MD–000002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000k.

———. “Features, Events, and Processes in UZ Flow and Transport.” ANL–NBS–MD–000001. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000l.

———. “Calibrated Properties Model.” ANL–NBS–HS–000003. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000m.

———. *“In-Situ* Field Testing of Processes.” ANL–NBS–HS–000005. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000n.

———. “Ventilation Model.” ANL–EBS–MD–000030. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000o.

———. “Development Plan or the Ventilation Test.” TWP–EBS–MD–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000p.

———. “Analysis of Hydrologic Properties Data.” ANL–NBS–HS–000002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000q.

———. “Thermal Tests Thermal-Hydrological Analyses/Model Report.” ANL–NBS–TH–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000r.

———. “Geologic Framework Model (GFM3.1).” MDL–NBS–GS–000002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000s.

———. “Drift-Scale Coupled Processes (Drift-Scale Test and Thermal-Hydrological-Chemical Seepage) Models.” MDL–NBS–HS–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000t.

———. “Multi-scale Thermohydrologic Model.” ANL–EBS–MD–000049. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000u.

———. “Analysis of Geochemical Data for the Unsaturated Zone.” ANL–NBS–GS–000004. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000v.

Flint, L.E. “Characterization of Hydrogeologic Units Using Matrix Properties, Yucca Mountain, Nevada.” U.S. Geological Survey Water Resources Investigations Report 97-4243. 1998.

LeCain, G.D. “Air-Injection Testing in Vertical Boreholes in Welded and Nonwelded Tuff, Yucca Mountain, Nevada.” U.S. Geological Survey Water Resources Investigations Report 96-4262. 1997.

Liu, H.H., C. Doughty, and G.S. Bodvarsson. “An Active Model for Unsaturated Flow and Transport in Fractured Rocks.” *Water Resources Research.* Vol. 34, No. 10. pp. 2,633–2,646. 1998.

McLaughlin, D. and L.R. Townley. “A Reassessment of the Groundwater Inverse Problem.” *Water Resources Research.* Vol. 32, No. 5. pp. 1,131–1,161. 1996.

NRC. “Issue Resolution Status Report, Key Technical Issue: Unsaturated and Saturated Flow Under Isothermal Conditions.” Revision 2. Washington, DC: NRC. 1999.

———. “Issue Resolution Status Report, Key Technical Issue: Radionuclide Transport.” Revision 2. Washington, DC: NRC. 2000a.

———. “Issue Resolution Status Report, Key Technical Issue: Structural Deformation and Seismicity.” Revision 3. Washington, DC: NRC. 2000b.

———. “Issue Resolution Status Report, Key Technical Issue: Thermal Effects on Flow.” Revision 3. Washington, DC: NRC. 2000c.

———. “Issue Resolution Status Report, Key Technical Issue: Repository Design and Thermal-Mechanical Effects.” Revision 3. Washington, DC: NRC. 2000d.

———. “Issue Resolution Status Report, Key Technical Issue: Evolution of the Near-Field Environment.” Revision 3. Washington, DC: NRC. 2000e.

———. “Issue Resolution Status Report Key Technical Issue: Total System Performance Assessment and Integration.” Revision 3. Washington, DC: NRC. 2000f.

———. NUREG–1804, "Yucca Mountain Review Plan—Draft Report for Comment." Revision 2. Washington, DC: NRC. March 2002.

Ofoegbu, G.I., S. Painter, R. Chen, R.W. Fedors, and D.A. Ferrill. “Geomechanical and Thermal Effects on Moisture Flow at the Proposed Yucca Mountain Nuclear Waste Repository.” *Nuclear Technology*. Vol. 13, No. 3. pp. 241–262. 2001.

Phillips, O.M. “Infiltration of a Liquid Finger Down a Fracture into Superheated Rock.” *Water Resources Research*. Vol. 32, No. 6. pp. 1,665–1,670. 1996.

3.3.7 Radionuclide Transport in the Unsaturated Zone

3.3.7.1 Description of Issue

The radionuclide transport in the unsaturated zone model abstraction addresses the migration of radionuclides through the unsaturated zone below the repository to the water table after waste package failure. The transport path through the unsaturated zone is defined to begin at the edge of the drift/invert part of the engineered barrier subsystem. The rate radionuclides migrate through the unsaturated zone depends on the medium through which the radionuclides travel—fractured rock or porous rock. This migration rate also depends on the water chemistry and mineralogy of the system because these control retardation processes. The relationship of this integrated subissue to other subissues is depicted in Figure 3.3.7-1. This figure shows the relationship between the radionuclide transport in the unsaturated zone model abstraction, the radionuclide release rates and solubility limits (see Section 3.3.4), and flow paths in the unsaturated zone (see Section 3.3.6) model abstractions. The overall organization and identification of all the integrated subissues are depicted in Figure 1.2-2.

3.3.7.2 Relationship to Key Technical Issue Subissues

Radionuclide transport in the Unsaturated Zone Integrated Subissue incorporates subject matter previously captured in the following key technical issue subissues:

- Radionuclide Transport: Subissue 1—Radionuclide Transport Through Porous Rock (NRC, 2000a)

- Radionuclide Transport: Subissue 2—Radionuclide Transport Through Fractured Rock (NRC, 2000a)

- Radionuclide Transport: Subissue 3—Radionuclide Transport Through Fractured Rock (NRC, 2000a)

- Radionuclide Transport: Subissue 4—Nuclear Criticality in the Far Field (NRC, 2000a)

- Unsaturated and Saturated Flow Under Isothermal Conditions: Subissue 4—Deep Percolation [Present and Future (Post-thermal Period)] (NRC, 2000b)

- Unsaturated and Saturated Flow Under Isothermal Conditions: Subissue 6—Matrix Diffusion (NRC, 2000b)

- Evolution of the Near-Field Environment: Subissue 3—The Effects of Coupled Thermal-Hydrologic-Chemical Processes on Chemical Environment for Radionuclide Release (NRC, 2000c)

- Evolution of the Near-Field Environment: Subissue 4—The Effects of Coupled Thermal-Hydrologic-Chemical Processes on Radionuclide Transport through Engineered and Natural Barriers (NRC, 2000c)

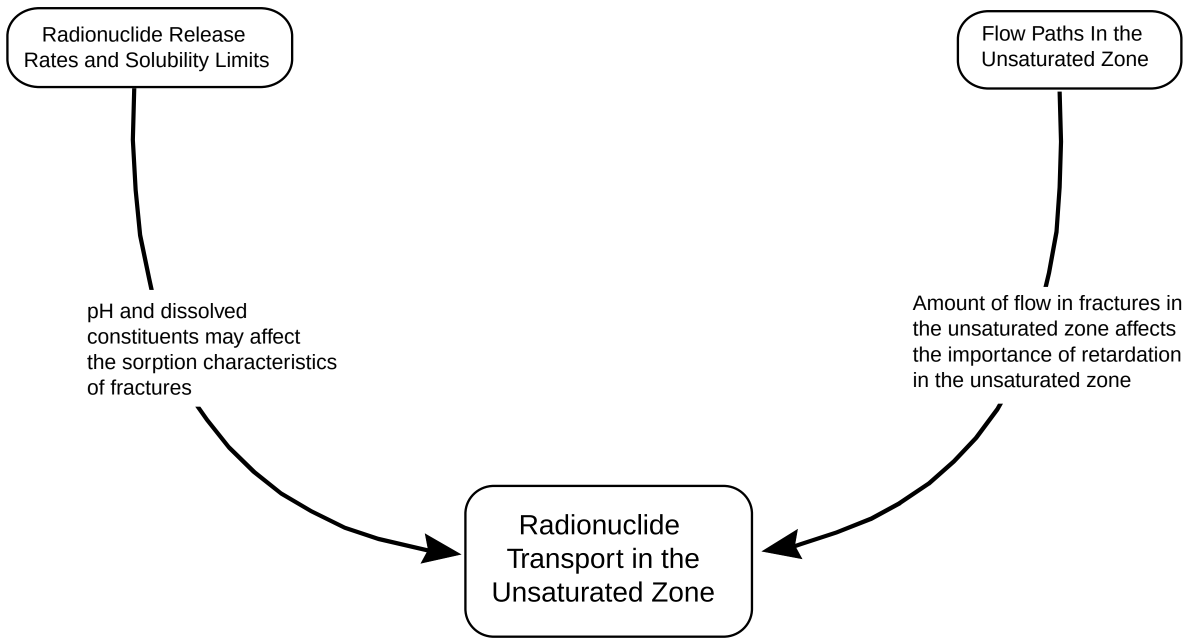

Figure 3.3.7-1. Diagram Illustrating the Relationship Between Radionuclide Transport in the Unsaturated Zone and Other Model Abstractions

- Structural Deformation and Seismicity: Subissue 3—Fracturing and Structural Framework of the Geological Setting (NRC, 2000d)

- Thermal Effects on Flow: Subissue 2—Is the DOE Thermohydrologic Modeling Approach Sufficient to Predict the Nature and Bounds of Thermal Effects on Flow in the Nearfield?(NRC, 2000e)

- Total System Performance Assessment and Integration: Subissue 1—System Description and Demonstration of Multiple Barriers (NRC, 2000f)

- Total System Performance Assessment and Integration: Subissue 2—Scenario Analysis and Event Probability (NRC, 2000f)

- Total System Performance Assessment Integration: Subissue 3—Model Abstraction (NRC, 2000f)

- Total System Performance Assessment and Integration: Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards (NRC, 2000f)

The key technical issue subissues formed the bases for the previous versions of the issue resolution status reports and also were the bases for technical exchanges with DOE where agreements were reached on what additional information DOE needed to provide to resolve the subissue. The resolution status of this integrated subissue is based on the resolution status of each of the contributing key technical issue subissues. The subsequent sections incorporate applicable portions of these key technical issues subissues, however, no effort was made to explicitly identify each subissue.

3.3.7.3 Importance to Postclosure Performance

One aspect of risk informing the NRC review was to determine how this integrated subissue is related to the DOE repository safety strategy. DOE identifies radionuclide transport in the unsaturated zone at Yucca Mountain in Revision 4.0 of the repository safety strategy (CRWMS M&O, 2000a) as a principal factor of the current postclosure safety case. In the DOE model abstraction, radionuclide transport in fractures in the volcanic tuffs is conservatively considered to be unretarded because of limited characterization regarding the distribution of fracture-lining minerals (CRWMS M&O, 2000b). The DOE conceptual model for radionuclide transport in the unsaturated zone in total system performance assessment is that delay of radionuclide migration by sorption onto minerals in the volcanic tuffs occurs only within the rock matrix where solutes enter only by matrix diffusion. Sorption parameters are based on a combination of batch experiments and expert elicitation (CRWMS M&O, 2000c).

The DOE approach for considering radionuclide transport in the unsaturated zone is essentially the same approach used previously in DOE (1998a). Transport parameter values, represented by sorption coefficient (K_d) probability distribution functions, have been modified slightly from CRWMS M&O (2000c). Other changes include using updated parameter values and inputs from the unsaturated zone flow model and incorporation of the active-fracture conceptual model.

Because the conceptual model only provides for retardation in the matrix, the process of matrix diffusion is an important factor in the abstraction of radionuclide transport in the unsaturated zone. In sensitivity analyses performed by the DOE for the Total System Performance Assessment–Site Recommendation, the mean dose rate from the undisturbed basecase was compared with a case with no matrix diffusion in the unsaturated zone and with a case where anion and cation matrix diffusion coefficients were set at 100 times the matrix diffusion coefficients in the basecase (CRWMS M&O, 2000b, Section 5.2.6.1). Results showed that matrix diffusion in the unsaturated zone has a moderate effect on the dose history, especially between 20,000 and 30,000 years, where dose rates predicted for the no-matrix-diffusion case exceed those for the basecase by as much as two orders of magnitude. Conversely, differences in predicted dose rates are negligible between the basecase and the case with matrix diffusion coefficients 100 times the basecase values.

3.3.7.4 Technical Basis

NRC developed a plan (2002) consistent with the acceptance criteria and review methods found in previous issue resolution status reports. A review of DOE approaches for including radionuclide transport in the unsaturated zone in total system performance assessment abstractions is provided in the following subsections. The review is organized according to the five acceptance criteria identified in Section 1.5: (i) System Description and Model Integration Are Adequate, (ii) Data Are Sufficient for Model Justification, (iii) Data Uncertainty Is Characterized and Propagated Through the Model Abstraction, (iv) Model Uncertainty Is Characterized and Propagated Through the Model Abstraction, and (v) Model Abstraction Output Is Supported by Objective Comparisons.

3.3.7.4.1 System Description and Model Integration Are Adequate

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.7.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess radionuclide transport in the unsaturated zone with respect to system description and model integration.

DOE is handing the abstraction of unsaturated zone radionuclide transport for the total system performance assessment (CRWMS M&O, 2000d) through a residence-time transfer function adapted to the FEHM particle-tracking algorithm (Zyvoloski, et al. 1997). The residence-time transfer function approach is a particle-tracking method that describes a cumulative probability distribution function of particle residence times that accounts for the influence of advective transport in fracture networks and rock matrix and diffusive transport of solutes from fractures into rock matrix. The travel time of any given particle through a particular cell is computed by generating a random number between 0 and 1 and determining the corresponding residence time from the residence-time transfer function. On average, if a large number of particles travel through this portion of the model domain, the cumulative residence time distribution of particles will reproduce the shape of the transfer function (CRWMS M&O, 2000d). After spending the assigned residence-time in a model cell, a particle then moves from the resident cell to an adjoining cell, randomly, with the probability of entering an adjoining cell set according to the proportion of efflux from the resident cell into each of the adjoining cells (CRWMS M&O, 2000e), as determined by flow fields derived from the site-scale unsaturated zone flow model.

The residence-time transfer function used to assign particle residence times for transport in the fracture continuum is based on the analytical solution of Sudicky and Frind (1982), which takes into account advective transport in the fractures, molecular diffusion from the fracture to the porous matrix, adsorption on the fracture face, and adsorption within the matrix (CRWMS M&O, 2000c). Although this method allows consideration of solute sorption on fracture surfaces, this option is not used in the unsaturated zone transport abstraction model because of the lack of conclusive information about sorption in fractures and the anticipated small impact on model predictions (CRWMS M&O, 2000e). This approach is conservative with respect to repository performance.

A significant change from the unsaturated zone radionuclide transport model abstraction used for the Total System Performance Assessment–Viability Assessment is the incorporation of the active-fracture concept described by Liu, et al. (1998). The active-fracture concept accounts for the fact that not all fractures in an unsaturated flow system actively conduct water, and the number of active fractures in a flow system increases with increased flow rate. As described in CRWMS M&O (2000d), the active-fracture concept is implemented in the transport model by adjusting the flow interval spacing in the transport equation according to the equation

$$B = B_g S_e^{-\gamma} \tag{3.3.7-1}$$

where B is the adjusted flowing interval spacing; B_g is the geometric fracture spacing; S_e is the effective fracture saturation ($0 \leq S_e \leq 1$); and γ is the active-fracture fitting parameter ($0 \leq \gamma \leq 1$). The effect of incorporating the active-fracture conceptual model is that the effective flowing interval spacing is considerably larger when fracture saturations are low, which is generally the case for units such as the Topopah Spring Tuff. Larger flow interval spacing translates into less matrix diffusion because there is less available fracture-matrix interface area and greater isolation of the rock matrix between flowing intervals. In nonwelded vitric units, where flow is predominantly in rock matrix, the process of matrix diffusion would be of little benefit to performance. Although the active-fracture approach is a reasonable conceptual model, the methods of model parameter estimation and the numerical implementation of the transport model are not transparent in the analysis and model report (CRWMS M&O, 2000d). For example, it is not clear how fracture spacing, fracture porosity, and mean fracture aperture values in Table 3 of the analysis and model report (CRWMS M&O, 2000d) are derived. The mean fracture aperture values given in the analysis and model report seem quite large, but there is no discussion of how they relate to aperture measurements at depth; if the listed aperture values have been adjusted to account for the active-fracture concept, it is not stated in the analysis and model report. Also, it is not clear how or whether the fraction of active fractures is factored into the calculation of fluid velocity in the transport model. It would seem that velocity must increase for a given flux if the number of active fractures is reduced, but calculation of velocity is not discussed in the analysis and model report. DOE agreed[1] to provide independent lines of evidence to support the use of the active fracture model continuum concept in the transport model.

DOE relies on linear sorption isotherms and represents all retardation processes using K_d (CRWMS M&O, 2000b,c,e,f). Sorption coefficients for the radionuclides of interest are selected based on an initial and informal expert elicitation conducted for Total System Performance Assessment–1993, involving three experts (Wilson, et al., 1994). The sorption parameters probability distribution functions were constrained, assuming that water from saturated volcanic tuff (Well J–13) and the Paleozoic (UE–25p#1) aquifer bound the chemistry of the groundwaters at Yucca Mountain. Total System Performance Assessment–1993 used only geochemical information indirectly through expert elicitation to estimate probability distribution functions for K_d and did not explicitly incorporate geochemistry or geochemical modeling

[1]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

results. The approach has remained essentially unchanged since Total System Performance Assessment–1993, although the specific constraints on the transport parameters have been modified, particularly for uranium, neptunium, and plutonium (Wilson, et al., 1994; Triay, et al., 1997; CRWMS M&O, 2000c). Sorption probability distribution functions are abstracted into four rock types: devitrified, vitric, and zeolitic tuff, and iron oxide. The iron oxide is intended to represent waste package corrosion products and is not used to simulate retardation by fracture-lining minerals. Radionuclide retardation is related to K_d, the sorption coefficient, by the equation

$$R_f = 1 + \frac{\rho_b}{n} K_d \tag{3.3.7-2}$$

where R_f is the retardation factor, ρ_b is the bulk density, and n is the porosity. This equation is for saturated flow. For unsaturated flow, the moisture content, θ, is substituted for n. Retardation by adsorption is assumed to occur only in the matrix, and the degree to which retardation contributes to overall repository performance depends on the nature of coupling between the matrix/fracture.

The technical basis for selecting radionuclides for transport modeling via reversible and irreversible colloid attachment is not transparent and traceable in all cases. The analysis and model report (CRWMS M&O, 2000g) identifies radionuclides for the total system performance assessment model abstraction based on contribution to dose, inventory, and mobility considerations, but does not explicitly identify those radionuclides that will be transported as colloids. Discussions in the analysis and model report (CRWMS M&O, 2000b,f,h) do not fully consider the possibility that waste form colloids could significantly transport radioelements other than plutonium and americium or the potential contribution of reversible colloid attachment to transport of less sorbing elements such as neptunium and uranium. In addition, there still exists, among the cited reports, confusion about the disposition of specific radioelements in colloid modeling. For example, CRWMS M&O (2000f) lists U-234 and Np-237 as radionuclides irreversibly attached to colloids, but CRWMS M&O (2000b) says that neptunium and uranium isotopes are not included in colloid transport models. DOE agreed[2] to address this issue.

The occurrence of nuclear criticality has been screened from Total System Performance Assessment–Site Recommendation based on its low probability of occurrence within 10,000 years (CRWMS M&O, 2000i). The basis for screening criticality from the postclosure performance assessment of the Yucca Mountain repository is contained in CRWMS M&O (2000j) which references a document (CRWMS M&O, 2000i). In the CRWMS M&O (2000i), report DOE screened out the far-field criticality, both in the unsaturated and the saturated zones, based on no waste package failure before 10,000 years. When there is no waste package, there is no release of fissile material; therefore, no fissile material to accumulate before 10,000 years in either unsaturated or saturated zones. The DOE screening argument for criticality relies heavily on the argument that the probability of a waste package failing within 10,000 years in the absence of a volcanic intrusion is very small. More recent analyses

[2]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7, 2000)." Letter (December 12) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

documented in the supplemental science and performance analyses (Bechtel SAIC Company, LLC, 2001a,b) indicate that waste package failure can occur within the first 10,000 years after repository closure because of stress corrosion cracking of welds that have been improperly heat-treated. Additional concerns about the Probability of Criticality Before 10,000 Years report are discussed in Section 3.2 of this report.

DOE described a methodology to determine the probability and consequences of a nuclear criticality event within the repository system (1998b). NRC staff accepted this methodology pending closure of 28 open items (2000g). As agreed at the DOE and NRC Technical Exchange on Criticality,[3] DOE provided NRC with Revision 1 of a topical report, which should address 27 of the open items (DOE, 2000). (The final open item on burnup verification will be addressed in the preclosure criticality analysis methodology.) Concerns relevant to criticality in the unsaturated zone are chiefly related to the methodology and validation for transport and redeposition models. To address these concerns, DOE agreed to update the topical report (2000). If this new revision of the topical report (2000) is acceptable, it will provide confidence that DOE will be able to address far-field criticality in the unsaturated zone in a potential license application even if DOE chooses to perform consequence analysis for far-field criticality to support its arguments for screening such criticality from the total system performance assessment.

DOE used arguments based on low probability and/or low consequence to exclude a number of features, events, and processes from the radionuclide transport in the unsaturated zone Total System Performance Assessment–Site Recommendation abstraction. The screening arguments are outlined in the process model report (CRWMS M&O, 2000e) and the features, events, and processes in another analysis and model report (CRWMS M&O, 2000k, 2001a). In general, the geochemical description of radionuclide transport in the unsaturated zone, including features, events, and processes, requires either a stronger technical basis for exclusion and verification of assumptions or needs to be included in the performance assessment calculations. In a number of cases, the screening arguments are adequate, and the exclusion of a particular feature, event, and process is appropriate. In other cases, however, the DOE argument is incomplete, based on assumptions that are to be verified or are otherwise inadequate at this time. Also, in some cases, DOE has not identified a feature, event, or process as either included or excluded. Scenario analysis and the NRC assessment of the DOE screening arguments are provided in Section 3.2 of this report. DOE agreed[4] to address these concerns relating to the features, events, and processes. Some specific examples of NRC concerns related to features, events, and processes are provided next.

The DOE (CRWMS M&O, 2001b) states that particles larger than colloids (2.1.09.21.00) will be included and treated as colloids, but this radionuclide transport process is not identified as

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[3]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Criticality (October 23–24, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[4]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (May 15–17, 2001)." Letter (May 30) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

either included or excluded from the Total System Performance Assessment–Site Recommendation abstraction of Radionuclide Transport in the Unsaturated Zone (CRWMS M&O, 2000e,k, 2001a). In the preliminary features, events, and processes database (Swift, et al., 1999), this process was excluded based on the assumption that, although particles may be transported through fractures in the unsaturated zone, low groundwater velocities through the saturated zone would lead to particle settling, suggesting inconsistency in the screening analysis. Qualitative comparison to colloid size distributions from wells in the Yucca Mountain region was also used as part of the exclusion rationale suggesting inconsistency in the screening analysis. This process is also noted as excluded under two other model components in the features, events, and processes database (CRWMS M&O, 2001c). Because DOE includes colloid formation processes in its screening analysis, and because of the large amounts of iron particles that may be introduced in the engineered barrier subsystem, particle transport through the engineered barrier subsystem into the unsaturated zone is plausible. Exclusion of the particle transport process may be acceptable but will remain open until DOE provides a more complete technical basis and calculations to support an assumption of low consequence. DOE should also consider the possible effects of settled or trapped particles acting as sources of dissolved radionuclide.

Radionuclide solubility limits in the geosphere (2.2.08.07.00) are excluded from the Total System Performance Assessment–Site Recommendation abstraction of radionuclide transport in the unsaturated zone on the basis of low consequence (CRWMS M&O, 2000e,k, 2001a). The DOE screening argument assumes that radionuclide solubility limits in the geosphere may be different from the near-field environment but indicates that this process is conservatively ignored with respect to solubility reduction in the far field (CRWMS M&O, 2000e). Although this argument makes valid points, the possibility of either creating a secondary source or increasing solubility limits should also be considered. Solubility limits in the geosphere will be determined by interaction between the contaminant plume and the host rock. Neglecting processes that control radionuclide sorption in the geosphere has not been demonstrated to be a conservative assumption and should be constrained by calculations including sensitivity analyses, bounding calculations, and comparison with natural analog systems.

Naturally occurring gases in the geosphere (2.2.11.01.00) are excluded from the Total System Performance Assessment–Site Recommendation abstraction of radionuclide transport in the unsaturated zone on the basis of both low consequence and low probability (CRWMS M&O, 2000e,k, 2001a). Screening arguments for this process expect naturally occurring gases to escape to the atmosphere through a well-connected unsaturated zone, preventing buildup in the repository. Although carbon dioxide is mentioned, its potential effects on water chemistry are not evaluated as part of the screening argument. Near-field modeling (CRWMS M&O, 2000l) suggests that thermal effects on carbon dioxide partial pressures and aqueous carbonate concentrations are small. This minimal effect would suggest that changes in the far field may also be small and have a minimal effect on radionuclide transport in the unsaturated zone (CRWMS M&O, 2000k, Assumption 11), and the exclusion is appropriate with regard to radionuclide transport.

Changes to rock properties caused by igneous activity (1.2.04.02.00) are excluded from the Total System Performance Assessment–Site Recommendation abstraction of radionuclide transport in the unsaturated zone on the basis of low consequence (CRWMS M&O, 2000e,k, 2001a). Although several of the arguments presented (scale, duration) may be reasonable, no specific technical basis, such as comparison with a natural analog (Matyskiela, 1997), is provided in the screening argument for this process. This discussion also does not include the effect of plugging of pores with remobilized silica or the effect of intruding a low-permeability igneous feature on hydrologic flow. Probability may also be an aspect to use in the screening argument for this process, provided it is consistent with the probabilities used for the igneous disruptive scenario.

Advection and dispersion are included in the radionuclide transport in the saturated zone model abstraction (CRWMS M&O, 2000b,f, 2001b) but are not identified as either included or excluded from the Total System Performance Assessment–Site Recommendation abstraction of radionuclide transport in the unsaturated zone (CRWMS M&O, 2000e,k, 2001a). Because advection and dispersion are key components of the DOE radionuclide transport in the unsaturated zone model abstraction, these processes should be included.

DOE included the current ambient groundwater chemistry (2.2.08.01.00) and composition conditions in the Total System Performance Assessment–Site Recommendation abstraction of radionuclide transport in the unsaturated zone but excluded future changes (CRWMS M&O, 2000e,k, 2001a). The thermal effects from waste emplacement in the repository are expected to be larger than any effects caused by climate change (CRWMS M&O, 2000k, Assumptions 10 and 11). These assumptions seem to be reasonable, but they are identified as to be verified in CRWMS M&O (2000k) and need to be verified. DOE asserts that the thermal effects on chemistry are minimal, but this assertion focuses mainly on the effects of dissolution and precipitation on hydrologic properties. Predicted changes in key geochemical parameters (pH and total carbon) are large enough to have an effect on sorption coefficients. It is assumed that the K_d uncertainty ranges bound possible variations from chemistry variations (CRWMS M&O, 2001a). The discussion of total system performance assessment disposition, however, does not address the potential for covariation among radioelement K_ds and possible performance effects. Furthermore, CRWMS M&O (2000m) states that K_d values derived from experiments are not considered to be influenced by microbial and precipitation and dissolution processes. The technical basis for this exclusion is not satisfactory. Without the details on how expert judgment was used to derive the Total System Performance Assessment–Site Recommendation sorption parameters, it is not clear how the effects of changes in the ambient system chemistry are incorporated in the transport calculations. DOE agreed to provide documentation of how its K_d distributions were derived.[5] The argument that K_d uncertainty accounts for microbial and precipitation and dissolution effects needs to be reconciled with the suggestion elsewhere that these effects were not considered in deriving K_ds.

Radionuclide transport in a carrier plume in the geosphere (2.2.08.02.00) is excluded from the Total System Performance Assessment–Site Recommendation abstraction of radionuclide transport in the unsaturated zone on the basis of low consequence. The key assumption (CRWMS M&O, 2000k, Assumption 11) is that results from the near-field thermal-hydrological-

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[5]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7, 2000).” Letter (December 12) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

chemical coupled processes model (CRWMS M&O, 2000l) can be used to bound the effects of similar coupled processes on far-field flow and transport. This assumption is to be verified. Because the screening argument for this process is focused primarily on thermal effects on the chemistry of seepage water entering the emplacement drifts, it does not appear to include other potential effects (colloids, interactions with waste forms, and engineered barrier subsystem materials). This argument also ignores the aspects of retardation that suggest sorption is dominated by solution chemistry rather than rock type and that these chemical changes may be either beneficial or adverse. It seems that carrier plume chemistry should be explicitly modeled as it evolves in the geosphere. Also, the properties of a carrier plume in the engineered barrier subsystem are included in the engineered barrier subsystem process model report (CRWMS M&O, 2000k,l), suggesting that radionuclide transport in a carrier plume should be included in transport beyond the engineered barrier subsystem. The arguments presented for exclusion of this process (CRWMS M&O, 2000j) are not sufficient.

Geochemical interactions in the geosphere (dissolution, precipitation, and weathering) and effects on radionuclide transport (2.2.08.03.00) are excluded from the Total System Performance–Site Recommendation abstraction of radionuclide transport in the unsaturated zone on the basis of low consequence. The key assumption (CRWMS M&O, 2000k, Assumption 11) is that results from the near-field, thermal-hydrological-chemical coupled processes model (CRWMS M&O, 2000l) can be used to bound the effects of similar coupled processes on far-field flow and transport. This assumption is to be verified. Predicted mineralogical changes (CRWMS M&O, 2000l) in response to the thermal effects of the repository are small (only calcite precipitation and dissolution). Predicted changes in porosity and permeability are also small. Transport through fractures is conservatively modeled in the Total System Performance Assessment–Site Recommendation, assuming no retardation. The screening argument addresses only changes in seepage water chemistry. The argument does not address the possibility of reduced (or enhanced) matrix diffusion through precipitation and dissolution. As described in Revision 4.0 of CRWMS M&O (2000a), diffusion into the matrix and sorption on matrix minerals are important retardation mechanisms. The effect of small-volume changes on fracture armoring and diffusion into the matrix may be important. Also, this process is included in the radionuclide transport in the saturated zone model abstraction (CRWMS M&O, 2001b), suggesting inconsistency in the DOE flow and transport models. The current screening arguments are not sufficient and will depend in part on the verification of Assumption 11 that far-field changes to radionuclide transport in the unsaturated zone will be less than calculated near-field changes (CRWMS M&O, 2000k).

DOE included the effects of ambient condition complexation (2.2.08.06.00) in the Total System Performance Assessment–Site Recommendation abstraction of radionuclide transport in the unsaturated zone, but excluded future changes (CRWMS M&O, 2000e,k, 2001a). The effects of complexation are "… implicitly included in the radionuclide sorption coefficients, ..." but there is no clear technical basis regarding how the effects of organics or other ligands were used in establishing the K_d distributions (CRWMS M&O, 2000j). Experimental results, reported in Triay, et al. (1997), that form much of the basis for the sorption coefficient distributions address only the effects of organics on neptunium and plutonium sorption. The analysis and model report (CRWMS M&O, 2000c) does not provide any additional information on the effect of organics on other radionuclides. It is also not clear how the potential effects of hydrolysis or inorganic complexation on retardation were factored into the original K_ds. The current process models do not address the effects of complexation on transport parameters, and the exclusion of changes to complex formation does not have sufficient support. DOE agreed to provide documentation of how its K_d distributions were derived.[6]

DOE excluded microbial activity in the geosphere (2.2.09.01.00) from the Total System Performance Assessment–Site Recommendation abstraction of radionuclide transport in the unsaturated zone on the basis of low consequence (CRWMS M&O, 2000e,k, 2001a) because of the low amounts of organic materials anticipated to be emplaced or generated in the postclosure repository environment (CRWMS M&O, 2000k, Assumption 12). This assumption is identified as to-be-verified, so the technical basis supporting the exclusion is not sufficient.

Repository-induced thermal effects in the geosphere (2.2.10.01.00) are excluded from the Total System Performance Assessment–Site Recommendation abstraction of radionuclide transport in the unsaturated zone on the basis of low consequence (CRWMS M&O, 2000e,k, 2001a). The screening argument is only partially supported by near-field thermal-chemical modeling for a limited number of hydrochemical constituents and minerals (CRWMS M&O, 2000i) and is not directly related to temperature effects on radionuclide transport (e.g., effect of temperature on sorption coefficients). The technical basis for the screening is not sufficient, and future evaluation will depend, in part, on the DOE verification of Assumption 11 that far-field changes to radionuclide transport in the unsaturated zone will be less than calculated near-field changes (CRWMS M&O, 2000k).

Thermal-chemical alteration (solubility, speciation, phase changes, precipitation and dissolution) (2.2.10.06.99) is excluded from the Total System Performance Assessment–Site Recommendation abstraction of radionuclide transport in the unsaturated zone on the basis of low consequence and low probability (CRWMS M&O, 2000e,k, 2001a). Thermal effects on chemistry at the mountain scale are expected to be low on the basis of near-field coupled thermal-hydrological-chemical models that indicate the thermal effects of the repository result in only small changes in major hydrochemical constituents and limited changes in mineralogy. Results in the cited report (CRWMS M&O, 2000l) consider only a few components in hydrochemistry important to container life (e.g., pH, total carbon, and calcium) and are limited to calcite precipitation and dissolution. Although it is reasonable to assume that far-field changes are likely to be less than near-field changes, this assumption it is identified as to-be-verified (CRWMS M&O, 2000k). The technical basis is not sufficient to demonstrate low consequence, and the low-probability argument is not developed at all. The evaluation of this exclusion will depend, in part, on the verification of Assumption 11 that far-field changes to radionuclide transport in the unsaturated zone will be less than calculated near-field changes (CRWMS M&O, 2000k). DOE should provide a technical basis based on verified assumptions and/or analyses that thermal repository effects will have negligible effects on transport in the saturated zone. This argument should address the effects of thermal-chemical rock alteration and temperature effects on geochemical processes such as sorption. This analysis should be presented in the context of modeling results showing temperatures as high as 65–70 °C

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[6]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7, 2000)." Letter (December 12) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[149–158 °F] at the water table during the thermal pulse (CRWMS M&O, 2000n). If DOE is to rely on the argument that any such effects are accounted for by transport property uncertainty ranges, then DOE should provide additional documentation to explain how transport parameter distributions were derived in a manner consistent with NUREG–1563 (NRC, 1996), as agreed at the Radionuclide Transport Key Technical Issue Technical Exchange.[7]

DOE excludes thermal-chemical alteration of the Calico Hills unit (2.2.10.07.00) and the Topopah Spring basal vitrophyre from the Total System Performance Assessment–Site Recommendation abstraction of radionuclide transport in the unsaturated zone on the basis of low consequence (CRWMS M&O, 2000l). The screening argument is based on prediction of small changes in aqueous geochemistry and mineralogy in response to coupled thermal-hydrological-chemistry processes in the nearfield (CRWMS M&O, 2000l). Thermal-chemical changes in the far field, including the Calico Hills unit and Topopah Spring basal vitrophyre, are expected to be even less significant (CRWMS M&O, 2000k, Assumption 11). It is important to note that the near-field analyses (CRWMS M&O, 2000o) are performed with a focus on seepage chemistry and how it might affect container life, rather than with the purpose of considering thermal effects on radionuclide transport. The screening argument indicates that temperatures in the zeolite-bearing Calico Hills unit will not be high enough to cause significant zeolite alteration (CRWMS M&O, 2000k). Final evaluation of excluding thermal alteration effects will depend in part on the verification of Assumption 11 that far-field changes to radionuclide transport in the unsaturated zone will be less than calculated near-field changes (CRWMS M&O, 2000m). This analysis should be presented in the context of modeling results showing temperatures as high as {65–70 °C [149–158 °F]} at the water table during the thermal pulse (CRWMS M&O, 2000n). If DOE is to rely on the argument that potential thermal alteration effects are accounted for by transport property uncertainty ranges, DOE should provide additional documentation to explain how transport parameter distributions were derived in a manner consistent with NUREG–1563 (NRC, 1996), as agreed at the Radionuclide Transport Key Technical Issue Technical Exchange.[8]

In summary, system description and model integration for radionuclide transport in the unsaturated zone are not adequate. As discussed, DOE agreed[9] to address these concerns in future documents. Scenario analysis and the NRC assessment of the DOE screening arguments for features, events, and processes are provided in Section 3.2 of this report.

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[7]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7, 2000)." Letter (December 12) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[8]Ibid.

[9]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

3.3.7.4.2 Data Are Sufficient for Model Justification

Overall, the current information, along with the agreements reached between the DOE and NRC (Section 3.3.7.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess radionuclide transport in the unsaturated zone with respect to data being sufficient for model justification.

The DOE abstraction approach to radionuclide transport in the unsaturated zone requires the definition of a number of parameters to describe solute transport properties of fracture networks and rock matrix in unsaturated zone below the proposed repository. These properties include fracture aperture, fracture porosity, effective fracture spacing (more correctly, flowing interval spacing), linear groundwater velocity within the fracture, porosity of the rock matrix, sorption coefficients (K_d values), and the effective matrix diffusion coefficient. Laboratory tests include measurements of rock matrix porosity (Flint,1998) and diffusion-cell and rock-beaker experiments using tuffs from the unsaturated zone at Yucca Mountain (CRWMS M&O, 2000c). A comprehensive data set to support estimates of hydrologic properties of rock matrix in the various hydrostratigraphic units at Yucca Mountain is presented by Flint (1998); that data set is sufficient to support estimates of rock matrix porosity for the transport model.

Data to support the conceptual model of diffusive solute transfer between fracture and matrix continua are supported by laboratory and field tests. Laboratory data from diffusion-cell, rock-beaker, and fractured-core experiments are used to estimate effective matrix diffusion coefficients to model diffusive mass transport in the volcanic tuffs of Yucca Mountain. Field data to provide *in-situ* evidence for matrix diffusion in the unsaturated zone at Yucca Mountain are still preliminary or ongoing. The preliminary analysis of tracer movement in the Alcove 1 infiltration experiments shows the tracer breakthrough data are fit best by a numerical model that includes the effects of matrix diffusion.[10] Ongoing tracer tests in the Alcove 8–Niche 3 are aimed at providing additional evidence for matrix diffusion in the Topopah Springs upper lithophysal and middle nonlithophysal units. DOE agreed to complete the Alcove 8–Niche 3 test and is expected to incorporate the results, as appropriate, in the total system performance assessment abstraction.[11]

The DOE abstraction for radionuclide transport in the unsaturated zone is based on a conceptual model that assumes radionuclide sorption occurs only within rock matrix and that solutes can migrate by diffusion from flowing fractures into rock matrix, a process referred to as matrix diffusion. NRC staff are concerned that insufficient data are presently available to justify the inclusion of matrix diffusion in the abstraction of radionuclide transport in the unsaturated zone. Data from tracer studies in the Alcove 1 infiltration experiments support the matrix diffusion conceptual model. These tests, however, were not conducted in the same host-rock

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[10]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Unsaturated and Saturated Flow Under Isothermal Conditions (August 16–17, 2000)." Letter (September 8) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[11]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7, 2000)." Letter (December 12) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

formation proposed for possible construction of a repository. DOE agreed[12] to conduct tests of tracer transport between Alcove 8 (of the enhanced characterization of the repository block drift) and Niche 3 (of the exploratory studies facility) to provide sufficient data to justify or refute the inclusion of matrix diffusion processes in the proposed repository host rock.

The geochemical data used to support the flow field below the repository are not sufficient. Uncertainty with regard to the composition of fracture and pore water compositions (Yang, et al., 1996, 1998; Browning, et al., 2000) results from limited data sets and questions regarding whether DOE accounted for the effects of the extraction techniques on water chemistry. There is also some question with regard to Cl-36 results in the exploratory studies facility and the implications for fast paths. For example, the active-fracture model is not used to explain Cl-36 occurrence (Liu, et al., 1998) because of sparse spatial distribution. It is further hypothesized that the amount of water associated with the Cl-36 occurrences is a small part of the total flux through the mountain. The results of the study suggest active fractures are much more abundant than features associated with bomb-pulse Cl-36. In contrast, pneumatic monitoring evidence suggests that the fracture system is well-connected and can be viewed as a continuum. These types of uncertainty need to be resolved for the radionuclide transport in the unsaturated zone model abstraction. The DOE agreed[13] to provide the technical basis supporting its flow and transport models, including model calibration and *in-situ* field testing.

Faults can provide fast pathways for radionuclide transport in the unsaturated zone. Furthermore, the flow and transport characteristics of fault zone pathways can vary widely from those elsewhere in the tuff aquifer. The DOE transport parameters are assigned only by rock type and do not include any specific consideration of faults, unless they are treated explicitly as zones of fracture flow. It is not clear that DOE adequately accounted for the possible effects of these differences in formulating transport parameter distributions (CRWMS M&O, 2000c,m). DOE agreed[14] to provide a technical basis for the importance to performance of transport through fault zones below the repository and to provide the technical basis for the parameters and distributions if such transport is found to be important to performance.

DOE refers to the expert elicitation (CRWMS M&O, 2000c, p. 42) conducted for Total System Performance Assessment–1993 (Wilson, et al., 1994) as the original basis for the K_d distributions. Much of the text in a key document (Triay, et al., 1997) is virtually identical with the text in Wilson, et al., (1994), whose values were based on one elicitation session conducted with three experts involved in the DOE Yucca Mountain program. The methods used to arrive at the K_d probability distribution functions are described in general terms in Barnard, et al. (1992), but the specific process implemented for the K_d elicitation is not described. Many of the

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[12]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[13]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7, 2000)." Letter (December 12) to S.J. Brocoum, DOE. Washington, DC: NRC. 2000.

[14]Ibid.

methods normally used in expert elicitation (e.g., panel selection, training, mitigating bias, consensus building, incorporating dissenting opinions, aggregation of results, and documentation) are not discussed. This information is needed to understand how K_d probability distribution functions were selected, what data were used, and how the experts arrived at their conclusions. For example, Wilson, et al. (1994) notes that one of the experts believed that lead should be assigned a K_d of zero, however, a consensus value of 0–500 mL/g [0–0.05 m^3/kg], subsequently adjusted in CRWMS M&O (1998) to 100–500 mL/g [0.01–0.05 m^3/kg], was adopted. DOE agreed[15] to provide the documentation necessary to evaluate the adequacy of the DOE approach.

Subsequent changes in both K_d ranges and distribution type have been made to the Total System Performance Assessment–1993 distributions without documentation. For example, protactinium is assumed to exhibit sorption characteristics similar to neptunium (Triay, et al., 1997), but the K_d distributions are different, and the upper limits are significantly higher for protactinium 100 mL/g [0.11m^3/kg] versus 3 to 15 mL/g [0.003 to 0.015 m^3/kg] for neptunium. In addition, niobium was assigned a $K_d = 0$ in Chapter 7 of CRWMS M&O (1998), but has since been assigned high K_d values similar to americium (CRWMS M&O, 2000c). In another example, the tin K_d distribution is reportedly based on the compilation of Andersson (1988). Although there is an evaluation of tin solubility data in Andersson, there is no discussion of tin sorption. DOE agreed[16] to provide the documentation necessary to evaluate the adequacy of the DOE approach.

Despite the reference to bounding the groundwater characteristics using water from Wells J–13 and UE–25 p#1, the sorption data from the automatic technical data tracking system are limited in many instances only to experiments using J–13 water. Only uranium and plutonium have significant numbers of analyses using UE–25 p#1 water. The number of experiments at different pH values is limited: the experiments are generally controlled by CO_2 overpressuring, making it difficult to identify other effects. The support for the K_d distributions is largely empirical. Although there is discussion of chemical effects on sorption, there is no process modeling to support assertions used in selecting upper or lower bounds for K_d. Eh control is limited for much of the data. For example, in the dynamic column transport experiments, assertions are made regarding the predominance of pentavalent plutonium, without any description of how redox is controlled or how the dominant oxidation state is determined. This process is especially critical for a redox sensitive element such as plutonium. Finally, there is no apparent correlation among the different radionuclides, and the link through geochemical effects is lost. DOE agreed[17] to provide the documentation necessary to evaluate the adequacy of the DOE approach.

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[15]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7, 2000)." Letter (December 12) to S.J. Brocoum, DOE. Washington, DC: NRC. 2000.

[16]Ibid.

[17]Ibid.

Radionuclide Transport in the Unsaturated Zone

Documentation is necessary to determine how these types of geochemical uncertainties have been factored into the DOE total system performance assessment transport parameter distributions to support a licensing decision. The documentation should be adequate to allow an external reviewer to trace the origins of the judgments from initial assumptions through aggregation of results and parameter development. In particular, DOE should provide information that is sufficiently complete to allow the reviewer to evaluate the expert judgment(s), the technical information used to support the judgments, how the judgments are implemented in total system performance assessment, and why they are used instead of obtaining the needed objective information (NRC, 1996). DOE agreed to provide documentation of the technical basis for its expert elicitation[18] in accordance with NRC guidance in (1996).

The data used to support the screening criteria and transport parameters for colloid transport in the total system performance assessment are insufficient at this time. The radionuclides tracked in Total System Performance Assessment–Site Recommendation are identified in the analysis and model report inventory abstraction (CRWMS M&O, 2000g). The selection of radionuclides is appropriately based on considerations of dose, inventory, and mobility, but it is not clear in the inventory screening which radionuclides are to be modeled as colloids. DOE agreed[19] to document identification of radionuclides transported via colloids for total system performance assessment in an update to CRWMS M&O (2000m) and in CRWMS M&O (2000b,f).

The sources of data used to support estimates of fracture properties for the transport model are not readily apparent from the information provided by DOE in the unsaturated zone flow and transport process model report (CRWMS M&O, 2000e) or in supporting analysis and model reports (CRWMS M&O, 2000c,d). Additionally, the DOE model documentation does not provide a basis for relating effective fracture porosities, effective fracture apertures, or flowing interval spacings to observed fracture patterns. To address these shortcomings, DOE agreed[20] that results and analyses of ongoing seepage and transport studies in the Alcove 8–Niche 3 test will be fracture informed. In this regard, DOE should document how effective fracture porosities, effective fracture apertures, or flowing interval spacings used in the solute transport models for total system performance assessment (e.g., CRWMS M&O, 2000d, Table 3) compare with apertures and spacings typically observed *in situ*. The ability to relate unsaturated zone transport properties to observed fracture patterns will provide justification for extending results of underground tracer studies in niches and alcoves at Yucca Mountain to the area proposed for repository construction.

In summary, additional data are needed from DOE to support the inclusion of matrix diffusion and radionuclide sorption in the unsaturated zone transport model for the proposed Yucca Mountain repository. As discussed in the preceding paragraphs, DOE agreed to collect

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[18]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7, 2000)." Letter (December 12) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[19]Ibid.

[20]Ibid.

the additional data for model justification and provide it for review before a potential license application.

3.3.7.4.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with the agreements reached between the DOE and NRC (Section 3.3.7.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess radionuclide transport in the unsaturated zone with respect to data uncertainty being characterized and propagated through the model abstraction.

Uncertainty in the effective diffusion coefficient is a function of the uncertainty and variability in the molecular size of the radionuclide, temperature, heterogeneity of rock properties, and geochemical conditions along the transport pathway. The distributions of matrix diffusion values used to develop the total system performance assessment abstraction for radionuclide transport in the unsaturated zone are based on laboratory-measured diffusion coefficients of tritium for cationic radionuclide species and technetium for anionic species (CRWMS M&O, 2000c, Section 6.6.1). For both anionic and cationic species, the range of effective diffusion coefficients is sampled stochastically for each total system performance assessment realization from a beta-type distribution with a range of 0–10^{-9} m^2/s [0–1.1×10^{-8} ft^2/s]. The sampled distribution for the anionic species has a mean of 3.2×10^{-11} m^2/s [3.4×10^{-10} ft^2/s] and a standard deviation of 1×10^{-11} m^2/s [1.1×10^{-10} ft^2/s]. The distribution for the cationic species has a mean of 1.6×10^{-10} m^2/s [1.7×10^{-9} ft^2/s] and a standard deviation of 0.5×10^{-10} m^2/s [5.4×10^{-10} ft^2/s]. These distributions seem reasonably based on laboratory data and span a range that represents variability of centimeter-scale rock samples. Variability of diffusion coefficients can be expected to be much less for rock properties averaged over the scale of tens of meters in the transport model; hence, the ranges based on laboratory samples provide adequate bounds for model-scale diffusion coefficients.

Another important uncertainty is that of effective fracture aperture used in the total system performance assessment abstraction of unsaturated zone radionuclide transport. As discussed in the process model report (CRWMS M&O, 2000e), for a continuous, parallel fracture pattern, the inverse of the fracture aperture is half the area of contact between the fracture and matrix continua per unit volume of fracture pore space. Therefore, the larger the aperture, the less the diffusion (in a saturated system). For an unsaturated fracture, the relevant volume (per unit matrix area) is not the fracture pore volume itself, but the volume of water in the fracture. Apertures are sampled stochastically in the transport calculations for total system performance assessment. Aperture distributions are described using a log-normal distribution of apertures for all the model layers beneath the potential repository (values are listed in CRWMS M&O, 2000d, Table 4).

According to the supporting analysis and model report (CRWMS M&O, 2000d), fracture apertures used in the abstraction are derived from the fracture porosity and fracture-matrix connection area. It is not clear, however, what sources of data or analyses are used to support estimates of fracture porosity and fracture-matrix connection area. It is not clear how the active-fracture concept is factored into the estimates of fracture-matrix connection area. As previously mentioned, the mean fracture aperture values given in the analysis and model report seem quite large, and there is no discussion of how they relate to aperture measurements at depth. DOE should provide documentation to improve the transparency of how fracture aperture was determined. Fracture spacing also affects matrix diffusion because it sets the boundary for the depth of penetration from matrix diffusion. The sensitivity of transport to fracture spacing is low, however, owing to the relatively short transport distances through the unsaturated zone, so a constant value for each layer is used (CRWMS M&O, 2000e, Section 3.11.3.4). DOE agreed[21] to provide independent lines of evidence to support the use of the active-fracture model continuum concept in the transport model.

Although a significant amount of laboratory work and literature research is evident in the DOE process model report (CRWMS M&O, 2000a) and supporting analysis and model reports (CRWMS M&O, 2000c,m), the process used in conducting the expert elicitation (or expert judgment) for transport parameter distributions, particularly K_d values, is not described in sufficient detail. Many of the methods normally used in expert elicitation (panel selection, training, bias, consensus building, dissenting opinions, aggregation, and documentation) are not discussed. In addition, the information used by the expert panel is not described in a way that demonstrates how the strengths and weaknesses of different data sets were evaluated and considered to derive the K_d probability distribution functions. Also, subsequent changes from the initial elicitation are not documented in a transparent manner. This type of information is important to allow a reviewer to trace the process used to develop parameter distributions, from the original data and assumptions to the results and conclusions (NRC, 1996). Although the parameter distributions used may be appropriate without the underlying basis for the expert judgments, the radionuclide transport in the unsaturated zone model abstraction does not provide a sufficient treatment of data uncertainty. To support a licensing decision, documentation is necessary to determine how DOE developed the total system performance assessment transport parameter distributions and the type of information used to support the expert elicitation. DOE agreed to provide documentation of the technical basis for its expert elicitation[22] in accordance with NRC guidance in (1996).

DOE improved its capability to model colloid transport in recent total system performance assessment efforts (CRWMS M&O, 1998, 2000m), but many of the parameters (e.g., the colloid partitioning coefficient, K_c) used in the models are not supported by site characterization or laboratory data. DOE addressed this problem to some extent by using bounding analyses and sensitivity analyses, but there is insufficient radioelement-specific data to determine whether the uncertainty in colloid transport has been constrained in the radionuclide transport in

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[21]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[22]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7, 2000)." Letter (December 12) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

the unsaturated zone model abstraction. DOE agreed[23] to document the identification of radionuclides transport via colloids for Total System Performance Assessment.

The data used to support transport parameters for unsaturated zone colloid transport in the total system performance assessment are insufficient, and it is not apparent that uncertainty is reflected in parameters adopted in total system performance assessment. The four parameters that affect unsaturated zone colloid transport are the colloid size distribution, colloid K_c, colloid R_c, and colloid filtration factor; colloid matrix diffusion is neglected (CRWMS M&O, 2000f). In the unsaturated zone, R_c is conservatively set to one (i.e., there is no retardation of irreversible colloids), and the colloid filtration factor applies only to the small amount of advective flow between fracture and matrix (CRWMS M&O, 2000f). These two parameters, therefore, do not have a significant diminishing effect on unsaturated zone colloid transport. The colloid size distribution is used for calculating potentially significant colloid removal by filtration at matrix unit interfaces; it is not based on site-specific data but was chosen to be consistent with unrelated laboratory data (CRWMS M&O, 2000f). The K_c parameter, used to simulate reversible colloid attachment by lowering the radioelement K_d, is based on data for americium sorption to colloids and is applied to the K_d values for all reversibly attached radionuclides (CRWMS M&O, 2000m). Calculation of K_c also involves a term for colloid concentration in the water. The concentration adopted—0.03 mg/L [0.05 in^3/oz]—is claimed to be for conservatism, the highest observed or expected colloid concentration (CRWMS M&O, 2000f). This concentration, however, is well below the maximum values used in release models for waste form 5 mg/L [8.6 in^3/oz] and iron (hydr)oxide 1 mg/L [1.7 in^3/oz] colloids derived from the engineered barrier subsystem (CRWMS M&O, 2000h). DOE has not used any data, site-specific or not, to demonstrate that the reversible colloid attachment parameter will bound the range of possible effects of this process, nor have sensitivity analyses been employed to investigate the effects of parameter uncertainty on modeled repository performance. DOE agreed[24] to provide sensitivity analyses to test the importance of colloid transport parameters and models to performance for the unsaturated and saturated zones.

In summary, DOE used stochastic approaches to identify and constrain data uncertainty in its model abstraction on radionuclide transport in the unsaturated zone. In various cases, however, the technical basis for the probability distribution functions used to describe data uncertainty is not clear and transparent. To the extent possible, DOE needs to provide experimental and field information to constrain data uncertainty. Where it is not practical to obtain these data, DOE needs to document the expert elicitations or expert judgments used to provide uncertainty estimates in accordance with NRC guidance (1996) and its own quality assurance program. Sensitivity analyses and bounding calculations are an important means of providing a risk-informed, performance-based context for the DOE data uncertainty and

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[23]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7, 2000)." Letter (December 12) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[24]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

evaluating the need for additional data. DOE agreed[25] to provide technical support demonstrating appropriate handling of data uncertainty.

3.3.7.4.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with the agreements reached between the DOE and NRC (Section 3.3.7.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess radionuclide transport in the unsaturated zone with respect to model uncertainty being characterized and propagated through the model abstraction.

DOE evaluated how different approaches to represent matrix diffusion in the transport model could yield different transport behavior. For example, comparisons between the finite element heat and mass transfer particle-tracking approach and an alternative transport model, DCPT, were performed (CRWMS M&O, 2000o, Section 6.4.3). The two particle-tracking routines agree only if diffusion and dispersion are neglected. For the cases that include diffusion and dispersion, the median breakthrough for finite element heat and mass transfer algorithm occurs at times more than one or two orders of magnitude earlier. The difference is more pronounced for radionuclides undergoing sorption in the matrix. DOE believes these differences stem from different implementations of the diffusive mass flow between fractures and the matrix in the two codes (CRWMS M&O, 2000p, Section 7). The rather significant difference between the predictive results of the two models is troublesome. The finite element heat and mass transfer model used for total system performance assessment predicts faster breakthrough.

DOE consistently neglected radionuclide sorption in fractures and applied a linear sorption coefficient to simulate radionuclide transport through the matrix in the unsaturated zone in total system performance assessment (Wilson, et al., 1994; CRWMS M&O, 1998, 2000b,f). DOE asserts that model uncertainty is contained within the probability distribution functions defined for the retardation parameters. The potential for processes such as precipitation and colloid formation to contribute to the results from batch sorption experiments is also believed to be conservatively bounded by the K_d approach (CRWMS M&O, 2000c). The acceptability of this approach to model uncertainty will depend to a large extent on the documentation of the processes and information used in the expert judgments for sorption coefficient probability distribution functions, as discussed in the previous section. DOE agreed[26] to provide this documentation as part of its technical basis for transport parameter distributions. DOE also has *in-situ* testing planned for Alcove 8–Niche 3 and Busted Butte that is anticipated to support the characterization of model uncertainty. Laboratory column experiments will also help evaluate the uncertainty in using a linear sorption coefficient.

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[25]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7, 2000)." Letter (December 12) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[26]Ibid.

In summary, for unsaturated zone colloid transport modeling, DOE addresses model uncertainty chiefly by adopting each of two distinct attachment modes—reversible and irreversible (CRWMS M&O, 2000h). DOE has not provided sufficient evidence that its selection of colloid transport parameters bounds model uncertainties, so that the radionuclide transport in the unsaturated zone model abstraction realistically or conservatively bounds the possible effects of colloids. Although Total System Performance Assessment–Site Recommendation sensitivity analyses suggest that reversible attachment has a small effect (CRWMS M&O, 2000b), DOE needs to show, for example, that neglect of kinetic adsorption and desorption effects will not result in underestimating the effects of reversible attachment on performance. DOE agreed[27] to demonstrate adequate consideration of model uncertainty as documented in future reports.

3.3.7.4.5 Model Abstraction Output Is Supported by Objective Comparisons

Overall, the current information, along with the agreements reached between the DOE and NRC (Section 3.3.7.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess radionuclide transport in the unsaturated zone with respect to model abstraction output being supported by objective comparisons.

The residence-time transfer function method used to couple matrix diffusion to the FEHM (Zyvoloski et al., 1997) transfer particle-tracking transport model is supported by comparison to predictions from analytical solutions and other numerical models (CRWMS M&O, 2000d,q). For cases where large numbers of particles are used, predictions made using the residence-time transfer function particle-tracking approach compare well to one-dimensional analytical solutions (CRWMS M&O, 2000d, Section 6.3).

To check for proper implementation of the transport model in the total system performance assessment analyses, DOE tested the coupling between GoldSim, FEHM transfer, and other coupling components (CRWMS M&O, 2000f). DOE used FEHM to track 21 species through the unsaturated zone for a period of 1 million years, with a climate change sequence of present-day climate for the first 600 years, monsoonal climate from 600 to 2,000 years, and glacial-transition climate for times greater than 2,000 years. Median transport parameter values and a maximum of 525,000 particles were used. The results show that the finite element heat and mass transfer unsaturated zone outflow mass flux curves trace the corresponding engineered barrier subsystem release curves well. The results also provide support that the GoldSim–FEHM coupling worked as designed, and finite element heat and mass transfer tracked the transport of radionuclides in the unsaturated zone correctly (CRWMS M&O, 2000f, Figures 6-165 and 6-166).

In summary, DOE has not provided sufficient evidence, either through field tests or natural analogs, that results from laboratory sorption and transport experiments can be extended or used to bound transport over larger distances and longer times. Demonstration of scale effect

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[27]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

is possible at the Alluvial Tracer Complex if the distance between the wells is varied in the cross-hole tests and the duration of the tests is varied. If credit is to be taken for radionuclide attenuation, DOE should demonstrate that nonradioactive tracers used in field tests are appropriate homologues for radioelements. DOE expects to show that nonradioactive tracers used in field tests are appropriate homologues for radioelements, but results are not yet available. Ongoing testing at Alcove 8–Niche 3 in the Exploratory Studies Facility, Busted Butte, and large block studies at Atomic Energy of Canada Limited, laboratories in Pinawa, Manitoba, will provide transport data using a suite of tracers representative of conservative and weakly sorbing radionuclides (Vandergraaf, et al., 2000a,b). DOE considers these tests to be representative of transport of conservative radionuclides, sorbing radionuclides, and colloids. For dissolved radionuclides, DOE is using these results as a means of demonstrating the appropriateness of conceptual models rather than as a source of transport parameters for total system performance assessment. DOE agreed[28] to provide pretest predictions and results of field tests to demonstrate model abstraction is supported by objective comparisons.

3.3.7.5 Status and Path Forward

Table 3.3.7-1 provides the status of all key technical issue subissues, referenced in Section 3.3.7.2, for the Radionuclide Transport in the Unsaturated Zone Integrated Subissue. The table also provides the related DOE and NRC agreements pertaining to the Radionuclide Transport in the Unsaturated Zone Integrated Subissue. The agreements listed in the table are associated with one or all five generic acceptance criteria discussed in Section 3.3.7.4. Note the status and the detailed agreements (or path forward) pertaining to all the key technical issue subissues are provided in Table 1.1-3 and Appendix A.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (through specified testing, analyses, and the like), acceptably addresses the NRC questions so that no information beyond that provided, or agreed to, will likely be required at the time of a potential license application.

Table 3.3.7-1. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreements*
Radionuclide Transport	Subissue 1—Radionuclide Transport through Porous Rock	Closed-Pending	RT.1.01 through RT 1.05
	Subissue 2—Radionuclide Transport through Fractured Rock	Closed-Pending	RT.2.10

[28]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7, 2000)." Letter (December 12) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

Table 3.3.7-1. Related Key Technical Issue Subissues and Agreements (continued)

Key Technical Issue	Subissue	Status	Related Agreements*
Radionuclide Transport	Subissue 3—Radionuclide Transport through Fractured Rock		RT.3.01 RT.3.02 RT.3.04 through RT.3.08 RT.3.10
	Subissue 4—Nuclear Criticality in the Far Field	Closed-Pending	RT.4.01 RT.4.03
Unsaturated and Saturated Flow Under Isothermal Conditions	Subissue 4—Deep Percolation	Closed-Pending	USFIC.4.01
	Subissue 6—Matrix Diffusion	Closed-Pending	USFIC.6.01 USFIC.6.02 USFIC 6.03
Thermal Effects on Flow	Subissue 2—Is the DOE Thermohydrologic Modeling Approach Sufficient to Predict the Nature and Bounds of Thermal Effects on Flow in the Near Field?	Closed-Pending	TEF.2.12 TEF.2.13
Structural Deformation and Seismicity	Subissue 3—Fracturing and Structural Framework of the Geologic Setting	Closed-Pending	SDS.3.01 SDS.3.02
Evolution of the Near-Field Environment	Subissue 3—The Effects of Coupled Thermal-Hydrological-Chemical Processes on Chemical Environment for Radionuclide Release	Closed-Pending	ENFE.3.05
	Subissue 4—The Effects of Coupled Thermal-Hydrological-Chemical Processes on Radionuclide Transport through Engineered and Natural Barriers	Closed-Pending	None
Total System Performance Assessment and Integration	Subissue 1—System Description and Demonstration of Multiple Barriers	Closed-Pending	None
	Subissue 2—Scenario Analysis and Event Probability	Closed-Pending	TSPAI.2.01 TSPAI.2.02 TSPAI.2.03
	Subissue 3—Model Abstraction	Closed-Pending	TSPAI.3.28 TSPAI.3.29
	Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards	Closed-Pending	None

*Related DOE and NRC agreements are associated with one or all five generic acceptance criteria.

Note: Key Technical Issue Agreement GEN.1.01 pertains to multiple integrated subissues, as well as some specific issues related to this integrated subissue.

3.3.7.6 References

Andersson, K. "SKI Project-90: Chemical Data." SKI Technical Report 91: 21. Stockholm, Sweden: Swedish Nuclear Power Inspectorate. 1998.

Barnard, R.W., M.L. Wilson, H.A. Dockery, J.H. Gauthier, P.G. Kaplan, R.R. Eaton, F.W. Bingham, and T.H. Robey. "Total System Performance Assessment—1991: An Initial Total-System Performance Assessment for Yucca Mountain." SAND91–2795. Albuquerque, New Mexico: Sandia National Laboratories. 1992.

Bechtel SAIC Company, LLC. "FY01 Supplemental Science and Performance Analyses." Vol. 1: Scientific Bases and Analyses. TDR–MGR–MD–000007. Revision 00 ICN 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2001a.

Bechtel SAIC Company, LLC. "FY01 Supplemental Science and Performance Analyses." Vol. 2: Performance Analyses. TDR–MGR–PA–000001. Rev. 00. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2001b.

Browning, L., W.M. Murphy, B.W. Leslie, and W.L. Dam. "Thermodynamic Interpretations of Chemical Analyses of Unsaturated Zone Water From Yucca Mountain, Nevada." Symposium Proceedings. Vol. 608. R.W. Smith and D.W. Shoesmith, eds. Pittsburgh, Pennsylvania: Materials Research Society. pp. 237–242. 2000.

CRWMS M&O. "Total System Performance Assessment–Viability Assessment (TSPA-VA) Analyses Technical Basis Document." B00000000–01717–4301. Revision 01. Las Vegas, Nevada: TRW Environmental Safety Systems, Inc. 1998.

———. "Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations." TDR–WIS–RL–000001. Revision 04. Las Vegas, Nevada: CRWMS M&O. 2000a.

———. "Total System Performance Assessment for the Site Recommendation." TDF–WIS–PA–00001. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000b.

———. "Unsaturated Zone and Saturated Zone Transport Properties." ANL–NBS–HS–000019. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000c.

———. "Particle Tracking Model and Abstraction of Transport Processes." ANL–NBS–HS–000026. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000d.

———. "Unsaturated Zone Flow and Transport Model Process Model Report." TDR–NBS–HS–000002. Revision 00, ICN02. Las Vegas, Nevada: CRWMS M&O. 2000e.

———. "Total System Performance Assessment (TSPA) Model for Site Recommendation." MDL–WIS–PA–000002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000f.

———. "Inventory Abstraction." ANL–WIS–MD–000006. Revision 00. Las Vegas, Nevada. CRWMS M&O: 2000g.

———. "Waste Form Colloid-Associated Concentration Limits: Abstraction and Summary." ANL–WIS–MD–000012. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000h.

———. "Probability of Criticality Before 10,000 Years." CAL–EBS–NU–000014. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000i.

———. "Features, Events, and Processes—System Level and Criticality." ANL–WIS–MD–000019. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000j.

———. "Features, Events, and Processes in Unsaturated Zone Flow and Transport." ANL–NBS–MD–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000k.

———. "Drift-Scale Coupled Processes (DST and THC Seepage) Models." MDL–NBS–HS–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000l.

———. "Uncertainty Distributions for Stochastic Parameters." ANL–NBS–MD–000011. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000m.

———. "Mountain-Scale Coupled Processes (TH) Model." MDL–NBS–HS–000007. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000n.

———. "Engineered Barrier SubSystem Features, Events, and Processes/Degradation Modes Abstraction." ANL–WIS–PA–000002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000o.

———. "Engineered Barrier SubSystem Degradation, Flow, and Transport Process Model Report." TDR–EBS–MD–000006. Revision 00, ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000p.

———. "Analysis Comparing Advective-Dispersive Transport Solution to Particle Tracking." ANL–NBS–HS–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000q.

———. "Analysis of the Base-Case Particle Tracking Results of the Base-Case Flow Fields." ANL–NBS–HS–000024. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000r.

———. "Features, Events, and Processes in Unsaturated Zone Flow and Transport." ANL–NBS–MD–000001. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2001a.

———. "Features, Events, and Processes in Saturated Zone Flow and Transport." ANL–NBS–MD–000002. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2001b.

———. "Yucca Mountain FEP Database." TDR–WIS–MD–000003. Revision 00 ICN01. Las Vegas, Nevada: CRWMS M&O. 2001c.

Radionuclide Transport in the Unsaturated Zone

DOE. “Viability Assessment of a Repository at Yucca Mountain.” Overview and All Five Volumes. DOE/RW–0508. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 1998a.

———. “Disposal Criticality Analysis Methodology Topical Report.” YMP/TR–004Q. Revision 00. Las Vegas, Nevada: DOE. 1998b.

———. “Disposal Criticality Analysis Methodology Topical Report.” YMP/TR–004Q. Revision 01. Las Vegas, Nevada: DOE. 2000.

Flint, L.E. “Characterization of Hydrogeologic Units Using Matrix Properties, Yucca Mountain, Nevada.” U.S. Geological Survey Water Resources Investigations Report 97-4243. 1998.

Liu, H.H., C. Doughty, and G.S. Bodvarsson. “An Active Fracture Model for Unsaturated Flow and Transport in Fractured Rocks.” *Water Resources Research*. Vol.34. pp. 2,633–2,646. 1998.

Matyskiela, W. “Silica Redistribution and Hydrologic Changes in Heated Fractured Tuff.” *Geology*. Vol. 25. pp. 1,115–1,118. 1997.

Murphy, W.M. and R.T. Pabalan. NUREG/CR–6288. “Geochemical Investigations Related to the Yucca Mountain Environment and Potential Nuclear Waste Repository.” Washington, DC: NRC. 1994.

NRC. NUREG–1563, “Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program.” Washington, DC: NRC. November 1996.

——— “Issue Resolution Status Report, Key Technical Issue: Radionuclide Transport, Revision 2.” Washington, DC: NRC. 2000a.

———. “Issue Resolution Status Report, Key Technical Issue: Unsaturated and Saturated Flow Under Isothermal Conditions, Revision 3.” Washington, DC: NRC. 2000b.

———. “Issue Resolution Status Report, Key Technical Issue: Evolution of the Near-Field Environment, Revision 3.” Washington, DC: NRC. 2000c.

———. “Issue Resolution Status Report, Key Technical Issue: Structural Deformation and Seismicity, Revision 3.” Washington, DC: NRC. 2000d.

———. “Issue Resolution Status Report, Key Technical Issue: Thermal Effects on Flow, Revision 3.” Washington, DC: NRC. 2000e.

———. “Issue Resolution Status Report, Key Technical Issue: Total System Performance Assessment and Integration, Revision 3.” Washington, DC: NRC. 2000f.

———. “Safety Evaluation Report for Disposal Criticality Analysis Methodology Topical Report, Revision 0.” Washington, DC: NRC. 2000g.

———. NUREG–1804, "Yucca Mountain Review Plan—Draft Report for Comment." Revision 2. Washington, DC: NRC. March 2002.

Reimus, P.W., A. Adams, M.J. Haga, A. Humphrey, T. Callahan, I. Anghel, and D. Counce. "Results and Interpretation of Hydraulic and Tracer Testing in the Prow Pass Tuff at the C-Holes: Yucca Mountain Project Milestone SP32E7M4." Los Alamos, New Mexico: Los Alamos National Laboratory: 1999.

Sudicky, E.A. and E.O. Frind. "Contaminant Transport in Fractured Porous Media: Analytical Solutions for a System of Parallel Fractures." *Water Resources Research*. Vol. 18, No. 6. pp. 1,634–1,642. 1982.

Swift, P., G. Barr, R. Barnard, R. Rechard, A. Schenker, G. Freeze, and P. Burck. "Feature, Event, and Process Screening and Scenario Development for the Yucca Mountain Total System Performance Assessment." SAND98–2831C. Albuquerque, New Mexico: Sandia National Laboratories. 1999.

Triay, I.R., A. Meijer, J.L. Conca, K.S. Kung, R.S. Rundberg, and E.A. Strietelmeier. "Summary and Synthesis Report on Radionuclide Retardation for the Yucca Mountain Site Characterization Project, Milestone 3784M." LA–13262–MS. Los Alamos, New Mexico: Los Alamos National Laboratory, Chemical Science and Technology Division. 1997.

Vandergraaf, T.T., D.J. Drew, and K.V. Ticknor. "Radionuclide Migration Experiments in Non-Welded Tuff Under Unsaturated and Saturated Conditions. Bi-Monthly Report, July–August 2000." TDP–NBS–GS–000020. Pinawa, Manitoba: Atomic Energy of Canada Limited. 2000a.

———. "Radionuclide Migration Experiments in Non-Welded Tuff Under Unsaturated and Saturated Conditions. Bi-Monthly Report, September–October 2000." TDP–NBS–GS–000020. Pinawa, Manitoba: Atomic Energy of Canada Limited. 2000b.

Wilson, M.L., J.H. Gauthier, R.W. Barnard, G.E. Barr, H.A. Dockery, E. Dunn, R.R. Eaton, D.C. Guerin, N. Lu, M.J. Martinez, R. Nilson, C.A. Rautman, T.H. Robey, B. Ross, E.E. Ryder, A.R. Schenker, S.A. Shannon, L.H. Skinner, W.G. Halsey, J.D. Gansemer, L.C. Lewis, A.D. Lamont, I.R. Triay, A. Meijer, and D.E. Morris. "Total-System Performance Assessment for Yucca Mountain–SNL Second Iteration (TSPA–1993) Volume 2." SAND93–2675. Albuquerque, New Mexico: Sandia National Laboratories. 1994.

Yang I.C., G.W. Rattray, and P. Yu. "Interpretation of Chemical and Isotopic Data from Boreholes in the Unsaturated Zone at Yucca Mountain, Nevada." U.S. Geological Survey. Water Resources Investigations Report 96-4058. 1996.

Yang, I.C. P, Yu, G.W. Rattray, J.S. Ferarese, and J.N. Ryan. "Hydrochemical Investigations in Characterizing the Unsaturated Zone at Yucca Mountain, Nevada." U.S. Geological Survey. Water Resources Investigations Report 98-4132. 1998.

Zyvoloski, G.A., B.A. Robinson, Z.V. Dash, and L.L. Trease. "Summary of Models and Methods for the FEHM Application—A Finite Element Heat- and Mass-Transfer Code." LA–13307–MS. Los Alamos, New Mexico: Los Alamos National Laboratory. 1997.

3.3.8 Flow Paths in the Saturated Zone

3.3.8.1 Description of Issue

The Flow Paths in the Saturated Zone Integrated Subissue addresses features and processes that affect the saturated zone flow paths and flow velocities in the saturated zone between the area beneath the proposed repository site and the compliance boundary, and their effects on the radionuclide concentrations in the groundwater at the receptor location. The relationship of this integrated subissue to other integrated subissues are depicted in Figure 3.3.8-1. The overall organization and identification of all the integrated subissues is depicted in Figure 1.1-2. The DOE description and technical bases for abstraction of flow paths in the saturated zone are documented in CRWMS M&O (2000a) and several supporting analysis and model reports cited throughout this review. This section provides a review of the abstractions DOE developed to incorporate these features and processes in its total system performance assessment.

3.3.8.2 Relationship to Key Technical Issue Subissues

The Flow Paths in the Saturated Zone Integrated Subissue incorporates subject matter previously captured in the following key technical issue subissues:

- Unsaturated and Saturated Flow Under Isothermal Conditions: Subissue 2—Hydrologic Effects of Climate Change (NRC, 1999)

- Unsaturated and Saturated Flow Under Isothermal Conditions: Subissue 5—Saturated Zone Flow and Dilution Processes (NRC, 1999)

- Unsaturated and Saturated Flow Under Isothermal Conditions: Subissue 6—Matrix Diffusion (NRC, 1999)

- Structural Deformation and Seismicity: Subissue 1—Faulting (NRC, 2000a)

- Structural Deformation and Seismicity: Subissue 3—Fracturing (NRC, 2000a)

- Total System Performance Assessment and Integration: Subissue 1—System Description and Demonstration of Multiple Barriers (NRC, 2000b)

- Total System Performance Assessment and Integration: Subissue 2—Scenario Analysis and Event Probability (NRC, 2000b)

- Total System Performance Assessment and Integration: Subissue 3—Model Abstraction (NRC, 2000b)

- Total System Performance Assessment and Integration: Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards (NRC, 2000b)

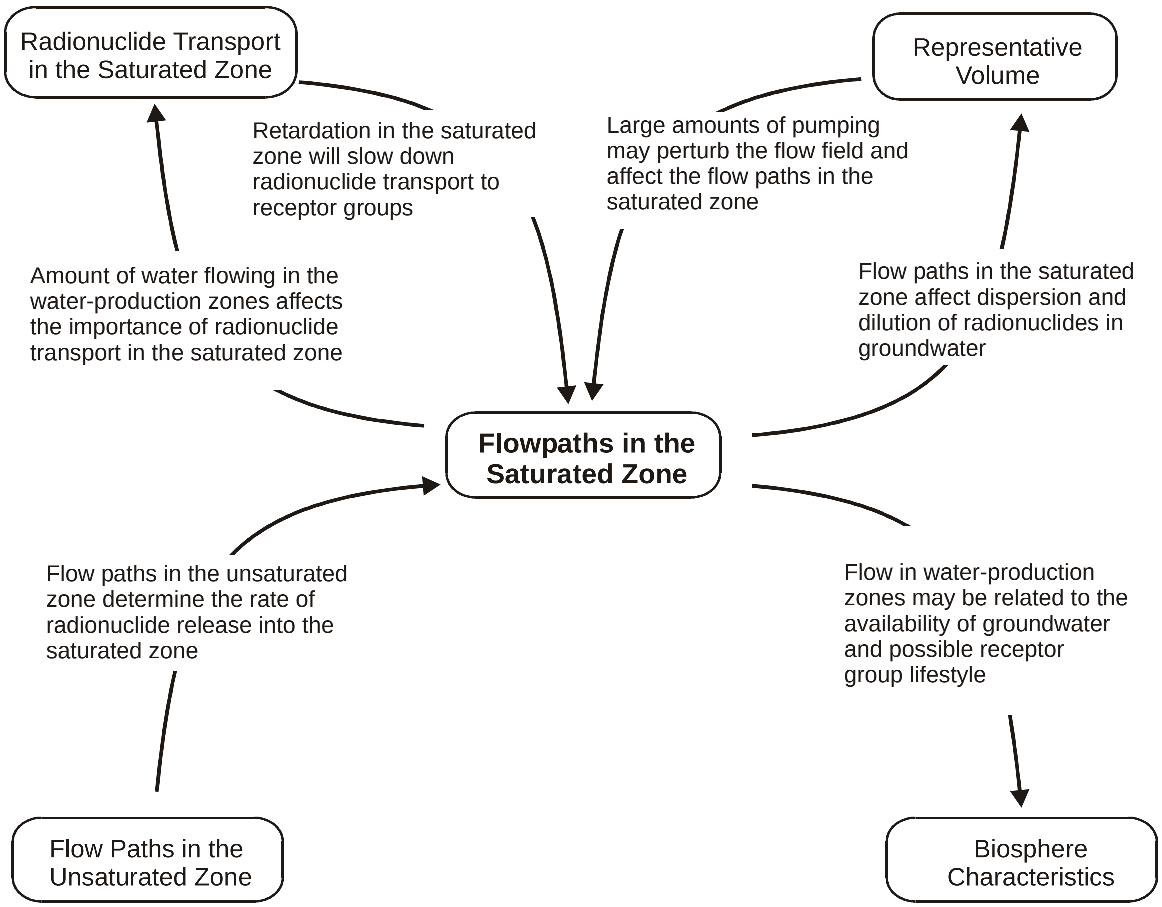

Figure 3.3.8-1. Diagram Illustrating the Relationship Between Flow Paths in the Saturated Zone and Other Integrated Subissues

- Radionuclide Transport: Subissue 1—Radionuclide Transport Through Porous Rock (NRC, 2000c)

- Radionuclide Transport: Subissue 2—Radionuclide Transport Through Fractured Rock (NRC, 2000c)

- Radionuclide Transport: Subissue 3—Radionuclide Transport Through Alluvium (NRC, 2000c)

The key technical issue subissues formed the bases for the previous versions of the issue resolution status reports and also were the bases for technical exchanges with DOE where agreements were reached on what additional information DOE needed to provide to resolve the subissue. The resolution status of this integrated subissue is based on the resolution status of each of the contributing key technical issue subissues. The subsequent sections incorporate applicable portions of these key technical issue subissues, however, no effort was made to explicitly identify each subissue.

3.3.8.3 Importance to Postclosure Performance

One aspect of risk informing the NRC review was to determine how this integrated subissue is related to the DOE repository safety strategy. DOE identified radionuclide delay through the saturated zone in CRWMS M&O (2000b) as one of eight principal factors of the current postclosure safety case for the proposed nuclear waste repository at Yucca Mountain.

DOE did not perform sensitivity analyses for individual saturated zone flow and transport parameters for the Total System Performance Assessment–Site Recommendation. Rather, analyses were performed to compare a degraded saturated zone barrier to an enhanced saturated zone barrier (CRWMS M&O, 2000c, Section 5.3.7). To evaluate degraded behavior, parameters known to increase radionuclide travel times were assigned values from the low end of their range (5th percentile), and parameters known to reduce radionuclide travel time were assigned values at the high end of their range (95th percentile). The opposite approach was used to achieve enhanced behavior. Performance estimates for 100,000 years show the difference in dose between the degraded and the enhanced cases is between one and two orders of magnitude (CRWMS M&O, 2000c, Figure 5.3-13). In this manner, it was demonstrated that the saturated zone is a potentially important barrier to radionuclide transport.

3.3.8.4 Technical Basis

NRC developed a review plan (NRC, 2002) that is consistent with the acceptance criteria and review methods found in previous issue resolution status reports. A review of DOE approaches for including flow paths in the saturated zone in total system performance assessment abstractions is provided in the following subsections. The review is organized according to the five acceptance criteria identified in Section 1.5 as follows: (i) System Description and Model Integration Are Adequate, (ii) Data Are Sufficient for Model Justification, (iii) Data Uncertainty Is Characterized and Propagated Through the Model Abstraction, (iv) Model Uncertainty Is Characterized and Propagated Through the Model Abstraction, and (v) Model Abstraction Output Is Supported by Objective Comparisons.

3.3.8.4.1 System Description and Model Integration Are Adequate

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.8.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess flow paths in the saturated zone with respect to system description and model integration.

A site-scale three-dimensional, steady-state saturated zone flow model of the Yucca Mountain region was developed to support saturated zone radionuclide transport calculations for total system performance assessment (CRWMS M&O, 2000a). The flow model domain lies within the Alkali Flat-Furnace Creek groundwater basin, which is part of the larger Death Valley regional groundwater flow system. A major assumption used to develop the site-scale model is that the Death Valley Regional Groundwater Flow Model (D’Agnese, et al., 1997) provides a reasonable representation of the groundwater flow patterns within the Alkali Flat-Furnace Creek groundwater basin and thus can be used to define boundary conditions and calibration targets for the site-scale model. Accordingly, constant-potential boundary conditions and distributed vertical recharge were derived from the regional model. Recharge from the unsaturated zone site-scale model area and from Fortymile Wash also is included in the model. NRC staff concern with the use of the Death Valley Regional Groundwater Flow Model is that the model has been updated significantly since it was last published by D'Agnese, et al. (1997). The U.S. Geological Survey has not made these updates available, and, as a result, their potential impacts on the DOE three-dimensional site-scale model have not been assessed. It was noted at a DOE and NRC technical exchange on saturated zone flow[1] that documentation of the Death Valley Regional Groundwater Flow Model is a U.S. Geological Survey product anticipated to be available in late 2001, and DOE agreed to update the saturated zone process model report, as necessary, to incorporate updates to the regional flow model.

The rectangular saturated zone site-scale flow model domain is 30 km [18.7 mi] wide by 45 km [28.0 mi] long and extends vertically from the water table to a depth 2,750 m [9,022 ft] below the water table (CRWMS M&O, 2000a). The numerical model grid is discretized horizontally into uniform 500 × 500-m [1640.4 × 1640.4-ft] grid cells producing a 60 × 90-cell horizontal grid. Vertically, the grid spacing varies from as little as 10 m [32.8 ft], for more permeable layers near the top of the model, to as large as 550 m [1,804.5 ft] at the bottom of the model, with a total of 39 layers (CRWMS M&O, 2000a, Table 3-4). Hydrologic properties assigned to grid cells are based on their spatial correspondence to one of 19 hydrogeologic units defined in the Hydrogeologic Framework Model (CRWMS M&O, 2000d, Table 6-2). The Hydrogeologic Framework Model incorporates the Geologic Framework Model (CRWMS M&O, 2000e) that was developed to support, among other issues, site-scale unsaturated zone modeling. To include the entire saturated zone flow model area, the Hydrogeologic Framework Model coverage extends well beyond the Geologic Framework Model area, and integrates additional data from borehole lithologic logs, geologic maps, geologic cross sections, topographic information, and geologic cross sections and stratigraphic surfaces developed for the Nevada Test Site (CRWMS M&O, 2000d). A concern NRC staff raised at the DOE and NRC technical exchange on saturated zone flow[2] is that discontinuities may have been introduced during extrapolation from the Geologic Framework Model domain to the Hydrogeologic Framework Model domain. DOE agreed at the technical exchange to evaluate the potential effects of such modeling-induced discontinuities and to report the results in an update to the Hydrogeologic Framework Model.

The hydrologic properties (e.g., hydraulic conductivity) of the individual hydrostratigraphic layers in the Hydrogeologic Framework Model are assumed to be homogeneous in the saturated zone flow model. Although this assumption neglects subunit heterogeneity, large-scale heterogeneity is accounted for in the model. Other contributions to the large-scale heterogeneity are considered in the model by modifying properties of grid cells corresponding

[1]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Unsaturated and Saturated Flow Under Isothermal Conditions (October 31–November 2, 2000)." Letter (November 17) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[2]Ibid.

to any of 17 hydrologic features that were added to represent faults, fault zones, areas of geochemical alteration, and an area of valley fill with large uncertainties (CRWMS M&O, 2000f, Table 6, Figure 4). Some of these features have enhanced permeability, some have reduced permeability, and some have anisotropic permeability. Each feature is said to have a significant impact on the calibration of the flow model (CRWMS M&O, 2000f, Section 6.3). Several of these hydrologic features represent faults that have enhanced permeability, are predominantly vertical, and are oriented roughly north-south. The inclusion of such features imparts to the saturated zone model an effect similar to that of a large-scale horizontal anisotropy. That is, flow can be diverted to a more southerly direction than might be inferred from the prevailing hydraulic gradient.

For the calibrated basecase saturated zone site-scale model (CRWMS M&O, 2000g), Hydrogeologic Framework Model units not associated with hydrologic features are assumed horizontally isotropic and assigned a horizontal-to-vertical anisotropy ratio of 10:1 (CRWMS M&O, 2000a). That is, horizontal permeability is independent of direction, and permeability in the vertical direction is assumed one-tenth the horizontal permeability. The reduced vertical permeability is intended to account for effects of horizontal stratification not explicitly incorporated into the model. To account for the potential effects of the preferential north-south orientation of fractures observed in boreholes penetrating saturated tuffs near Yucca Mountain (e.g., Geldon, 1996), an alternative conceptual model is considered by assigning a 5:1 anisotropy ratio to the permeability of hydrogeologic units representing fractured volcanic tuffs such that permeability in the north-south direction is five times as great as permeability in the east-west direction. This anisotropy is applied only to those tuff units located south of the repository. That is, the permeabilities of similar units north of the repository are assumed to be isotropic.

The three-dimensional saturated zone flow model was calibrated using an inverse optimization approach with the goal to minimize differences (residuals) between model estimates and observations. The observations, referred to as calibration targets, include 115 water level and head measurements (CRWMS M&O, 2000f, Table 7) and 10 side-boundary flux values derived from the Death Valley regional groundwater flow model. Weighting factors were used in the calibration to assign relative importance to each water level and head measurement. For example, weighting factors of 20 were used for water levels in wells along flow paths downstream of Yucca Mountain; factors as low as 0.05 were used for wells to the north beyond the large hydraulic gradient.

Uncertainty in both groundwater flow rates and flow directions is abstracted in total system performance assessment calculations by sampling six discrete cases of steady-state saturated zone flow fields developed using the site-scale saturated zone flow model. The first three cases are based on the isotropic conceptual model of horizontally isotropic permeability and consist of (i) the mean case (corresponding to saturated zone groundwater flux estimated from the Death Valley regional model) of the calibrated site-scale saturated zone flow model), (ii) the low-flux case (mean flux times 0.1), and (iii) the high-flux case (mean flux times 10). The other three cases represent the anisotropic conceptual model (5:1 horizontal anisotropy ratio for fractured volcanic tuffs) and include mean, low, and high fluxes. The mean-flux case for the isotropic conceptual model represents the calibrated saturated zone site-scale flow model. The other five flow model cases are not calibrated models. DOE notes, however, that only small variations exist in the simulated heads among the six flow solutions, generally less than 1 m [3.28 ft] (CRWMS M&O, 2000a, Section 3.6.3.2). In fact, the residual head differences were slightly less for the uncalibrated mean anisotropic case compared with the calibrated mean isotropic case. These differences suggest that the anisotropic conceptual model may be better suited as a basecase scenario for total system performance assessment calculations, as discussed further in Section 3.3.8.4.4.

The steady-state three-dimensional flow fields derived from the saturated zone site-scale flow model are coupled to the transport module of the saturated zone flow and transport model to generate breakthrough curves for the various radionuclides and for various stochastically sampled transport parameter sets. The breakthrough curves are then used by the abstraction model—the convolution integral method—in the total system performance assessment computer program to determine radionuclide flux to the biosphere (CRWMS M&O, 2000a, Section 3.6.3.3).

The DOE model couples climate and saturated groundwater flow through recharge. Saturated zone groundwater fluxes are expected to increase during future climates. The effects of climate change on transport of radionuclides in the saturated zone are incorporated in the total system performance assessment by scaling mass breakthrough arrival times in proportion to the estimated increase in groundwater flux during future climate conditions (CRWMS M&O, 2000a). This method treats the shift in climatic conditions as an instantaneous change from one steady-state groundwater flow condition to another. To estimate the increase in groundwater flux for future climate, simulations using inferred conditions of a past-climate state (21,000 years ago) were conducted using the Death Valley Regional Groundwater Flow Model (D’Agnese, et al., 1999). This climatic state is assumed to correspond approximately to the glacial-transition state. Results indicate a change in groundwater flux in the saturated zone near Yucca Mountain by a factor of 3.9, relative to present-day conditions. Coincidentally, the ratio of glacial-transition infiltration in the unsaturated zone model to the present-day infiltration is also about 3.9 (CRWMS M&O, 2000h). Based on this correspondence, DOE assumed that the unsaturated zone infiltration ratio provides a reasonable estimate of the flux ratio for the saturated zone. Accordingly, a value of 2.7 is used to scale the saturated zone model fluxes for the monsoon climate because it represents the ratio of predicted unsaturated zone infiltration for monsoon conditions to present-day infiltration. These inferred magnitudes of climate-induced changes in groundwater flux seem reasonable considering the range of uncertainty in groundwater flux accounted for by the mean-, low-, and high-flux cases spans two orders of magnitude (i.e., 0.1 times mean, mean, and 10 times mean). It should be noted that this approach ignores the effects of climate-induced water table rise on saturated zone flow paths. Given the scale of the saturated zone site-scale flow model, however, water table rise on the order of a few tens of meters (see Section 3.3.5) is not expected to have a significant effect on performance of the saturated zone as a natural barrier to radionuclide migration.

Several features, events, and processes are excluded from the Total System Performance Assessment–Site Recommendation abstraction of the saturated zone. These exclusions are based on screening arguments that the features, events, and processes are of low probability or of low consequence to performance estimates. The screening arguments pertaining to the abstraction of flow paths in the saturated zone are outlined in CRWMS M&O (2001). In most cases, the screening arguments are adequate and exclusion of the various features, events, and processes is appropriate. In other cases, however, DOE arguments are either incomplete, based on assumptions that are to be verified, or otherwise inadequate at this time. A list of features, events, and processes for which screening arguments proposed by DOE were not adequate or required verification, and their associated path forward (as agreed to by DOE and NRC during an August 2001 Technical Exchange Meeting on Total System Performance Assessment and Integration[3]), is provided in Section 3.2 of this report.

In summary, the technical basis for data sufficiency for model justification with respect to flow paths in the saturated zone, along with the agreements reached, will provide a sufficient basis for a satisfactory characterization of flow paths in the saturated zone and saturated zone abstraction in the total system performance assessment at the time of a potential license application.

3.3.8.4.2 Data Are Sufficient for Model Justification

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.8.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess flow paths in the saturated zone with respect to data being sufficient for model justification.

To assess the extent to which radionuclides may be delayed or immobilized in the saturated zone, it is necessary to understand the ambient flow conditions and the spatial distribution of hydrologic properties from the water table beneath the proposed repository to the compliance boundary. Ambient flow conditions are affected by the subsurface geology, areal recharge patterns, water use patterns, and interbasin or interaquifer mixing of groundwaters.

The Hydrogeologic Framework Model (CRWMS M&O, 2000e) provides the conceptual foundation for the hydrostratigraphy of the site-scale three-dimensional flow model. Available hydrogeologic data used by DOE to develop the Hydrogeologic Framework Model include the Geologic Framework Model, borehole lithologic logs, geologic maps, geologic cross sections, and topographic information. The Hydrogeologic Framework Model is generally consistent with the conceptual model developed by Luckey, et al. (1996), in which saturated zone flow from below Yucca Mountain goes through gently eastward-dipping volcanic-tuff aquifers and aquitards occasionally offset by faults, transitioning to a valley-fill alluvial aquifer some distance southeast of Yucca Mountain. NRC previously reviewed the Luckey, et al. (1996) conceptual model and found it provided an adequate basis for a groundwater flow model, with the exception of uncertainty in properties of the alluvial aquifer system and location of the tuff-alluvium interface (NRC, 1999). At a technical exchange between DOE and NRC, DOE agreed to delineate the tuff-alluvium contact based on ongoing drilling and testing.[4]

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[3]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[4]Ibid.

Data to support estimates of vertical and lateral recharge used for the saturated zone site-scale flow model are derived from three sources: (i) results of the unsaturated zone flow model (CRWMS M&O, 2000h), (ii) estimates of recharge from analysis of stream flows in Fortymile Wash (Savard, 1998), and (iii) regional fluxes predicted by the Death Valley Regional Groundwater Flow Model (D'Agnese, et al., 1997). Use of such data to develop input for the site-scale saturated zone model is reasonable. Lateral recharge estimated from the Death Valley Regional Groundwater Flow Model accounts for the vast majority of groundwater inflow to the site-scale saturated zone model. It should be noted that the Death Valley Regional Groundwater Flow Model has been significantly modified and refined since the 1997 version used for the abstraction of saturated zone flow paths. At the DOE and NRC Technical Exchange on Saturated Zone Flow issues,[5] DOE agreed to update the saturated zone flow model and process model report considering the updated Death Valley Regional Flow Model.

Water-level data collected in Yucca Mountain wells (CRWMS M&O, 2000i) indicate areas of moderate and high hydraulic gradients west and north of Yucca Mountain. East and southeast of Yucca Mountain, both the hydraulic potential and the hydraulic gradient reflected in water levels are significantly lower than those to the west and north. Water levels in wells east of the Solitario Canyon fault support the conceptual model of eastward flow of groundwater directly beneath Yucca Mountain that gradually turns to southward flow in the vicinity of Fortymile Wash.

The moderate hydraulic gradient area west of Yucca Mountain is characterized by significantly higher water table elevations in wells just west of the Solitario Canyon fault, indicating the moderate gradient likely is caused by a zone of reduced permeability in the volcanic tuffs along the Solitario Canyon fault (CRWMS M&O, 2000a, Section 3.2.2.3). It was expected that hydraulic testing of a new well, SD–6, on the crest of Yucca Mountain just east of the Solitario Canyon fault, would help to characterize further the cause of the moderate hydraulic gradient. It appears, however, that SD–6 is not sufficiently productive to produce a measurable hydraulic response in observation wells on the other side of the Solitario Canyon fault. At NRC request,[6] DOE agreed to provide the data collected during the pumping tests conducted at SD–6. Although insufficient to characterize properties of the Solitario Canyon fault, the fact that Well SD–6 produces little water supports the conceptual model of a zone of reduced permeability along or adjacent to the Solitario Canyon fault. Other wells drilled on Yucca Mountain just east of the Solitario Canyon fault also show low permeability. For example, transmissivity estimates for the volcanic tuffs in Wells USW H–3 (H–3) and USW H–5 are only 1.1 m^2/d [18.8 ft^2/d] and 36 m^2/d [387.5 ft^2/d] (e.g., Thordarson, et al., 1985; Robison and Craig, 1991). West of the Solitario Canyon fault, reported transmissivities are on the order of several hundred meters squared per day; transmissivities also increase rapidly with distances east of Solitario Canyon fault, from several hundred meters squared per day on the east flank of Yucca Mountain to a few thousand meters squared per day at the C-Holes Complex (e.g., Geldon, 1996). Thus, the moderate hydraulic gradient beneath the western portion of Yucca Mountain appears related to

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[5]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Unsaturated and Saturated Flow Under Isothermal Conditions (October 31–November 2, 2000)." Letter (November 17) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[6]Ibid.

a zone of reduced permeability associated with, or at least coincident with, the presence of the Solitario Canyon fault.

The completion of Well USW WT–24, just southeast of Well G–2, provided new insight for the large hydraulic gradient. Data show the presence of a perched-water zone. A fairly conductive fracture was eventually encountered near the base of the Calico Hills Tuff, and the water level rose over 100 m [328.08 ft], marking the location of the potentiometric surface. The water level was reported to be rising slowly as of early June 1998. In June 1998, the water level was reported at approximately 839.5 m [2,754.27 ft], implying a lateral southerly hydraulic gradient of approximately 0.059 between Well WT–24 and Wells WT–16 and USW H–1 (H–1) (shallow zone). The data verify that heads are indeed higher north of the Yucca Mountain site, and the relatively high heads in Wells G–2 and WT–6 are not entirely the result of perched water.

The cause of the large hydraulic gradient is still uncertain, but the evidence from Wells WT–24 and USW G–2 points to a simple model with a thick, low-permeability confining unit that perches water above and within it. The Calico Hills Tuff is relatively thicker to the north and occurs within the saturated zone; whereas, it is unsaturated at the Yucca Mountain site. The Calico Hills Tuff causes perched water at WT–24 and at numerous locations further south, demonstrating that vertical permeabilities are relatively low, and any fractures present are poorly conductive. Lateral permeabilities are also low, as demonstrated through testing at Well USW G–2. These low permabilities, combined with proximity to the water table, probably cause a lateral flow barrier that restricts flow and causes heads to build up to the north. The Yucca Mountain site appears to be bounded on both the north and west by zones of relatively low permeability. The high gradients across these features provide a driving force for groundwater to move laterally in the Yucca Mountain area.

The calibrated site-scale saturated zone flow model also attempts to reproduce the upward vertical hydraulic gradient observed between the Paleozoic carbonate aquifer and the overlying volcanic tuff and valley-fill (alluvial) aquifer systems. Data to support the existence of this upward gradient come from Wells UE–25 p#1 (p#1), H–1, H–3, and NC–EWDP–2DB (2DB). Hydraulic potentials in p#1 are approximately 20 m [65.6 ft] higher in the lower part of the volcanic tuffs and in the underlying carbonate aquifer system than in the upper part of the saturated volcanic tuffs. The carbonate and volcanic tuff aquifers in the vicinity of p#1 are separated by the lowermost volcanic confining unit (Luckey, et al. 1996). Well H–1 does not penetrate to the carbonate aquifer, but reaches the lower portion of the lowermost volcanic confining unit where observed potentials are about 50 m [164 ft] greater than in the overlying tuff aquifer (e.g., Graves, et al., 1997). Similarly, hydraulic potentials in Well H–3 are nearly 30 m [98.4 ft] higher in the lower interval than in the upper interval. Well 2DB, which was completed only recently, is the second well in the vicinity of Yucca Mountain to penetrate the carbonate aquifer. Data from Well 2DB are preliminary, but a Nye County representative reported at the Saturated Zone Technical Exchange[7] that hydraulic potentials are higher in the Paleozoic carbonates in Well 2DB. DOE agreed to provide an updated potentiometric map and

[7]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Unsaturated and Saturated Flow Under Isothermal Conditions (October 31–November 2, 2000)." Letter (November 17) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

supporting data for the uppermost aquifer in an update to the analysis and model report on water-level data (CRWMS M&O, 2000i), subject to receipt of data from the Nye County program. DOE also will provide an analysis of vertical hydraulic gradients in the next revision to the analysis and model report on calibration of the site-scale flow model (CRWMS M&O, 2000f).

Pumping test and water-level data from wells are used to characterize the permeabilities of the hydrogeologic units and to provide calibration targets for the saturated zone site-scale flow model. Several wells in the vicinity of Yucca Mountain can be used to reasonably estimate these characteristics for the saturated volcanic tuff aquifer system. To achieve calibration, 26 parameters representing permeabilities of hydrogeologic units and hydrologic features were adjusted to match hydraulic potentials inferred from water-level data. Adjustment of these parameters was constrained within ranges of values based on the judgment of model developers (CRWMS M&O, 2000f, Table 8). As shown in CRWMS M&O (2000a, Figure 3-22), calibrated permeability values for the saturated zone site-scale flow model did not always fall within the range of permeabilities estimated from pumping test data, but were generally within one order of magnitude. Given the limitations on the number of pumping tests that can be conducted, the uncertainties associated with interpretation of pumping test data, and the variability of the scale of the pumping tests, the calibrated permeability values compare reasonably well with those inferred from pumping test data.

Well data are sparse from about 10 km [6.2 mi] downgradient from Yucca Mountain to the compliance boundary, making it difficult to characterize the saturated zone flow and transport properties in the valley-fill deposits. Preliminary data from Nye County wells located near the compliance boundary show the water table to be occurring within the valley-fill aquifer, indicating flow paths may transition from a tuff to a valley-fill aquifer system before reaching the proposed compliance point. The exact location of the transition from the tuff to the valley-fill aquifer system and the transmissive properties of the valley-fill aquifer remain uncertain. The complexity of flow paths in the valley fill are supported by detailed examinations of the alluvial and fluvial sediments exposed in the modern entrenched channel of Fortymile Wash, which provides the best analog for features of the valley fill within the saturated flow system. Based on studies performed in the Fortymile Wash channel, Ressler (2001) concluded that the valley fill is best conceptualized by using a braided stream model consisting of eight diagnostic lithofacies defined by grain size, sedimentary features, and sedimentary geometry. Laboratory samples collected in Fortymile Wash and analyzed by Ressler (2001) indicate porosity contrasts between hydrofacies are approximately two orders of magnitude, and hydraulic conductivity contrasts ranging more than three orders of magnitude. These range can significantly impact groundwater flow and transport within the valley-fill deposits. The data gap in the valley-fill aquifer system is presently being addressed by Nye County installing several new wells. At the site of Nye County Wells NC–EWDP–19D and NC–EWDP–19P, located in Fortymile Wash approximately 2 km [1.2 mi] north of the proposed compliance boundary, DOE developed the Alluvial Testing Complex, where hydraulic and tracer testings are ongoing. The Alluvial Testing Complex, along with several new and planned Nye County wells, could possibly yield sufficient data to support parameter estimates for conceptual and numerical models of flow and transport in the valley fill. DOE agreed[8] to provide additional information to support the uncertainty distribution for flow path lengths in valley fill used in the total system performance assessment and also to provide hydrostratigraphic cross sections that include the Nye County well data.

Groundwater pore velocities are poorly constrained for flow paths from Yucca Mountain because of the difficulty in estimating effective flow porosities in the fractured tuff aquifer and the paucity of data for the valley-fill aquifer. Average linear groundwater velocities and residence times can be inferred through groundwater dating. Numerous Yucca Mountain groundwater samples have undergone C-14 dating, but it is difficult to correct for the significant amounts of dead carbon from various sources dissolved in the groundwater. A promising new approach using dissolved organic carbon may greatly improve C-14 dating of groundwater (e.g., Thomas, 1996). This technique has been applied to groundwater near Devils Hole and indicates that groundwater residence times in the carbonate aquifer feeding Devils Hole are about 2,000–3,000 years (Winograd, et al., 1997), significantly less than earlier estimates. NRC staff proposed[9] that DOE could apply this method to samples collected along saturated zone flow paths from Yucca Mountain to independently estimate the average groundwater residence times. Although DOE did not specify a method, it agreed to provide the technical basis for estimated saturated zone residence times in an update to the analysis and model report addressing geochemical and isotopic constraints on groundwater flow (CRWMS M&O, 2000j). DOE also agreed[10] to provide further justification for the range of effective porosity assumed for alluvium, considering the possible effects of contrasts in hydrologic properties of layers observed in wells. To this end, DOE will use data obtained from the Nye County Drilling Program, available geophysical data, including the Nye County aeromagnetic data, and results from the Alluvial Testing Complex. For example, borehole gravimeter data collected at Well 2DB show total porosity within the valley fill varies from approximately 20 to 30 percent.

In summary, sources of data currently analyzed lend support to the DOE conceptual and numerical models for saturated zone site-scale flow at Yucca Mountain. Considerable data collected for the tuff and alluvial aquifer systems, however, remain to be analyzed, interpreted, and published for review (e.g., the final results of the analyses of the long-term pumping and tracer tests at the C-Holes Complex). Data collection is also ongoing in the Nye County Drilling Program and the Alluvial Testing Complex hydraulic and tracer testings. In addition, the U.S. Geological Survey is assessing geochemical constraints on groundwater flow by developing detailed stratigraphic and structural models of the subsurface of Fortymile Wash using data from Nye County wells. These data are necessary to assess the DOE conceptual model for flow paths in the valley-fill aquifer system. As mentioned in the preceding discussion, DOE agreed to provide the additional data and analyses for NRC review.

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[8]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Unsaturated and Saturated Flow Under Isothermal Conditions (October 31–November 2, 2000)." Letter (November 17) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[9]Ibid.

[10]DOE and NRC. Presentation at the DOE and NRC Technical Exchange and Management Meeting on Radionuclide Transport, December 5–7, 2000. Berkeley, California. 2000.

3.3.8.4.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.8.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess flow paths in the saturated zone with respect to data uncertainty being characterized and propagated through model abstraction.

To reasonably account for the uncertainty in saturated zone flow paths and flow rates from beneath the proposed repository to the compliance boundary, the DOE radionuclide transport abstraction for total system performance assessment analyzes samples from among six different sets of saturated zone groundwater flow fields. These flow fields, derived from the saturated zone site-scale flow model, are intended to bound the uncertainties in groundwater fluxes and flow directions. The range of uncertainty considered for the groundwater flux (i.e., mean flux × 0.1 to mean flux × 10) is based on the results of the saturated zone expert elicitation (CRWMS M&O, 1998). Uncertainty in flow direction is treated by developing groundwater flow fields for both isotropic conditions and an alternative model that incorporates horizontally anisotropic conditions with a north-south orientated 5:1 permeability anisotropy ratio for fractured tuffs located south of the repository. The anisotropic cases result in more southerly flow, keeping the flow paths in fractured tuffs for greater distances, thus reducing transport distances through the valley-fill deposits. The six sets of flow fields are among several variables stochastically sampled to generate a set of 100 input files for the radionuclide transport model, which, in turn, is used to generate a library of unit breakthrough curves that subsequently can be sampled for total system performance assessment calculations. Of the 100 stochastically generated input files, the flow fields are sampled as follows: 24 realizations of the mean-flux isotropic case, 28 realizations of the mean-flux anisotropic case, 12 realizations each for the low-flux isotropic and anisotropic cases, 14 realizations of the high-flux isotropic case, and 10 realizations of the high-flux anisotropic case (CRWMS M&O, 2000g, Table 6). Thus, the mean-flux scenario is selected for approximately half the simulations in a stochastic total system performance assessment analysis; flow fields selected for the remainder of simulations are divided equally among low- and high-flux cases. Note, also, that while the anisotropic case is treated as an alternative conceptual model, the total system performance assessment abstraction uses an equal number of realizations representing isotropic and anisotropic flow fields.

Another important uncertainty in the saturated zone site-scale flow model is where saturated zone flow transitions from the volcanic tuff aquifer into the overlying valley-fill sediments along the flow path from Yucca Mountain to the compliance location. This uncertainty is important because the relatively slow flow anticipated in the porous valley fill is thought to have much greater potential for attenuation of radionuclide transport than fast, fracture-dominated flow in the tuffs. Uncertainty in the tuff-valley-fill contact is accounted for stochastically in the total system performance assessment. The tuff-valley-fill transition area is incorporated in the particle-tracking transport simulations for total system performance assessment as a trapezoidal region with a maximum north-south extent of approximately 10 km [6.21 mi], constrained by well log data. The east-west extent of this area averages approximately 5 km [3.1 mi] in width and is bounded by surface outcrops of volcanic units on the west (CRWMS M&O, 2000g, Figure 2). The northern boundary is varied throughout the full north-south extent of the uncertainty zone, from a full 10 km [6.21 mi] of flow in valley fill to a case with no flow in valley fill. The western boundary is varied only about 2 km [1.2 mi] from its most westerly position. Moving this latter boundary farther to the east results in longer transport distances in tuffs for those flow paths that fall to the west of the boundary. For each particle-tracking transport simulation, the locations of these boundaries are selected stochastically within their geometric constraints, assuming a uniform distribution. DOE assumes that the uniform distribution is the least biased, in the absence of more data, to constrain the zone geometry (CRWMS M&O, 2000j). As discussed in Section 3.3.8.4.2, DOE agreed to use data obtained from the Nye County Drilling Program, available geophysical data, and results from the Alluvial Testing Complex testing to justify the range of effective porosity in the valley fill, considering possible effects of contrasts in hydrologic properties of layers observed in wells along potential flow paths.[11]

The effective porosity of the saturated formations along flow paths from Yucca Mountain is another uncertain parameter. Flow velocities in the saturated zone are important in that they determine groundwater and radionuclide travel times from the repository to the compliance location. Effective porosity is defined as volume fraction of the saturated formation occupied by connected pore space in which groundwater movement is dominated by advection. For model layers that represent fractured tuffs, DOE refers to effective porosity as flowing interval porosity. Uncertainty in flowing interval porosity in the fractured tuffs is handled in the saturated zone transport model by assuming a log-uniform distribution of effective porosity from 10^{-5} to 10^{-1}. Based on a previous CNWRA review of effective porosity (Farrell, et al., 2000), this range provides a reasonable bound on the wide range of this highly uncertain parameter. Further, the log-uniform distribution for tuff porosity is skewed toward lower values, which is a conservative approach compared to a normal or uniform distribution. Uncertainty in effective porosity for the valley-fill aquifer is handled by sampling from a truncated normal distribution with a mean value of 0.18 and a standard deviation of 0.051 (CRWMS M&O, 2000j, Table 15). This distribution for effective porosity of the valley fill comes from a study of hydraulic characteristics of alluvium within the North American Basin and Range Province by Bedinger, et al. (1989). It is not clear, however, to what extent the study of Bedinger, et al. (1989) is applicable to the alluvial sediments along the saturated zone flow path from Yucca Mountain. Porosity data are needed from the Nye County wells completed in the valley fill along the flow path from Yucca Mountain. As mentioned in Section 3.3.8.4.2, DOE agreed to use data obtained from the Nye County Drilling Program, available geophysical data, aeromagnetic data, and results from the Alluvial Testing Complex testing to justify the range of effective porosity in the valley fill, considering possible effects of contrasts in hydrologic properties of layers observed in wells along potential flow paths.[12]

In summary, the DOE approach to incorporating data uncertainty into total system performance assessment abstractions by using a stochastic sampling approach is reasonable. More data are needed, however, to constrain the size of the alluvial uncertainty zone and to support the

[11]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Unsaturated and Saturated Flow Under Isothermal Conditions (October 31–November 2, 2000)." Letter (November 17) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[12]Ibid

range and statistical distribution used to account for uncertainty for the effective porosity of alluvium. As mentioned in the preceding discussion, DOE agreed to provide the additional data and analyses for NRC review.

3.3.8.4.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.8.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess flow paths in the saturated zone with respect to model uncertainty being characterized and propagated through model abstraction.

Model uncertainty refers to uncertainty about the validity of the conceptual models that provide the foundations for the saturated zone site-scale flow and transport models, and the numerical and mathematical approaches employed to develop total system performance assessment abstractions for consideration of flow and transport in the saturated zone. Conceptual model uncertainty requires consideration of alternative conceptual models that cannot be ruled out based on the available data. DOE agreed with this approach.[13]

A model uncertainty in the DOE approach is that changes in transport pathways could result from a potential climate-induced water table rise. This potential flow path variability is not evaluated in the current DOE site-scale saturated zone flow model. Potential water table rise on the order of a few tens of meters is inferred by the Lathrop Wells diatomite deposits that lie above a shallow water table and is likely to have only a small effect on flow patterns relative to the scale of the saturated zone flow model. Such effects might include changes in locations where the water table transitions from the tuff to the alluvial aquifer. DOE should either demonstrate that potential effects of water table rise are negligible or conservative, or incorporate water table rise into the site-scale saturated zone model. DOE researchers are presently revising the site-scale Hydrogeologic Framework Model to change the top boundary of the model from the water table to the ground surface.[14] This revision will allow consideration of water table rise with increased groundwater flux in the site-scale saturated zone flow model, if DOE determines that water table rise should be considered.

The hydraulic potentials observed in the lowermost saturated units of the volcanic tuff aquifer and in the underlying Paleozoic carbonate aquifer east of Yucca Mountain are similar in magnitude to the hydraulic potentials in the uppermost saturated units of the volcanic aquifer west of Yucca Mountain. This observation led to a proposed alternative conceptual model wherein the deep volcanic tuffs and carbonates share a good hydraulic connection with the uppermost saturated volcanic tuffs west of the Solitario Canyon fault. This conceptual model cannot be ruled out based on available data and is potentially important because the western edge of the proposed repository horizon overlies a portion of the moderate hydraulic gradient

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[13]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Unsaturated and Saturated Flow Under Isothermal Conditions (October 31–November 2, 2000)." Letter (November 17) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[14]Ibid.

area. It is, therefore, conceivable that potential releases of contaminants from the proposed repository could enter a flow system connected to the regional carbonate aquifer system, as opposed to the current conceptual model where potential contaminant releases are assumed to enter only the uppermost volcanic aquifer system. DOE agreed to consider an alternative conceptual model in which the assumed low permeability zone along the Solitario Canyon fault diminishes with depth, thereby allowing a significant hydraulic connection between the regional carbonate aquifer system below Yucca Mountain and the volcanic tuff aquifer system on the west side of the Solitario Canyon fault.[15]

DOE considers horizontal anisotropy in the permeability of fractured tuffs to be an alternative conceptual model (CRWMS M&O, 2000a). As mentioned in Section 3.3.8.4.1, however, calibration of the site-scale model was improved using a north-south orientated 5:1 horizontal anisotropy ratio. This result suggests the anisotropic model may be better suited as a basecase scenario for total system performance assessment calculations. Other support for assuming anisotropic conditions as the basecase include the preferential north-south orientation of fractures and faults in the area (e.g., Geldon, 1996; Luckey, et al., 1996), possible effects of *in-situ* stress field on conductivity of faults and fractures (Ferrill, et al., 1999), and the observed response in wells during the C-Holes Complex tests (e.g., Ferrill, et al., 1999; Winterle and La Femina, 1999). Whether anisotropic conditions are referred to as the basecase or an alternative model may be a matter of semantics because, as discussed in the preceding section, stochastic sampling of saturated zone flow fields produces an equal number of isotropic and anisotropic realizations. A concern NRC raised at the Technical Exchange on Saturated Zone Flow[16] is that the 5:1 ratio assumed for horizontal anisotropy is based on an analysis by Winterle and La Femina (1999), who noted this estimate was poorly constrained and highly uncertain. Unpublished data and analyses from long-term pumping at the C-Holes Complex could provide an improved technical basis for estimating horizontal anisotropy. At the technical exchange, DOE agreed to provide an analysis of horizontal anisotropy of permeability in the tuff aquifer based on observations in wells that responded to the long-term tests at the C-Holes Complex. Results of these analyses will be carried forward to the site-scale model, as appropriate.

Preliminary interpretations of data from the Nye County Early Warning Drilling Project wells and logs from wells in the town of Amargosa Valley indicate the presence of thick, horizontally continuous, low-permeability clay sediments in the alluvial aquifer system. The heterogeneous nature of juxtaposed clay layers and sand and gravel deposits could cause flow paths to be diverted above, below, or around such layers. Fast pathways also may exist in sand and gravel channels within clay sediments. Such juxtaposition could exert significant control on potential flow velocities and sorption capacities along flow paths within the valley-fill sediments. Presently, DOE is engaged in several data-collection efforts in the alluvial aquifer related to the Nye County Drilling Program and the Alluvial Testing Complex hydraulic and tracer testings. At

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[15]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Unsaturated and Saturated Flow Under Isothermal Conditions (October 31–November 2, 2000)." Letter (November 17) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[16]Ibid.

the previously mentioned DOE and NRC technical exchange, DOE agreed to provide additional technical bases for flow paths in alluvium, effective porosities, and transport parameters.[17]

Another alternative conceptual model is that of the potential for seismically activated geothermal perturbations of the saturated zone flow system to flood the potential location of the repository during its planned 10,000-year life. It is important to note that the State of Nevada has not provided details explaining how seismic events can trigger a significant water table rise; there are several comprehensive reviews that show water table changes from earthquakes are transitory and of a limited extent (Carrigan, et al., 1991; Gauthier, et al., 1995; Arnold and Barr, 1996). Scientists working for the State of Nevada asserted that upwelling of geothermal fluids to volcanic units above the proposed repository horizon occurred several times in the recent geologic past, including at least once within the late Quaternary (last 125,000 years). The State of Nevada scientists cite as supporting evidence abundant two-phase fluid inclusions in calcite minerals within the unsaturated zone exposed in the Exploration Studies Facility and Cross Drift, and in calcite veins found in trenches of faults (Szymanski, 1992; Archambeau and Price, 1991; Dublyansky, et al., 2001). These scientists are concerned that such geothermal activity would flood the repository with warm and chemically active fluids that would corrode the waste packages and lead to large-scale release of radionuclides to the accessible environment.

Recently, the University of Nevada, Las Vegas, concluded a 2-year study of the fluid inclusions, designed to determine the ages and temperatures of secondary mineralization at Yucca Mountain. The study focused on assemblages of two-phase fluid inclusions (gas and liquid) because such inclusions are deemed to be reasonably reliable indicators of the temperatures and pressure conditions during growth of the secondary minerals. These paleotemperatures and paleopressures are determined from measurements of the homogenization temperature, (i.e., the laboratory-heating temperature at which vapor bubbles disappear from two-phase fluid inclusions). The investigation analyzed data from the so-called two-phase fluid inclusions found in the calcite deposits beneath Yucca Mountain, and sought to determine the presence and timing of fluids with elevated temperatures that may be indicative of geothermal activity. The University of Nevada, Las Vegas, work involved collection and study of 155 samples from throughout the Exploration Studies Facility and Cross Drift. Two-phase fluid inclusion assemblages were found in secondary minerals in all areas of the Exploration Studies Facility and Cross Drift. Two-phase fluid inclusion assemblages with consistent liquid-vapor ratios were found in 78 samples. Although the study identified two-phase fluid inclusion assemblages in all minerals regardless of relative age, no two-phase fluid inclusion assemblages with consistent liquid-vapor ratios were found in the youngest (outermost) calcite in lithophysal cavities. This calcite is typically enriched in magnesium and lacks two-phase fluid inclusions (at least those two-phase fluid inclusions that have consistent liquid/vapor ratios). The magnesium-rich calcite is found in 65 percent of all samples and is dated at approximately 2 million years ago. In addition, the University of Nevada, Las Vegas, study shows that all two-phase fluid inclusion assemblages with consistent liquid-vapor ratios may be only as young as the basal portion of intermediate-age calcite, which is constrained to ages older than approximately 4 million years

[17]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Unsaturated and Saturated Flow Under Isothermal Conditions (October 31–November 2, 2000)." Letter (November 17) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

ago. Assumed maximum homogenization temperatures for the fluid inclusions were estimated to be between 40–60 °C [104–140 °F] for the Exploration Studies Facility and Cross Drift, with the intensely fractured zone having homogenization temperatures of 40–50 °C [104–140 °F], and between 60–80 °C [140–176 °F] for the north and south ramps. No fluid inclusions were found with homogenization temperatures below 35 °C [95 °F]. Note that 35 °C [95 °F] temperature was cited as a cut-off temperature, in the sense that fluids trapped at or below 35 °C [95 °F] would be sufficiently metastable to effectively inhibit formation of vapor phase bubbles.

The University of Nevada, Las Vegas, study concluded that the fluid inclusion results do not support the upwelling fluid model and, that if upwelling occurred, there should be (i) extensive mineralization throughout Yucca Mountain, (ii) greater evidence for wall rock alteration usually associated with geothermal activity, and (iii) significantly higher homogenization temperatures.

At present, there is no consensus among the project scientists on the interpretation of the University of Nevada, Las Vegas, results. State of Nevada scientists, although accepting the fluid inclusion data as valid, assert that the data do not capture the most recent thermal history of the mountain, and question the conclusion that the thermal source ceased to affect Yucca Mountain between 2 and 4 million years ago. U.S. Geological Survey staff also appear to be divided on the interpretation of the fluid inclusions, with some staff suggesting the fluid inclusion data may yield misleading results because the inclusions were trapped in the vadose zone, whereas, other staff are attempting to incorporate the temperatures into a general thermal history model, in which Yucca Mountain cooled slowly after the eruption of the Timber Mountain Caldera approximately 11 million years ago. DOE agreed to evaluate the results of ongoing fluid inclusion studies in a future update to the saturated zone flow and transport process model report.[18]

The effect of future changes in water use patterns in Amargosa Valley is not considered in the current DOE saturated zone site-scale flow model. Nye County representatives suggested that water demands will increase in the future, with potentially all available groundwater being pumped for use by the community within the next 50 years.[19] Greater rates of groundwater pumping could potentially cause saturated zone flow patterns and radionuclide travel times to differ from those DOE abstracted in the total system performance assessment. The NRC staff considers it speculative whether Nye County will significantly increase use of groundwater resources in the near future. Current high-level waste regulations for Yucca Mountain do not require DOE to evaluate all the possible scenarios that could occur during the next 10,000 years with respect to biosphere characteristics.

In summary, the DOE approach for total-system performance assessment abstraction for flow paths in the saturated zone allows consideration of a range of alternative conceptual models.

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[18]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Unsaturated and Saturated Flow Under Isothermal Conditions (October 31–November 2, 2000)." Letter (November 17) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[19] Buqo, T. *Comments made to Nuclear Waste Technical Review Board meeting, January 31, 2001*. Amargosa Valley, Nevada. 2001.

To support the alternative conceptual models that are considered or excluded, however, DOE needs to provide additional information or scenario analyses regarding horizontal anisotropy in volcanic tuffs, flow across the Solitario Canyon fault, fluid inclusion studies in the unsaturated zone, and additional technical bases for flow paths in alluvium. As mentioned in the preceding discussion, DOE agreed to provide the additional data and analyses for NRC review.

3.3.8.4.5 Model Abstraction Output Is Supported by Objective Comparisons

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.8.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess flow paths in the saturated zone with respect to model abstraction output being supported by objective comparisons.

Currently available data, including hydraulic head, groundwater chemistry, geophysics, stratigraphy, slip- and dilation-tendency analyses of faults, and analysis of horizontal anisotropy suggest that radionuclide arrival locations at the compliance boundary for groundwater flowing beneath Yucca Mountain can be constrained within the area between a point along the compliance boundary directly south of Yucca Mountain and the vicinity of Nye County Well NC–EWDP–5S, which is about 3 km [1.9 mi] northeast of the town of Amargosa Valley (Coleman, et al., 2000). The range of predicted flow paths from the basecase saturated zone site-scale flow model generally spans a large portion of this area (CRWMS M&O, 2000g, Figure 10).

DOE gained confidence in the results of its site-scale saturated zone flow model by comparing calculated to observed hydraulic heads, estimated to measured permeabilities and by comparing lateral flow rates calculated by the site-scale model to those calculated by the regional-scale flow model (CRWMS M&O, 2000a). In addition, predicted flow paths from the region of the proposed repository appear to be consistent with flow paths inferred from gradients of measured hydraulic heads and also from water chemistry data (CRWMS M&O, 2000k). There is a concern, however, that much of the data used for comparison to model results are the same data used in the calibration process. Hence, additional objective comparisons of model results to site data not used in the calibration process are needed to improve confidence in the saturated zone flow model. DOE agreed[20] to provide additional model support to be reported in a subsequent update to the Calibration of the Site-Scale Saturated Zone Flow Model analysis and model report.

In summary, the DOE approach for total-system performance assessment abstraction for flow paths in the saturated zone is consistent with and supported by available geologic, hydrologic, and geochemical data, but additional objective comparisons of model results with site data not used for model calibration are needed. As mentioned in the preceding discussion, DOE agreed to provide the additional data and analyses for NRC review.

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[20]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Unsaturated and Saturated Flow Under Isothermal Conditions (October 31–November 2, 2000)." Letter (November 17) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

3.3.8.5 Status and Path Forward

Table 3.3.8-1 provides the status of all key technical issue subissues, referenced in Section 3.3.8.2, for the Flow Paths in the Saturated Zone Integrated Subissue. The table also provides the related DOE and NRC agreements pertaining to the Flow Paths in the Saturated Zone Integrated Subissue. The agreements listed in the table are associated with one or all five generic acceptance criteria discussed in Section 3.3.8.4. Note that the status and the detailed agreements (or path forward) pertaining to all the key technical issue subissues are provided in Table 1.1-3 and Appendix A.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (through specified testing, analyses, and the like), acceptably addresses the NRC questions so that no information beyond that provided, or agreed to, will likely be required at the time of a potential license application.

Table 3.3.8-1. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreements*
Unsaturated and Saturated Flow Under Isothermal Conditions	Subissue 2—Hydrologic Effects of Climate Change	Closed	None
	Subissue 5—Saturated Zone Flow and Dilution Processes	Closed-Pending	USFIC.5.01 through USFIC.5.14
	Subissue 6—Matrix Diffusion	Closed-Pending	USFIC.6.04
Structural Deformation and Seismicity	Subissue 1—Faulting	Closed-Pending	None
	Subissue 3—Fracturing	Closed-Pending	None
Total System Performance Assessment and Integration	Subissue 1—System Description and Demonstration of Multiple Barriers	Closed-Pending	None
	Subissue 2—Scenario Analysis and Event Probability	Closed-Pending	TSPAI.2.01 TSPAI.2.02 TSPAI.2.03
	Subissue 3—Model Abstraction	Closed-Pending	None
	Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards	Closed-Pending	None

Table 3.3.8-1. Related Key Technical Issue Subissues and Agreements (continued)			
Key Technical Issue	Subissue	Status	Related Agreements*
Radionuclide Transport	Subissue 1—Radionuclide Transport through Porous Rock	Closed-Pending	RT.1.05
	Subissue 2—Radionuclide Transport though Fractured Rock	Closed-Pending	RT.2.01 through RT.2.04 RT.2.08 RT.2.09 RT.2.11
	Subissue 3—Radionuclide Transport through Alluvium	Closed-Pending	RT.3.01 RT.3.03
*Related DOE and NRC agreements are associated with one or all five generic acceptance criteria. Note: Key Technical Issue Agreement GEN.1.01 pertains to multiple integrated subissues, as well as some specific issues related to this integrated subissue.			

3.3.8.6 References

Archambeau, C.B. and N.J. Price. “An Assessment of J.S. Szymanski’s Conceptual Hydro-Tectonic Model and Its Relevance to Hydrologic and Geologic Processes at the Proposed Yucca Mountain Nuclear Waste Repository, Minority Report of the Special DOE Review Panel.” Washington, DC: DOE. 1991.

Arnold, B.W. and G.E. Barr. “Numerical Modeling for Saturated-Zone Groundwater Travel Time Analysis at Yucca Mountain.” High-Level Radioactive Waste Management: Proceedings of the Seventh Annual International Conference, Las Vegas, Nevada, April 29–May 3, 1996. La Grange Park, Illinois: American Nuclear Society. pp. 187–189. 1996.

Bedinger, M.S., W.H. Langer, and J.E. Reed. “Hydraulic Properties of Rocks in the Basin and Range Province. Studies of Geology and Hydrology in the Basin and Range Province, Southwestern United States, for Isolation of High-Level Radioactive Waste—Basis of Characterization and Evaluation.” U.S. Geological Survey Professional Paper 1370–A. 1989.

Carrigan, C.R., G.C.P. King, G.E. Barr, and N.E. Bxleil. “Potential for Water-Table Excursions Induced by Seismic Events at Yucca Mountain, Nevada.” *Geology.* Vol. 19, No. 12. pp. 1,157–1,160. 1991.

Coleman, N.M., J.R. Winterle, R.W. Fedors, A. Armstrong, D.A. Farrell, and D.L. Hughson. “Resolving Issues in Isothermal Hydrology at Yucca Mountain, Nevada—A Regulatory Perspective.” Waste Management 2000 WM2K Proceedings: HLW, LLW, Mixed Wastes and Environmental Restoration—Working Toward a Cleaner Environment, University of Arizona, Tucson, Arizona, February 27–March 2, 2000. Tucson, Arizona: WM Symposia, Inc. No page numbers. Published on CD-ROM. 2000.

CRWMS M&O. “Saturated Zone Flow and Transport Expert Elicitation Project.” SL5X4AM3. Las Vegas, Nevada: CRWMS M&O. 1998.

———. “Saturated Zone Flow and Transport Process Model Report.” TDR–NBS–HS–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000a.

———. “Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations.” TDR–WIS–RL–000001. Revision 04. Las Vegas, Nevada: CRWMS M&O. 2000b.

———. “Total System Performance Assessment for the Site Recommendation.” TDR–WIS–PA–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000c.

———. “Hydrogeologic Framework Model for the Saturated Zone Site-Scale Flow and Transport Model.” ANL–NBS–HS–000033. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000d.

———. “Geologic Framework Model (GFM3.1).” MDL–NBS–GS–000002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000e.

———. “Calibration of the Site-Scale Saturated Zone Flow Model.” MDL–NBS–HS–000011. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000f.

———. “Input and Results of the Base-Case Saturated Zone Flow and Transport Model for TSPA.” ANL–NBS–HS–000030. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000g.

———. “Unsaturated Zone Flow and Transport Model PMR.” TDR–NBS–HS–000002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000h.

———. “Water-Level Data Analysis for the Saturated Zone Site-Scale Flow and Transport Model.” ANL–NBS–HS–000034. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000i.

———. “Uncertainty Distribution for Stochastic Parameters.” ANL–NBS–MD–000011. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000j.

———. “Geochemical and Isotopic Constraints on Groundwater Flow Directions, Mixing, and Recharge at Yucca Mountain, Nevada.” ANL–NBS–HS–000021. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000k.

———. “Features, Events, and Processes in Saturated Zone Flow and Transport.” ANL–NBS–MD–000002. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2001.

D’Agnese, F.A., C.C. Faunt, A.K. Turner, and M.C. Hill. “Hydrogeologic Evaluation and Numerical Simulation of the Death Valley Regional Ground-Water Flow System, Nevada and California.” U.S. Geological Survey Water Resources Investigations Report 96-4300. 1997.

D'Agnese, F.A., G.M. O'Brien, C.C. Faunt, and C.A. San Juan. "Simulated Effects of Climate Change on the Death Valley Regional Ground-Water Flow System, Nevada and California." U.S. Geological Survey Water Resources Investigations Report 98-4041. 1999.

DOE. "Viability Assessment of a Repository at Yucca Mountain." Overview and all five volumes. DOE/RW–0508. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 1998.

Dublyansky, Y., D. Ford, and V. Reutski. "Traces of Epigenetic Hydrothermal Activity at Yucca Mountain, Nevada: Preliminary Data on the Fluid Inclusion and Stable Isotope Evidence." *Chemical Geology*. Vol. 173. pp. 125–149. 2001.

Farrell, D.A., J. Winterle, W.A. Illman, and R.W. Fedors. "Review of Porosity Distributions in the Yucca Mountain Region." San Antonio, Texas: CNWRA. 2000.

Ferrill, D.A., J. Winterle, G. Wittmeyer, D. Sims, S. Colton, A. Armstrong, and A.P. Morris. "Stressed Rock Strains Groundwater at Yucca Mountain, Nevada." *GSA Today*. Vol. 9, No. 5. pp. 1–8. 1999.

Gauthier, J.H., M.L. Wilson, D.J. Borns, and B.W. Arnold. "Impact of Seismic Activity on Long-Term Repository Performance at Yucca Mountain." Proceedings of the Topical Meeting on Methods of Seismic Hazard Evaluation, FOCUS 95, Las Vegas, Nevada, September 17–21, 1995. La Grange Park, Illinois: American Nuclear Society. pp. 159–168. 1995.

Geldon, A.L. "Results and Interpretation of Preliminary Aquifer Tests in Boreholes UE–25c#1, UE–25c#2, and UE–25c#3, Yucca Mountain, Nevada." U.S. Geological Survey Water Resources Investigations Report 94-4177. 1996.

Graves, R.P., P. Tucci, and G.M. O'Brien. "Analysis of Water Level Data in the Yucca Mountain Area, Nevada, 1985–95." U.S. Geological Survey Water Resources Investigations Report 96-4256. 1997.

Luckey, R.R., P. Tucci, C.C. Faunt, E.M. Ervin, W.C. Steinkampf, F.A. D'Agnese, and G.L. Patterson. "Status of Understanding of the Saturated-Zone Ground-Water Flow System at Yucca Mountain, Nevada, As of 1995." U.S. Geological Survey Water Resources Investigations Report 96-4077. 1996.

NRC. "Issue Resolution Status Report, Key Technical Issue: Unsaturated and Saturated Flow Under Isothermal Conditions." Revision 2. Washington, DC: NRC. 1999.

———. "Issue Resolution Status Report, Key Technical Issue: Structural Deformation and Seismicity." Revision 3. Washington, DC: NRC. 2000a.

———. "Issue Resolution Status Report, Key Technical Issue: Total System Performance Assessment and Integration." Revision 3. Washington, DC: NRC. 2000b.

———. “Issue Resolution Status Report, Key Technical Issue: Radionuclide Transport.” Revision 2. Washington, DC: NRC. 2000c.

———. NUREG–1804, "Yucca Mountain Review Plan—Draft Report for Comment." Revision 2. Washington, DC: NRC. March 2002.

Ressler, T.R. “Preliminary Characterization of the Valley-Fill Aquifer in Fortymile Wash, Southwestern Nevada.” M.Sc. thesis. The University of Texas. Austin, Texas. 2001.

Robison, J.H. and R.W. Craig. “Geohydrology of Rocks Penetrated by Test Well USW H–5, Yucca Mountain, Nye County, Nevada.” U.S. Geological Survey Water Resources Investigations Report 88-4168. 1991.

Savard, C.S. “Estimated Ground-Water Recharge from Streamflow in Fortymile Wash Near Yucca Mountain, Nevada.” U.S. Geological Survey Water Resources Investigations Report 97-73. 1998.

Szymanski, J.S. “The Origin and History of Alteration and Carbonatization of the Yucca Mountain Ignimbrites.” Two volumes. Las Vegas, Nevada: DOE. 1992.

Thomas, J.M. “Geochemical and Isotopic Interpretation of Groundwater Flow, Geochemical Processes, and Age Dating of Groundwater in the Carbonate-Rock Aquifers of the Southern Basin and Range.” Ph.D. dissertation. University of Nevada. Reno, Nevada. 1996.

Thordarson, W., F.E. Rush, and S.J. Waddell. “Geohydrology of Test Well USW H–3, Yucca Mountain, Nye County, Nevada.” U.S. Geological Survey Water Resources Investigations Report 84-4272. 1985.

Winograd, I.J., J.M. Landwehr, K.R. Ludwig, T.B. Coplen, and A.C. Riggs. “Duration and Structure of the Past Four Interglaciations.” *Quaternary Research*. Vol. 48. pp. 141–154. 1997.

Winterle, J.R. and P.C. La Femina. “Review and Analyses of Hydraulic and Tracer Testing at the C-Holes Complex Near Yucca Mountain, Nevada.” San Antonio, Texas: CNWRA. 1999.

3.3.9 Radionuclide Transport in the Saturated Zone

3.3.9.1 Description of Issue

The Radionuclide Transport in the Saturated Zone Integrated Subissue addresses features and processes that would affect movement of radionuclides in the saturated zone from the area beneath the proposed repository site at Yucca Mountain to the proposed 18-km [11-mi] compliance boundary. Figure 3.3.9-1 illustrates the relationship between the radionuclide transport in the saturated zone model abstraction and the flowpaths in the saturated zone model abstraction (see Section 3.3.8). The overall organization and identification of all the integrated subissues are depicted in Figure 1.1-2. The DOE description and technical basis for abstractions of radionuclide transport in the saturated zone are described in CRWMS M&O (2000a) and several supporting analysis and model reports. Implementation in Total System Performance Assessment–Site Recommendation is described in CRWMS M&O (2000b,c). This section provides a review of the abstractions DOE developed to incorporate these features and processes in its total system performance assessment.

3.3.9.2 Relationship to Key Technical Issue Subissues

This Radionuclide Transport in the Saturated Zone Integrated Subissue incorporates subject matter previously captured in the following key technical issue subissues:

- Radionuclide Transport: Subissue 1—Radionuclide Transport Through Porous Rock (NRC, 2000a)

- Radionuclide Transport: Subissue 2—Radionuclide Transport Through Alluvium (NRC, 2000a)

- Radionuclide Transport: Subissue 3—Radionuclide Transport Through Fractured Rock (NRC, 2000a)

- Radionuclide Transport: Subissue 4—Nuclear Criticality in the Far Field (NRC, 2000a)

- Unsaturated and Saturated Flow Under Isothermal Conditions: Subissue 5—Saturated Zone Ambient Flow Conditions and Dilution Processes (NRC, 1999a)

- Unsaturated and Saturated Flow Under Isothermal Conditions: Subissue 6—Matrix Diffusion (NRC, 1999b)

- Structural Deformation and Seismicity: Subissue 3—Fracturing and Structural Framework of the Geologic Setting (NRC, 2000b)

- Container Life and Source Term: Subissue 5—Effect of In-Package Criticality on Waste Package and Engineer Barrier System Performance (NRC, 2001a)

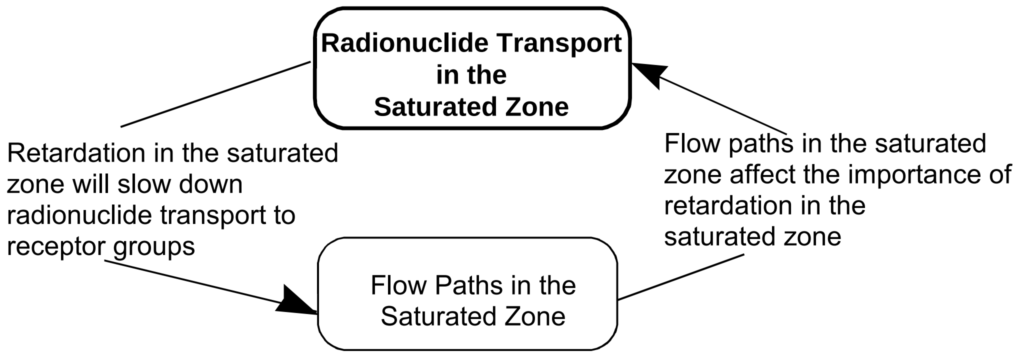

Figure 3.3.9-1. Diagram Illustrating the Relationship Between the Radionuclide Transport in the Saturated Zone and Flow Paths in the Saturated Zone Integrated Subissues

- Total System Performance Assessment and Integration: Subissue 1—System Description and Demonstration of Multiple Barriers (NRC, 2000c)

- Total System Performance Assessment and Integration: Subissue 2—Scenario Analysis and Event Probability (NRC, 2000c)

- Total System Performance Assessment and Integration: Subissue 3—Model Abstraction (NRC, 2000c)

- Total-System Performance Assessment and Integration: Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards (NRC, 2000c)

The key technical issue subissues formed the bases for the previous versions of the issue resolution status reports and also were the bases for technical exchanges with DOE where agreements were reached on what additional information DOE needed to provide to resolve the subissue. The resolution status of this integrated subissue is based on the resolution status of each of the contributing key technical issue subissues. The subsequent sections incorporate applicable portions of these key technical issue subissues, however, no effort was made to explicitly identify each subissue.

3.3.9.3 Importance to Postclosure Performance

One aspect of risk informing the NRC review was to determine how this integrated subissue is related to the DOE repository safety strategy. DOE identifies radionuclide delay through the saturated zone at Yucca Mountain as a principal factor of the current postclosure safety case (CRWMS M&O, 2000d). The degree of radionuclide sorption on mineral surfaces within the rock matrix of the tuff aquifer system and in the alluvial aquifer system is the most important process affecting the ability of the saturated zone to act as a natural barrier by attenuating and delaying potentially released radionuclides. In the current DOE abstraction approach, sorption of radionuclides in the tuff aquifer system is assumed to occur only within the relatively stagnant rock matrix, whereas flow occurs primarily in fracture networks. Matrix diffusion, a process whereby aqueous radionuclides diffuse from actively flowing pore spaces into the relatively stagnant pore space within the rock matrix, is thus another important process to be considered because the majority of saturated pore volume in the saturated tuff aquifer system comprises relatively stagnant water within rock matrix.

DOE has investigated the importance of saturated zone transport through robustness and neutralization analyses (CRWMS M&O, 2000b,d). The degraded barrier analysis, in which 5^{th} percentile values are used for parameters that positively promote delay of radionuclides in the saturated zone and 95^{th} percentile values for parameters that positively promote transport in the saturated zone, suggests modest sensitivity (CRWMS M&O, 2000d) to the saturated zone transport barrier. The similarity of the degraded and basecases is attributed to the dominance in the basecase average dose of the high-dose realizations (CRWMS M&O, 2000b). A saturated zone transport barrier neutralization analysis, in which the unsaturated zone output is fed directly to the biosphere, yields a curve nearly identical to the robustness analysis (CRWMS M&O, 2000d). It is apparent that the modeled unsaturated zone barrier in the DOE total system performance assessment is the more important barrier; this may mask the potential importance of the saturated zone barrier. Nevertheless, the importance of the saturated zone is reflected in its status as a principal factor, chiefly as a component of defense in depth (CRWMS M&O, 2000d). Furthermore, an independent NRC performance assessment sensitivity analysis concluded that retardation in the saturated zone is important, based on much higher modeled doses that result from its removal from the analysis (NRC, 1999b). In particular, neptunium retardation has been shown to have a significant dose effect (NRC, 1999b, 2001b).

3.3.9.4 Technical Basis

NRC has developed a Yucca Mountain Review Plan (NRC, 2002) that is consistent with the acceptance criteria and review methods found in previous issue resolution status reports. A review of DOE approaches for including radionuclide transport in the saturated zone in total system performance assessment abstractions is provided in the following subsections. The review is organized according to the five acceptance criteria identified in Section 1.5: (i) System Description and Model Integration Are Adequate, (ii) Data Are Sufficient for Model Justification, (iii) Data Uncertainty Is Characterized and Propagated Through the Model Abstraction, (iv) Model Uncertainty Is Characterized and Propagated Through the Model Abstraction, and (v) Model Abstraction Output Is Supported by Objective Comparisons.

NRC previously reviewed the DOE abstraction approach for radionuclide transport in the saturated zone(1999b, 2000a,b,c) after DOE publication of the viability assessment (1998a). The DOE approach for the abstraction of saturated zone radionuclide transport has changed substantially since then, moving from a one-dimensional streamtube transport model to a three-dimensional particle-tracking model.

3.3.9.4.1 System Description and Model Integration Are Adequate

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.9.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess radionuclide transport in the saturated zone with respect to system description and model integration.

The abstraction of radionuclide transport in the saturated zone for total system performance assessment analyses is developed by DOE using a site-scale, three-dimensional, single-continuum, particle-tracking transport model. Particle transport pathways are calculated based on spatially variable groundwater flux vectors (flow fields) derived from the site-scale saturated zone flow model (CRWMS M&O, 2000a). The influences of macro-scale dispersion, matrix diffusion, and adsorption of radionuclides to mineral surfaces (sorption) are incorporated through use of a residence-time transfer function that has been adapted to the finite element heat and mass transfer particle-tracking algorithm (Zyvoloski, et al., 1997). The residence-time transfer function describes a cumulative probability distribution function of particle residence times that is used to adjust travel times of particles through model cells to account for longitudinal dispersion and the delaying effects of sorption and matrix diffusion. The travel time of any given particle through a particular portion of its path is computed by generating a random number between 0 and 1 and determining the corresponding residence time from the residence-time transfer function. On average, if numerous particles travel through this portion of the model domain, the cumulative residence time distribution of particles will match the shape of the transfer function (CRWMS M&O, 2000e).

The residence-time transfer function used for the fractured tuff portion of the saturated zone flow paths is based on the Sudicky and Frind (1982) analytical solution, which takes into account advective transport in the fractures, molecular diffusion from the fracture to the porous matrix, radionuclide sorption on the fracture face, and adsorption within the matrix (CRWMS M&O, 2000f). Although the analytical solution provides for incorporating sorption on the fracture face, this option is not used in the model because of the lack of conclusive information on this process and the anticipated small impact of this option on the radionuclide transport simulations (CRWMS M&O, 2000a). It should also be noted that neglecting radionuclide sorption of fracture surfaces is a conservative approach.

The saturated zone radionuclide transport component of total system performance assessment is coupled to the unsaturated zone input and the output to the biosphere using the convolution integral method (CRWMS M&O, 2000a). In this method, a unit saturated-zone radionuclide mass breakthrough curve is computed for a step-function mass flux source; this breakthrough curve is then convoluted with the radionuclide mass flux history from the unsaturated zone to produce a radionuclide mass flux history curve that is output to the biosphere. The convolution integral method is computationally efficient and rests on the key assumptions of linear behavior and steady-state saturated zone flow conditions.

DOE relies on linear sorption isotherms and represents all noncolloidal retardation processes using the sorption coefficient (K_d) (CRWMS M&O, 2000a,g). Sorption coefficients for the radionuclides of interest are selected based on an initial informal expert elicitation conducted for Total System Performance Assessment–93, involving three experts (Wilson, et al., 1994).

Sorption parameter probability distribution functions were constrained assuming that water from the saturated volcanic tuff (Well J–13) and the Paleozoic aquifer (UE–25p#1) bound the chemistry of the groundwaters at Yucca Mountain. Total System Performance Assessment–93 only used geochemical information indirectly through expert elicitation to estimate probability distribution functions for K_d and did not explicitly incorporate geochemistry or geochemical modeling results. The approach has remained essentially unchanged since Total System Performance Assessment–93, although the specific constraints on the transport parameters have been modified, particularly for uranium, neptunium, and plutonium (Wilson, et al., 1994; CRWMS M&O, 2000g; Triay, et al., 1997). Sorption probability distribution functions are abstracted for four rock types: devitrified, vitric, and zeolitic tuff, and iron oxide. The iron oxide is intended to represent waste package corrosion products and is not used to simulate retardation by fracture-lining minerals. Radionuclide retardation is related to K_d, the sorption coefficient, by the equation

$$R_f = 1 + \frac{\rho_b}{n} K_d \qquad (3.3.9\text{-}1)$$

where R_f is the retardation factor, ρ_b is the bulk density, and n is the porosity. In fractured rocks, retardation by adsorption is assumed to occur only in the matrix, and the degree to which retardation contributes to overall repository performance depends on the nature of coupling between the matrix and fracture. In Total System Performance Assessment—Site Recommendation, K_ds are individually defined for the following radioelements assumed to not be affected by colloids: uranium, neptunium, iodine, technetium, and carbon (CRWMS M&O, 2000h). Radionuclides modeled to be reversibly attached to colloids (see later in this section) are given one of two K_d distributions in the following groups: americium, plutonium, protactinium, and thorium; and cesium and strontium (CRWMS M&O, 2000h).

The saturated zone transport simulation includes the effects of radioactive decay and ingrowth; radionuclide concentrations can increase or decrease according to decay constants. Decay of a transported radionuclide is applied directly to the convolution integral mass flux by decreasing the mass flux for the appropriate time interval using the decay equation. Decay and ingrowth during saturated zone transport for daughter radionuclides in the actinium, neptunium, thorium, and uranium decay series are treated under a one-dimensional transport model employed directly in total system performance assessment, rather than the offline three-dimensional model employed for radionuclides in general (CRWMS M&O, 2000a,i). The one-dimensional model simulates transport along pipe segments that use the average flow and transport characteristics of the corresponding flow path in the three-dimensional model. The only transport process not included in the one-dimensional model is transverse dispersion—the neglect of which is conservative (CRWMS M&O, 2000i).

Colloidal transport in the saturated zone is handled, as elsewhere in total system performance assessment, with two types of radionuclide attachment—reversible and irreversible (CRWMS M&O, 2000a,j). Colloids with irreversibly attached radionuclides are modeled as solutes, with a retardation factor applied specifically to the fractured tuff and alluvial aquifers; matrix diffusion of irreversible colloids in the saturated zone is conservatively neglected (CRWMS M&O, 2000c,j). Reversible colloidal transport is modeled using the K_c factor, representing equilibrium sorption of aqueous radionuclide onto colloids. One value for K_c, based on values representing sorption of americium to colloids at a fixed concentration in groundwater, is used for reversible attachment. Inclusion of reversible sorption to colloids lowers the effective diffusion coefficient D_e and the sorption coefficient K_d for the radionuclide (CRWMS M&O, 2000i), enhancing advective transport.

The technical basis for selecting radionuclides for saturated zone transport modeling via reversible and irreversible colloid attachment is not transparent and traceable in all cases. CRWMS M&O (2000k) identifies radionuclides for the total system performance assessment model abstraction based on contribution to dose, inventory, and mobility considerations, but does not explicitly identify those radionuclides that will be transported as colloids. DOE agreed[1] to address this issue.

The basis for screening criticality from the postclosure performance assessment of the Yucca Mountain repository is contained in a DOE analysis and model report (CRWMS M&O, 2000l) that references CRWMS M&O (2000m), DOE addressed the potential for far-field criticality in the saturated zone (CRWMS M&O, 2000l) using two features, events, and processes: far-field criticality, precipitation in organic reducing zone in or near water table (2.2.14.02.00), and far-field criticality, precipitation caused by hydrothermal upwell or redox front in the saturated zone (2.2.14.04.00). Both features, events, and processes have been excluded from the total system performance assessment. The DOE screening argument for criticality relies heavily on the argument that the probability of a waste package failing within 10,000 years in absence of a volcanic intrusion is small. When there is no waste package failure, there is no release of fissile material; therefore, no fissile material can accumulate before 10,000 years in either unsaturated or saturated zones. More recent analyses documented in the Supplemental Science and Performance Analyses (Bechtel SAIC Company, LLC, 2001a,b), however, indicate that waste package failure can occur within the first 10,000 years following repository closure due to stress corrosion cracking of welds that have been improperly heat-treated. In light of the latest results, DOE agreed to reexamine the screening argument for postclosure criticality.[2]

DOE also developed a topical report that includes the description of a methodology to determine the probability and consequences of a nuclear criticality event within the saturated zone (DOE, 1998a). NRC staff accepted this topical report pending closure of 28 open items.[3] According to an agreement made during the DOE and NRC Technical Exchange on Criticality,[4] DOE provided the NRC with Revision 1 of this topical report, which should address 27 of the

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[1]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7, 2000)." Letter (December 12) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[2]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Range of Operating Temperature (September 18–19, 2001)." Letter (October 2) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[3]Ibid.

[4]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Criticality (October 23–24, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

open items (DOE, 2000). The open items relevant to criticality in the saturated zone are concerned chiefly with methodology and validation for transport and redeposition models employed. If this new revision of the topical report is found to be acceptable, it will provide confidence that DOE will be able to address far-field criticality in the saturated zone in any potential license application even if it is unable to support its arguments for screening such criticality from the total system performance assessment.

DOE has used arguments based on low probability, low consequence, or both, to exclude various features, events, and processes from the radionuclide transport in the saturated zone Total System Performance Assessment–Site Recommendation abstraction. The screening arguments are outlined in CRWMS M&O (2001). The screening arguments for the following excluded features, events, and processes are insufficient:

- 1.2.06.00.00—Hydrothermal activity
- 1.3.07.01.00—Drought/water table decline
- 2.1.09.21.00—Suspension of particles larger than colloids
- 2.2.10.03.00—Natural geothermal effects
- 2.2.10.06.00—Thermo-chemical alteration (solubility, speciation, phase changes, precipitation/dissolution)
- 2.2.10.08.00—Thermo-chemical alteration of the saturated zone
- 2.2.10.13.00—Density-driven groundwater flow (thermal)
- 2.3.11.04.00—Groundwater discharge to surface

The comments on these features, events, and processes and possible pathways to resolution are discussed in more detail in Section 3.2 of this report. A general comment is that the analysis and model report (CRWMS M&O, 2001) neglects issues associated with transport in the alluvium. Several screening arguments focus on aspects other than those in the alluvium that might be influenced by those features, events, and processes (e.g., dissolution). DOE agreed[5] to address these concerns relating to the features, events, and processes.

In summary, system description and model integration for radionuclide transport in the saturated zone are not yet adequate, but DOE agreed to address these concerns in future documents.

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[5]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

3.3.9.4.2 Data Are Sufficient for Model Justification

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.9.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess radionuclide transport in the saturated zone with respect to data being sufficient for model justification.

Matrix Diffusion

The analytical solution of Sudicky and Frind (1982), which was used to develop the residence-time transfer function, requires estimation of several parameters, including fracture aperture, mean fracture spacing (flowing interval spacing), linear groundwater velocity within the fracture, porosity of the rock matrix, retardation factors in the rock matrix and in the fracture, and the effective matrix diffusion coefficient. Data to support estimates of these parameters and the conceptual model that matrix diffusion occurs in the saturated zone are obtained from laboratory and field testing and from the literature (CRWMS M&O, 2000h). Laboratory tests include measurements of rock matrix porosity (Flint, 1998) and diffusion-cell and rock-beaker experiments using tuffs from the saturated zone at Yucca Mountain (CRWMS M&O, 2000g). Field testing consisted of cross-hole tracer tests within the Prow Pass Tuff and Bullfrog Tuff intervals of the C-Wells Complex, which showed that tracers with differing diffusion coefficients were attenuated differently, with greater attenuation of the solute with a higher diffusion coefficient, as qualitatively predicted by the conceptual model (CRWMS M&O, 2000a; Reimus, et al., 1999).

Data obtained from flow-meter surveys of several wells in the Yucca Mountain area were used to estimate a statistical distribution of the spacing between flowing intervals in the saturated volcanic tuffs. As conceptualized for the analytical solution of Sudicky and Frind (1982), flowing interval spacing is the distance between equally spaced, parallel, planar-flowing fractures. As it applies to the volcanic tuffs beneath Yucca Mountain, this property can be thought to represent the surface area available for diffusion from flowing pore space into stagnant pore space. Smaller flowing interval spacing represents more flowing intervals and, hence, more surface area to accommodate matrix diffusion. The data to support flowing interval spacing have several limitations. For example, there was significant variability in the amount of water produced by the various features identified as flowing intervals: some features were associated with fracture zones, others were associated with permeable rock matrix—yet, the features were treated equally with regard to flowing interval spacing. Also, the flowing interval spacing parameter was used to support a conceptual model of flow through a series of parallel fractures, but there was considerable variability in the strike directions and dips of the identified flowing features. Finally, the spacing between flowing intervals was not correlated to particular hydrogeologic units of the volcanic tuffs. Thus, the estimated flowing interval spacings should be considered an effective property of the transport model that has considerable uncertainty. The combination of effective flowing interval spacing and of estimated flowing interval porosity (reviewed in Section 2.3) is used to infer the effective fracture (flowing interval) aperture used for the residence-time transfer function approach.

DOE described[6] a sensitivity analysis on the effect of flowing interval spacing on radionuclide breakthrough. As the spacing increases, the separation of the breakthrough curves decreases, such that the breakthrough curves for spacing of 50 m [160 ft] and 100 m [330 ft] are coincident. The DOE expected value of flowing interval spacing of 21 m [69 ft] results in a radionuclide breakthrough near the conservative limit of behavior.

In summary, the models DOE employed to simulate radionuclide transport in the saturated zone in performance assessment should be justified by reference to site-specific data, or data otherwise qualified for inclusion. Although there are uncertainties regarding the appropriate values for model parameters such as flowing interval spacing and diffusion coefficients, there are sufficient data to support conceptual and numerical models that include the process of matrix diffusion to predict radionuclide transport in volcanic tuffs. The DOE approach to treating uncertainty in these data is discussed in Section 3.3.9.3.3.

Sorption Coefficients

Although a significant amount of laboratory work and literature research is evident in the CRWMS M&O (2000g), the degree to which these data are used to support the total system performance assessment model abstraction of radionuclide transport in the saturated zone is not transparent and traceable. DOE refers to the expert elicitation (CRWMS M&O, 2000g, p. 42) conducted for Wilson, et al. (1994) as the original basis for K_d distributions for sorption modeling, and much of the text in a key document (Triay, et al., 1997) is virtually identical with the text in Wilson, et al. (1994). The Wilson, et al. (1994) values were based on one elicitation session conducted with three experts involved in the DOE Yucca Mountain program. The methods used to arrive at the K_d probability distribution functions are described in general terms in Barnard, et al. (1992), but the specific process implemented for the K_d elicitation is not described. Many of the methods normally used in expert elicitation (e.g., panel selection, training, mitigating bias, consensus building, incorporating dissenting opinions, aggregation of results, documentation) are not discussed. This information is needed to understand how K_d probability distribution functions were selected, what data were used, and how the experts arrived at their conclusions. For example, Wilson, et al. (1994) note that one of the experts believed that lead should be assigned a K_d of zero, but a consensus value of 0 to 500 [subsequently adjusted for Total System Performance Assessment—Viability Assessment (DOE, 1998b) to a consensus value of 100 to 500] was adopted. DOE agreed[7] to document how such differing opinions were reconciled.

Subsequent changes in both K_d ranges and distribution type have been made to the Wilson, et al. (1994) distributions without documentation. For example, protactinium is assumed to exhibit sorption characteristics similar to neptunium (Triay, et al., 1997), but the K_d distributions are different, and the upper limits are significantly higher {100 mL/g [173 in^3/oz] for protactinium versus 3 to 15 mL/g [5.2 to 25.9 in^3/oz] for neptunium}. In addition, niobium was assigned a

[6]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7, 2000)." Letter (December 12) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[7]Ibid.

K_d = 0 for Total System Performance Assessment—Viability Assessment (DOE, 1998b, Chapter 7), but has since been assigned high K_d values similar to americium (CRWMS M&O, 2000g). DOE agreed[8] to supply technical bases for the sorption coefficients used in its performance assessments.

Despite the reference to bounding the groundwater characteristics using water from Wells J–13 and UE–25 p#1, the sorption data from the automatic technical data tracking system are limited in many instances only to experiments using J–13 water. Only uranium and plutonium appear to have significant numbers of analyses using UE–25 p#1 water. The number of experiments at different pH values is limited, and they are generally controlled by CO_2 overpressuring, making it difficult to identify other effects. The support for the K_d distributions is largely empirical. Although there is discussion of chemical effects on sorption, there is no process modeling to support assertions used in picking upper or lower bounds for K_d. Control of Eh is limited for much of the data. For example, in the dynamic column transport experiments, assertions are made regarding the predominance of Pu(V), without any description of how redox is controlled or how the dominant oxidation state is determined. Such description is especially critical for a redox sensitive element such as plutonium. In addition, there is no apparent correlation among the different radionuclides, and the link through geochemical effects is lost. There is also some uncertainty about the applicability of the effective K_d approach, given the potential for processes other than sorption (precipitation, colloid formation, etc.) and reaction kinetics to complicate data interpretation (NRC, 2000a). DOE agreed[9] to analyze column test data to determine whether plutonium sorption kinetics are important to performance. If found to be important, DOE will perform sensitivity analyses to evaluate the adequacy of uranium, plutonium, and protactinium sorption coefficients.

Documentation to determine how these types of geochemical uncertainties have been factored into the DOE assembly and selection of transport parameters for total system performance assessment is necessary. DOE agreed[10] to provide documentation of the technical basis for its expert elicitation of K_d values in accordance with NRC guidance in NUREG–1563 (1996). In addition, DOE will investigate the sensitivity of repository performance to K_d for uranium, protactinium, and plutonium to determine if available data are adequate.

Colloidal Transport

The data used to support transport parameters for colloid transport in the total system performance assessment are insufficient. The two categories of colloid transport parameters employed in Total System Performance Assessment—Site Recommendation are the irreversible colloid retardation factor R_c and the K_c parameter used to simulate reversible colloid attachment by lowering the radioelement K_d (see Section 3.3.9.3.1). The irreversible colloid

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[8]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7, 2000)." Letter (December 12) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[9]Ibid.

[10]Ibid.

retardation factor distribution for volcanic units is based on the poorly constrained results of a single microsphere tracer test, whereas the value for the alluvium uses no site-specific data (CRWMS M&O, 2000j). Applicability of the microsphere results rests on assumptions regarding size distributions of microspheres versus colloids. The K_c parameter is based on data for americium sorption to colloids and is applied to the K_d values for americium, plutonium, protactinium, thorium, cesium, and strontium (CRWMS M&O, 2000h). DOE has not used any data, site-specific or not, to demonstrate that the reversible colloid attachment parameter will bound the range of possible effects of this process. DOE agreed to address these issues by providing justification that microspheres can be used as analogs for colloids (e.g., equivalent ranges in size and charge) and providing constraints on colloid transport model parameters; this justification will accompany reports on C-Wells test results.[11] DOE also agreed[12] to use sensitivity analyses to constrain colloid transport parameters used in modeling reversible and irreversible attachment and the effects of colloid transport on the radionuclide transport in the saturated zone model abstraction.

Alluvium

The alluvial flow path is a source of great uncertainty in modeling radionuclide transport in the saturated zone (NRC, 2000a). DOE models of transport through alluvium depend to a large degree on nonsite-specific data (CRWMS M&O, 2000h,g), and NRC staff raised questions regarding the adequacy of the DOE plans for future data gathering. It is desirable for drill hole samples to be representative of the full range of lithologies and water chemistries present within the expected flow paths. For example, DOE may rely on drill cuttings to obtain alluvium samples, which may adversely affect its ability to accurately measure sorption coefficients, surface area, and effective porosity—all of which may vary considerably in alluvial strata. In cuttings, sample disruption during drilling could alter these critical transport properties. The number and placement of drill holes through the alluvium needs to be adequate for characterizing spatial variations in mineralogy and lithology. The Alluvium Testing Complex alone may not assure that DOE has adequately characterized the range of alluvium properties possible over the modeled flow path; available Alluvium Testing Complex planning documents, including those cited by DOE at the radionuclide transport technical exchange[13] do not provide a level of detail necessary to resolve this question. Questions still remain about the length of the flow path along the alluvial aquifer, which has quite different transport characteristics from the tuff aquifer; this issue is discussed in Section 3.3.8. In addition, DOE has not yet obtained sufficient information on colloid transport characteristics of the alluvium. In response to these concerns, DOE agreed to demonstrate that its site characterization plans, including work on Early Warning Drilling Program Wells, the Alluvium Testing Complex, and related laboratory studies, will ensure that data on transport properties of the alluvium are sufficient to support a

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[11]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7, 2000)." Letter (December 12) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[12]Ibid.

[13]Ibid.

license application.[14] These agreements included frequent updating of field and laboratory test plans as they develop. DOE will also consider supplementing laboratory K_d studies with data from analog sites or detailed process modeling to address issues of sample integrity and representativeness.

In summary, the models DOE employed to simulate radionuclide transport in the saturated zone for performance assessment should be justified by reference to site-specific data, or data otherwise qualified for inclusion. Revisions of the process model report and the supporting analysis model reports, along with DOE agreements described previously, will provide the needed information for addressing concerns related to data sufficiency and model justification.

3.3.9.4.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Matrix Diffusion

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.9.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess radionuclide transport in the saturated zone with respect to data uncertainty being characterized and propagated through the model abstraction.

Uncertainty in data used to support the inclusion of matrix diffusion in the transport model is treated in the total system performance assessment abstraction of saturated zone radionuclide transport by stochastically sampling two parameters: the effective diffusion coefficient and the effective flowing interval spacing. Uncertainty in the effective diffusion coefficient is a function of the uncertainty and variability in the radionuclide size, temperature, heterogeneity of rock properties, and geochemical conditions along the transport pathway. DOE analyses (CRWMS M&O, 2000h, Section 6.8.4) show that most of the uncertainty in this parameter can be attributed to variability in the tortuosity of the connected pore space in the rock matrix. Based on its analyses, DOE estimated a range of possible values for effective diffusion coefficients in volcanic tuffs from 10^{-9} to 10^{-6} cm^2/s [10^{-10} to 10^{-7} in^2/s]. To ensure the effective diffusion coefficient is not overestimated, the upper bound of this range is set to below the smallest observed molecular diffusion coefficient. A log-uniform distribution is assumed for this range because it is considered unbiased with respect to the order of magnitude of the sampled parameter value and skewed toward lower values. This approach reasonably encompasses the uncertainty of this parameter.

Another important uncertainty is that of flowing interval spacing. Smaller values for effective flowing interval spacing would result in predictions of more rapid matrix diffusion. Analyses were performed to estimate a lognormally distributed range of flowing interval spacing with a mean $\log_{10}$ value of 1.29 and a standard deviation of 0.43 (CRWMS M&O, 2000p). This

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[14]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7, 2000).” Letter (December 12) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

estimate results in a range of approximately 2–200 m [7–700 ft] with a median flowing interval spacing of approximately 20 m [70 ft]. This wide range of values reasonably encompasses the uncertainty of flowing interval spacing and, given the highly fractured nature of the volcanic tuffs beneath Yucca Mountain, does not seem overly optimistic. It should be noted that the effective flowing interval spacing is used only as a transport parameter that affects the rate of matrix diffusion; it does not affect modeled groundwater fluxes or flow velocities.

Sorption Coefficients

Although a significant amount of laboratory work and literature research is evident in the DOE process model report (CRWMS M&O, 2000a) and supporting analysis and model reports (CRWMS M&O, 2000h,g), the process used in conducting the expert elicitation (or expert judgment) for transport parameter distributions, particularly K_d values, is not described in sufficient detail. Many of the methods normally used in expert elicitation (e.g., panel selection, training, bias, consensus building, dissenting opinions, aggregation, and documentation) are not discussed. In addition, the information used by the expert panel is not described in a way that demonstrates how the strengths and weaknesses of different data sets were evaluated and considered to derive the K_d probability distribution functions. Also, subsequent changes from the initial elicitation are not documented in a transparent manner. This type of information is important to allow a reviewer to trace the process used to develop parameter distributions from the original data and assumptions to the results and conclusions (NRC, 1996). Although the parameter distributions used may be appropriate, without the underlying basis for the expert judgments, the radionuclide transport in the saturated zone model abstraction does not provide a sufficient treatment of data uncertainty. DOE agreed[15] to provide the underlying basis for the expert judgments concerning sorption coefficient distributions.

In discussions of geochemical effects on saturated zone transport outlined in CRWMS M&O (2001), DOE states that the specific effects are included because uncertainty distributions of sorption coefficients are broad enough to encompass them. In each case, staff conclude that DOE has not provided sufficient technical basis that the uncertainty distributions account for the effects. Specific comments on the included features, events, and processes follow.

2.2.08.01.00—Groundwater Chemistry/Composition in Unsaturated and Saturated Zone: This feature, event, and process is included for the saturated zone on the basis that K_d uncertainty ranges bound possible variations because of chemistry variations (CRWMS M&O, 2001). The discussion of total system performance assessment disposition, however, does not address the potential for correlation among radioelement K_ds and possible performance effects. Furthermore, CRWMS M&O (2000g) states that K_d values derived from experiments are not considered to be influenced by microbial and precipitation/dissolution processes—the effects of which are asserted to be included.

[15]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7, 2000)." Letter (December 12) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

2.2.08.02.00—Radionuclide Transport in a Carrier Plume: This feature, event, and process is included in the saturated zone, based on the assertion that no credit is taken for chemical changes within the plume that would decrease the transport rate (CRWMS M&O, 2001, p. 56). However, the feature, event, and process discussion does not state how potentially adverse plume effects are accounted for; it is apparent that DOE is relying on K_d distributions. This argument appears to ignore the aspects of retardation that suggest sorption is dominated by solution chemistry rather than rock type. Because DOE does not explicitly model evolving water chemistry in the migrating carrier plume, including transport effects, DOE should provide a technical basis that states that ignoring this process is conservative or has negligible consequences.

2.2.08.03.00—Geochemical Interactions in the Geosphere: This feature, event, and process, which addresses processes such as dissolution and precipitation, is included (CRWMS M&O, 2001). There is an inconsistency in that, while DOE claims its K_d uncertainty distributions account for variations from possible interactions along the transport path, it is not clear that these processes were considered in deriving the distributions (CRWMS M&O, 2000g).

2.2.08.06.00—Complexation in the Geosphere: This feature, event, and process is stated to be included because the effects of complexation agents in the existing groundwater system are included implicitly in the distribution for the K_d value for each element (CRWMS M&O, 2001, p. 59). Parameter distributions and current DOE process models do not appear to address adequately the effects of organic complexation on transport parameters (CRWMS M&O, 2000g).

2.2.09.01.00—Microbial Activity in Geosphere: This feature, event, and process is said to be included (CRWMS M&O, 2001) based on the argument that K_d uncertainty ranges account for effects of microbial activity. The analysis and model report (CRWMS M&O, 2000g), however, states that K_d values derived from experiments are not considered to be influenced by microbial processes.

The issue common to these five included features, events, and processes is the same as that addressed in the preceding paragraphs—DOE has not adequately demonstrated that uncertainty distributions include all the possible variations in K_d in the saturated zone below Yucca Mountain. DOE can address these issues within the bounds of the existing agreement (see following paragraph) on expert judgment and transport parameter distributions.[16] Resolution of two open issues—on excluded FEP 1.2.06.00.00 and FEP 2.2.10.06.00—to be discussed in Section 3.2.1 of this report, could be addressed in the same way.

Documentation is necessary to determine how DOE developed the total system performance assessment transport parameter distributions and the type of information used to support the

[16] Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7, 2000)." Letter (December 12) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

expert elicitation. DOE agreed to provide documentation of the technical basis for its expert elicitation[17] in accordance with NRC guidance in NUREG–1563 (1996).

In summary, DOE needs to provide experimental and field information to constrain data uncertainty for all transport parameters. Where it is not practical to obtain these data, DOE needs to document the expert elicitations used to provide uncertainty estimates in accordance with NRC guidance in NUREG–1563 (1996) and its own quality assurance program. Sensitivity analyses and bounding calculations are important means of providing a risk-informed, performance-based context for the DOE data uncertainty and for evaluating the need for additional data.

DOE agreed[18] to justify the sorption coefficient distributions used in total system performance assessment.

Fault Zones

Faults can provide fast pathways for radionuclide transport in the saturated zone. Furthermore, the flow and transport characteristics of fault zone pathways can differ widely from those elsewhere in the tuff aquifer. It is not clear that DOE has adequately accounted for the possible effects of these differences in formulating transport parameter distributions (CRWMS M&O, 2000h,g). DOE agreed[19] to provide a technical basis for the importance to performance of transport through fault zones below the repository and to provide the technical basis for the parameters and distributions if such transport is found to be important to performance.

Colloidal Transport

DOE has improved its capability to model saturated zone colloid transport in recent total system performance assessment efforts (CRWMS M&O, 2000a,b), but many of the parameters (e.g., the colloid partitioning coefficient, K_c) used in the models are not supported by site characterization or laboratory data. DOE addressed this problem, to some extent, by using bounding analyses and sensitivity analyses, but there are insufficient radioelement specific data to determine whether the uncertainty in colloid transport has been constrained in the radionuclide transport in the saturated zone model abstraction. As discussed in Section 3.3.9.3.2, the two key parameters that affect saturated zone colloid transport are colloid partition coefficient K_c and colloid retardation factor R_c; colloid matrix diffusion is neglected (CRWMS M&O, 2000c). In the saturated zone, R_c is defined for the tuff aquifer on the basis of one field test, and no site-specific data are available for the alluvial aquifer (CRWMS M&O, 2000h,j). The microspheres used in the tests had diameters between 280 nm [1.1×10^{-5} in] and 640 nm [2.5×10^{-5} in] (CRWMS M&O, 2000j); this value is large compared with a typical

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[17]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7, 2000)." Letter (December 12) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[18]Ibid.

[19]Ibid.

size range in colloids from 1 nm to 450 nm [4×10^{-8} in to 2×10^{-5} in]. Smaller colloids will have a much higher specific surface area and perhaps be greater contributors to the potential colloid load. Conversely, these smaller colloids may be small enough to diffuse into the matrix and be physically filtered, reducing their impact on repository performance. DOE discusses these limitations in Section 6.1.5 of CRWMS M&O (2000j), but does not provide sensitivity analyses to test their effects on repository performance. Finally, in calculating R_c from the field data, assigning equal weight to results from the lower Prow Pass Tuff and the lower Bullfrog Tuff may not be conservative because the lower Bullfrog Tuff is the most transmissive interval at the C-Wells (CRWMS M&O, 2000a, p. 3-29). DOE agreed[20] to provide additional justification for the use of microspheres as analogs for colloids.

The K_c parameter, used to simulate reversible colloid attachment by lowering the radioelement K_d, is based on data for americium sorption to colloids and is applied to the K_d values for all reversibly attached radionuclides (CRWMS M&O, 2000h). Calculation of K_c also involves a term for colloid concentration in the water. The colloid concentration adopted is 0.03 mg/L [0.03 ppm]. This value is claimed to be conservative because it corresponds to the highest observed or expected colloid concentration (CRWMS M&O, 2000c). This concentration, however, is well below the maximum values used in release models for waste form 5 mg/L [5 ppm] and iron (hydr)oxide 1 mg/L [1 ppm] colloids derived from the engineered barrier system (CRWMS M&O, 2000o). DOE has not used any data, site-specific or not, to demonstrate that the reversible colloid attachment parameter will bound the range of possible effects of this process, nor have sensitivity analyses been employed to investigate the effects of parameter uncertainty on modeled repository performance. DOE agreed to perform such sensitivity analyses.[21,22]

<u>Alluvium</u>

As discussed in Section 3.3.9.3.2, characterization of the alluvial transport path is incomplete and uncertain. It is, therefore, important that parameter distributions used in total system performance assessment reflect those uncertainties. As acknowledged in CRWMS M&O (2000g), Total System Performance Assessment—Site Recommendation K_d distributions for alluvium are based on a limited number of site-specific tests that do not allow strong conclusions to be drawn (CRWMS M&O, 2000g, p. 92). Furthermore, DOE states in the discussion of assumptions in CRWMS M&O (2000g, p. 36) that it has not confirmed that sorption data are adequate for the alluvium. Parameter uncertainty could be particularly important for relatively poorly sorbing radioelements such as neptunium, iodine, and technetium. The distribution for alluvial effective porosity (CRWMS M&O, 2000h) uses no

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[20]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7, 2000)." Letter (December 12) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[21]Ibid.

[22]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

site-specific data and rests on unconfirmed assumptions. The alluvial aquifer is subject to possibly large spatial and stratigraphic variations in transport parameters (NRC, 2000a), which DOE has not demonstrated that uncertainty distributions accommodate. DOE agreed[23] to accomplish further alluvium characterization that should better define parameter variability (see also Section 3.3.9.3.2).

In summary, DOE has not yet assembled the information relating to methods used to characterize and propagate data uncertainty through the radionuclide transport in the saturated zone model abstraction, but has agreed to do so before submitting any license application. Key areas of data uncertainty to be addressed are K_d distributions, colloid transport parameters, and parameters specific to fault zone and alluvial transport paths.

3.3.9.4.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.9.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess radionuclide transport in the saturated zone with respect to model uncertainty being characterized and propagated through the model abstraction.

DOE does not have an alternative conceptual model for matrix diffusion in the saturated zone for total system performance assessment analyses. A sensitivity analysis would presumably provide a comparison to an alternative conceptual model with no matrix diffusion, which would provide a better understanding of the relative importance of matrix diffusion in the saturated zone. DOE agreed to provide a sensitivity analysis for matrix diffusion in the saturated zone.[24]

DOE has neglected radionuclide sorption in fractures and applied a linear sorption model to simulate radionuclide transport through the matrix and in unfractured rocks in the saturated zone in total system performance assessment (Wilson, et al., 1994; DOE, 1998b; CRWMS M&O, 2000b). Parameter variability caused by model uncertainty is believed to be contained within the probability distribution functions defined for the retardation parameters. The potential for processes such as precipitation and colloid formation to contribute to the results from batch sorption experiments is also believed to be conservatively bounded by the K_d approach (CRWMS M&O, 2000g). The acceptability of this approach to model uncertainty will depend to a large extent on the documentation of the processes and information used in the expert judgments for sorption coefficient probability distribution functions as discussed in Sections 3.3.9.3.2 and 3.3.9.3.3.

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[23]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7, 2000)." Letter (December 12) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[24]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Unsaturated and Saturated Flow Under Isothermal Conditions (October 31–November 2, 2000)." Letter (November 17) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

For saturated zone colloid transport modeling, DOE addresses model uncertainty chiefly by adopting each of two distinct attachment modes—reversible and irreversible (CRWMS M&O, 2000o). DOE has not provided sufficient evidence that its selection of colloid transport parameters bounds model uncertainties, so that the radionuclide transport in the saturated zone model abstraction realistically or conservatively bounds the possible effects of colloids. Total System Performance Assessment–Site Recommendation sensitivity analyses do not provide a clear indication of the relative importance to performance of colloid transport, and more general sensitivity analyses allow adjustment of transport parameters only within the established distribution ranges (CRWMS M&O, 2000b). Such analyses do not address the adequacy of the model itself. DOE needs to show, for example, that neglect of kinetic adsorption and desorption effects will not result in an underestimate of the effects on performance of reversible attachment. In addition, the R_c model for retardation of irreversible colloids rests on interpretation of field test results that are highly model-dependent. Breakthrough curves of microspheres at the C-Wells Complex formed a bimodal distribution that would not readily fit a simple retardation model; for example, it was necessary to assume five separate subpathways of undefined physical significance for microsphere transport (CRWMS M&O, 2000j). The irreversible colloid retardation factor distribution for the alluvial aquifer is based on a theoretical analysis (CRWMS M&O, 2000j). No site data can presently be used to confirm if the retardation model is an appropriate approach to colloidal transport in the alluvium. DOE agreed to obtain such data in the future.[25] More generally, DOE agreed to perform sensitivity analyses on the importance of colloidal transport that will address, in part, the adequacy of parameter uncertainty ranges to account for model uncertainty.[26]

In summary, DOE has not adequately assembled the information relating to methods used to characterize and propagate model uncertainty through the radionuclide transport in the saturated zone model abstraction. DOE agreed[27] to address staff concerns. These issues will be addressed chiefly through sensitivity analyses and as a result of continued data acquisition.

3.3.9.4.5 Model Abstraction Output Is Supported by Objective Comparisons

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.9.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess radionuclide transport in the saturated zone with respect to model abstraction output being supported by objective comparisons.

The available C-Wells Complex tracer test results provide convincing evidence that matrix diffusion occurs in the saturated volcanic tuffs along flow paths from Yucca Mountain (CRWMS M&O, 2000a). Not all results from the C-Wells Complex testing have been published, however. Thus, DOE agreed to provide documentation for the C-Wells Complex testing and to

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[25]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7, 2000)." Letter (December 12) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[26]Ibid.

[27]Ibid.

use the field-test data to provide justification that the data from the laboratory tests used for parameter estimations are consistent with the data from the field tests.[28]

The residence-time transfer function method for coupling matrix diffusion to the particle-tracking transport was compared with predictions from analytical solutions and other numerical models (CRWMS M&O, 2000f,q). For cases where many particles are used, predictions made using the residence-time transfer function particle-tracking approach compare well with one-dimensional analytical solutions (CRWMS M&O, 2000f, Section 6.3). A comparison of the residence-time transfer function approach to the results of a three-dimensional unsaturated zone simulation using an alternative Lagrangian-approach numerical model showed that, of the two models, the residence-time transfer function approach predicts much faster solute breakthrough times (CRWMS M&O, 2000p, Section 6.2.5). Although this verification exercise was performed using the unsaturated zone model and may not be strictly applicable for the model parameters estimated for the saturated zone transport model, the result suggests the residence-time transfer function predictions are not overly optimistic.

Verification of the ability of the particle-tracking approach to simulate advective transport of sorbing solute was also reported in CRWMS M&O (2000f). For the Total System Performance Assessment–Site Recommendation, correct implementation of the saturated zone radionuclide transport abstraction was addressed by checking that model inputs were correctly selected, that parameter functions were calculated properly, that the relationships between unsaturated zone and saturated zone outputs correctly reflected intended saturated zone behavior (e.g., more sorbing radionuclides were delayed relative to less sorbing radionuclides), and that ingrowth of radioactive daughters was simulated (CRWMS M&O, 2000c, Figures 6-176 to 6-181). The verification exercises checked both the one-dimensional and three-dimensional transport models (see Section 3.3.9.3.1), and included colloidal species. Another DOE report (CRWMS M&O, 2000i) compared one- and three-dimensional saturated zone model results for carbon and neptunium. Breakthrough curves for the two models were not identical; fractional discrepancies were largest at the 5-km [3.1-mi] points. This discrepancy is not relevant to proposed regulations concerning Yucca Mountain. At 20 and 30 km [12.4 mi and 18.6 mi], differences in breakthrough times for the two models were less than a factor of two. Although it is true that these results are, as the report says, generally comparable (CRWMS M&O, 2000i, p. 63), the differences should not be ignored in interpreting transport. Effectively, this exercise was a comparison between a detailed process-level model (three-dimensional) and a total system performance assessment implementation (one-dimensional). More such comparisons using other radionuclides and including colloidal transport may prove useful to further verify the total system performance assessment abstraction.

DOE has not provided sufficient evidence, either through field tests or natural analogs, that results from laboratory sorption/transport experiments can be extended or used to bound transport over larger distances and longer times. For example, if credit is to be taken for radionuclide attenuation, DOE should demonstrate that nonradioactive tracers used in field

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[28]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Unsaturated and Saturated Flow Under Isothermal Conditions (October 31–November 2, 2000)." Letter (November 17) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

tests (e.g., C-Wells) are appropriate homologues for radioelements. DOE agreed to provide the technical basis for the reconciliation of field and laboratory data in a future report on the C-Wells tests.[29]

The DOE discussion of model validation for the saturated zone flow and transport process model focuses on flow issues (CRWMS M&O, 2000a, Section 3.4). For transport, DOE claims that independent, quantitative comparisons are not possible because of differences between the Yucca Mountain environment and those at natural and anthropogenic analog sites. Therefore, DOE appears to be relying on site-specific tests, such as at the C-Wells Complex, for validation of data obtained in other ways such as laboratory tests. The explicit application of such exercises to model validation, however, is not apparent in available reports (e.g., CRWMS M&O, 2000a, Section 3.2.4.1.1; 2000f); an exception is the observation of matrix diffusion at the C-Wells (Section 3.3.9.3.2). DOE should clarify how field tests are used for validation of laboratory results and model abstractions (see previous paragraph). In contrast, DOE is using the C-Wells and Alluvial Testing Complex results for parameter development for colloid transport (CRWMS M&O, 2000o). At this point, no objective comparisons have been made for validating the colloidal transport parameters or abstraction. DOE needs to develop such comparisons or test colloid models by sensitivity studies and more quantitative comparisons to analogs (see Section 3.3.9.3.2). DOE agreed to perform sensitivity studies as the basis for consideration of the importance of colloid transport parameters and models to performance for the saturated zone and will document the results in updates to appropriate analyses and model reports in fiscal year 2003.

In summary, DOE has not yet adequately assembled the information relating to methods used to support model abstractions of the radionuclide transport in the saturated zone. DOE has agreed to address staff concerns chiefly through sensitivity analyses and as a result of continued data acquisition.

3.3.9.5 Status and Path Forward

Table 3.3.9-1 provides the status of all key technical issue subissues, referenced in Section 3.3.9.2, for the Radionuclide Transport in the Saturated Zone Integrated Subissue. The table also provides the related DOE and NRC agreements pertaining to the Radionuclide Transport in the Saturated Zone Integrated Subissue. The agreements listed in the table are associated with one or all five generic acceptance criteria discussed in Section 3.3.9.4. Note that the status and the detailed agreements (or path forward) pertaining to all the key technical issue subissues are provided in Table 1.1-3 and Appendix A.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (through specified testing, analyses, and the like), acceptably addresses the NRC questions so that no information beyond that provided, or agreed to, will likely be required at the time of a potential license application.

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[29]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7, 2000).” Letter (December 12) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

Table 3.3.9-1. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreement*
Radionuclide Transport	Subissue 1—Radionuclide Transport Through Porous Rock	Closed-Pending	RT.1.02 through RT.1.05
	Subissue 2—Radionuclide Transport Alluvium	Closed-Pending	RT.2.01 through RT.2.07 RT.2.10
	Subissue 3—Radionuclide Transport Through Fractured Rock	Closed-Pending	RT.3.07 RT.3.08 RT.3.09
	Subissue 4—Nuclear Criticality in the Far Field	Closed-Pending	RT.4.01 RT.4.03
Unsaturated and Saturated Flow Under Isothermal Conditions	Subissue 5—Saturated Zone Ambient Flow Conditions and Dilution Processes	Closed-Pending	USFIC.5.03
	Subissue 6—Matrix Diffusion	Closed-Pending	USFIC.6.04
Structural Deformation and Seismicity	Subissue 3—Fracturing and Structural Framework of the Geologic Setting	Closed-Pending	None
Container Life and Source Term	Subissue 5—Effect of In-Package Criticality on Waste Package and Engineered Barrier System Performance	Closed-Pending	CLST.5.04
Total System Performance Assessment Integration	Subissue—1 System Description and Demonstration of Multiple Barriers	Closed Pending	None
	Subissue 2—Scenario Analysis and Event Probability	Closed Pending	TSPAI.2.01 TSPAI.2.02 TSPAI.2.03
	Subissue 3—Model Abstraction	Closed Pending	TSPAI.3.30 TSPAI.3.31 TSPAI.3.32
	Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards	Closed Pending	None

*Related DOE and NRC agreements are associated with one or all five generic acceptance criteria.

Note: Key Technical Issue Agreement GEN.1.01 pertains to multiple integrated subissues, as well as some specific issues related to this integrated subissue.

3.3.9.6 References

Barnard, R.W., M.L. Wilson, H.A. Dockery, J.H. Gauthier, P.G. Kaplan, R.R. Eaton, F.W. Bingham, and T.H. Robey. "TSPA 1991: An Initial Total-System Performance Assessment for Yucca Mountain." SAND91–2795. Albuquerque, New Mexico: Sandia National Laboratories. 1992.

Bechtel SAIC Company, LLC. "FY01 Supplemental Science and Performance Analyses." Vol. 1: Scientific Bases and Analyses. TDR–MGR–MD–000007. Revision 00 ICN 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2001a.

Bechtel SAIC Company, LLC. "FY01 Supplemental Science and Performance Analyses." Vol. 2: Performance and Analyses. TDR–MGR–PA–000001. Revision 00. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2001b.

CRWMS M&O. "Saturated Zone Flow and Transport Process Model Report." TDR–NBS–HS–000001. Revision 00 ICN 02. Las Vegas, Nevada: CRWMS M&O. 2000a.

———. "Total System Performance Assessment for the Site Recommendation." TDR–WIS–PA–000001. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000b.

———. "Total System Performance Assessment (TSPA) Model for Site Recommendation." MDL–WIS–PA–000002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000c.

———. "Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations." TDR–WIS–RL–000001. Revision 04. Las Vegas, Nevada: CRWMS M&O. 2000d.

———. "Particle Tracking Model and Abstraction of Transport Processes." ANL–NBS–HS–000026. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000e.

———. "Saturated Zone Transport Methodology and Transport Component Integration." ANL–NBS–HS–000010. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000f.

———. "Unsaturated Zone and Saturated Zone Transport Properties." ANL–NBS–HS–000019. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000g.

———. "Uncertainty Distribution for Stochastic Parameters." ANL–NBS–MD–000011. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000h.

———. "Input and Results of the Base Case Saturated Zone Flow and Transport Model for TSPA." ANL–NBS–HS–000030. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000i.

———. "Saturated Zone Colloid-Facilitated Transport." ANL–NBS–HS–000031. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000j.

———. "Inventory Abstraction." ANL–WIS–MD–000006. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000k.

———. "Features, Events, and Processes—System Level and Criticality." ANL–WIS–MD–000019. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000l.

———. "Probability of Criticality Before 10,000 Years." CAL–EBS–NU–000014. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000m.

———. "Probability Distribution for Flowing Interval Spacing." ANL–NBS–MD–000003. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000n.

———. "Waste Form Colloid-Associated Concentration Limits: Abstraction and Summary." ANL–WIS–MD–000012. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000o.

———. "Analysis of the Base-Case Particle Tracking Results of the Base-Case Flow Fields." ANL–NBS–HS–000024. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000p.

———. "Features, Events, and Processes in SZ Flow and Transport." ANL–NBS–MD–000002. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2001.

DOE. "Disposal Criticality Analysis Methodology Topical Report." YMP/TR–004Q. Revision 00. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 1998a.

———. "Viability Assessment of a Repository at Yucca Mountain." Overview and all five volumes. DOE/RW–0508. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 1998b.

———. "Disposal Criticality Analysis Methodology Topical Report." YMP/TR–004Q. Revision 01. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 2000.

Flint, L.E. "Characterization of Hydrogeologic Units Using Matrix Properties, Yucca Mountain, Nevada." U.S. Geological Survey Water Resources Investigations Report 97-4243. 1998.

NRC. NUREG–1563, "Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program." Washington, DC: NRC. November 1996.

———. "Issue Resolution Status Report, Key Technical Issue: Unsaturated and Saturated Flow Under Isothermal Conditions." Revision 2. Washington, DC: NRC. 1999a.

———. NUREG–1668, "NRC Sensitivity and Uncertainty Analyses for a Proposed HLW Repository at Yucca Mountain, Nevada, Using TPA 3.1." Washington, DC: NRC. March 1999b.

———. "Issue Resolution Status Report Key Technical Issue: Radionuclide Transport." Revision 2. Washington, DC: NRC. 2000a.

———. "Issue Resolution Status Report, Key Technical Issue: Structural Deformation and Seismicity." Revision 3. Washington, DC: NRC. 2000b.

———. "Issue Resolution Status Report, Key Technical Issue: Total System Performance Assessment and Integration." Revision 3. Washington, DC: NRC. 2000c.

———. NUREG–1746, "System-Level Repository Sensitivity Analyses Using TPA Version 3.2 Code." Washington, DC: NRC. August 2001a.

———. "Issue Resolution Status Report, Key Technical Issue: Container Life and Source Term." Revision 3. Washington, DC: NRC. 2001b.

———. NUREG–1804, "Yucca Mountain Review Plan—Draft Report for Comment." Revision 2. Washington, DC: NRC. March 2002.

Reimus, P.W., A. Adams, M.J. Haga, A. Humphrey, T. Callahan, I. Anghel, and D. Counce. "Results and Interpretation of Hydraulic and Tracer Testing in the Prow Pass Tuff at the C-Holes." Yucca Mountain Project Milestone SP32E7M4. Los Alamos, New Mexico: Los Alamos National Laboratory. 1999.

Sudicky, E.A. and E.O. Frind. "Contaminant Transport in Fractured Porous Media: Analytical Solutions for a System of Parallel Fractures." *Water Resources Research.* Vol. 18, No. 6. pp. 1,634–1,642. 1982.

Triay, I.R., A. Meijer, J.L. Conca, K.S. Kung, R.S. Rundberg, and E.A. Strietelmeier. "Summary and Synthesis Report on Radionuclide Retardation for the Yucca Mountain Site Characterization Project, Milestone 3784M." LA–13262–MS. Los Alamos, New Mexico: Los Alamos National Laboratory, Chemical Science and Technology Division. 1997.

Wilson, M.L., J.G. Gauthier, R.W. Barnard, G.E. Barr, H.A. Dockery, E. Dunn, R.R. Eaton, D.C. Guerin, N. Lu, M.J. Martinez, R. Nilson, C.A. Rautman, T.H. Robey, B. Ross, E.E. Ryder, A.R. Schenker, S.A. Shannon, L.H. Skinner, W.G. Halsey, J.D. Gansemer, L.C. Lewis, A.D. Lamont, I.R. Triay, A. Meijer, and D.E. Morris. "Total-System Performance Assessment for Yucca Mountain—SNL Second Iteration (TSPA–1993)." SAND93–2675. Albuquerque, New Mexico: Sandia National Laboratories. 1994.

Zyvoloski, G.A., B.A. Robinson, Z.V. Dash, and L.L. Trease. "Summary of Models and Methods for the FEHM Application—A Finite Element Heat—and Mass-Transfer Code." LA–13307–MS. Los Alamos, New Mexico: Los Alamos National Laboratory. 1997.

3.3.10 Volcanic Disruption of Waste Packages

3.3.10.1 Description of Issue

The Volcanic Disruption of Waste Packages Integrated Subissue evaluates the interaction of ascending basaltic magma with subsurface repository systems and the establishment of flow paths to the surface as part of a possible volcanic eruption. Key processes associated with this integrated subissue are (i) ascent of basaltic magma in the Yucca Mountain region, (ii) interaction of the ascending magma with rock in the modified stress regime around repository drifts, (iii) initial interactions between ascending magma and repository drifts, (iv) interactions between magma in drifts and engineered barriers, (v) establishment of magma flow paths to the surface, and (vi) effect of sustained magma flow on engineered barrier performance and possible waste package and high-level waste disaggregation. The transition to the Airborne Transport of Radionuclides Integrated Subissue occurs when high-level waste is incorporated into the flowing basaltic magma that is erupting subaerially. Interactions between basaltic magma and waste packages not located along a subvolcanic conduit to the surface are evaluated in the Mechanical Disruption of Engineered Barriers Integrated Subissue. The relationship of this integrated subissue to other integrated subissues is depicted in Figure 3.3.10-1. The overall organization and identification of all the integrated subissues are depicted in Figure 1.1-2.

This section provides a review of the abstractions of volcanic disruption of waste packages by DOE in the Total System Performance Assessment–Site Recommendation (CRWMS M&O, 2000a). The DOE description and technical basis for its analyses of volcanic disruption of waste packages are documented in CRWMS M&O (2000b) and three supporting analysis and model reports (CRWMS M&O, 2000c,d,e). Calculation documents (CRWMS M&O, 2000f,g) and the analysis and model report (CRWMS M&O, 2000h) also provide information relevant to this integrated subissue.

3.3.10.2 Relationship to Key Technical Issue Subissues

The Volcanic Disruption of Waste Packages Integrated Subissue incorporates subject matter previously captured in the following key technical issue subissues:

- Igneous Activity: Subissue 1—Probability of Igneous Activity (NRC, 1999)
- Igneous Activity: Subissue 2—Consequences of Igneous Activity (NRC, 1999)
- Container Life and Source Term: Subissue 2—Mechanical Disruption of Waste Packages (NRC, 1999)

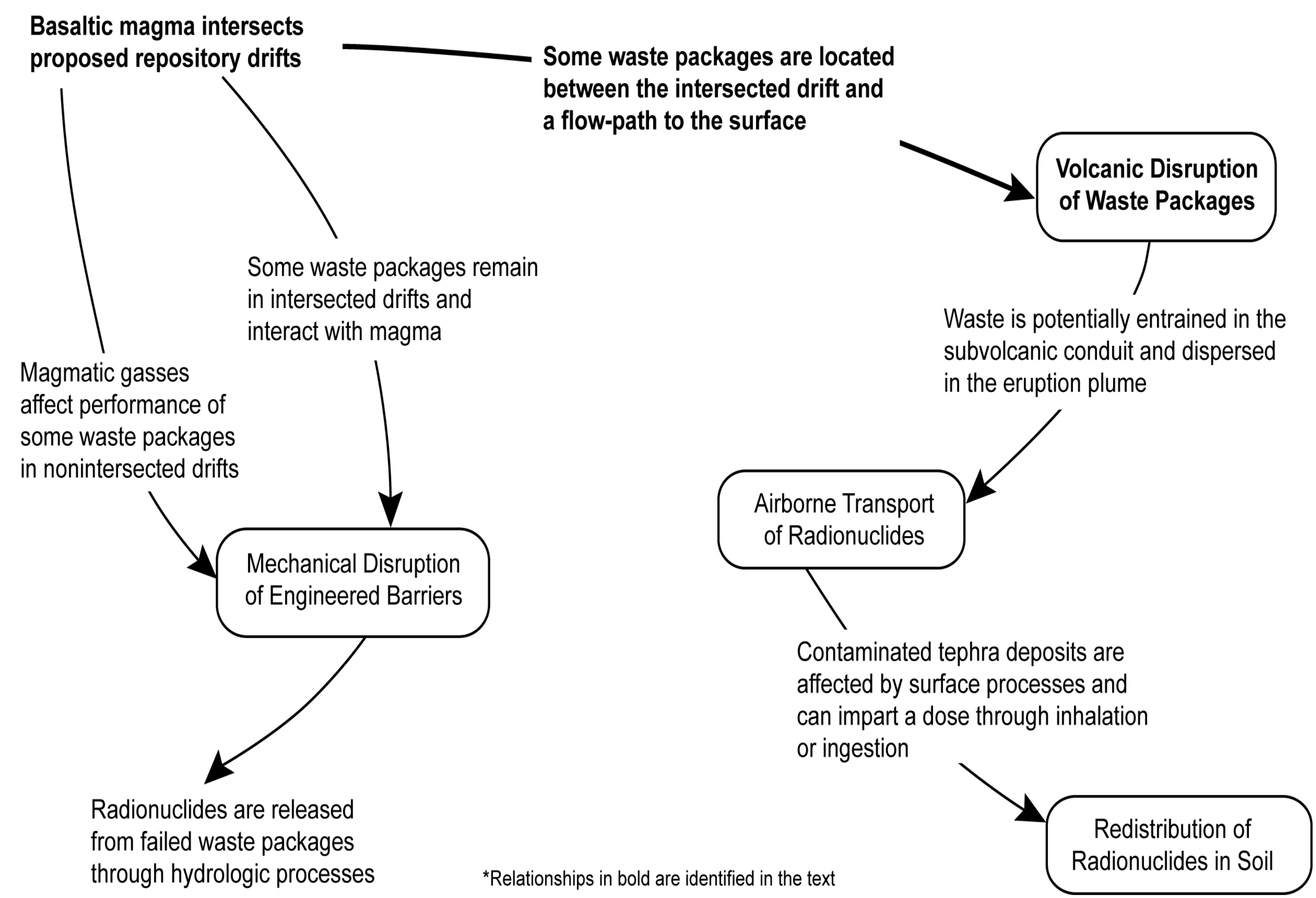

Figure 3.3.10-1. Diagram Illustrating the Relationship Between Volcanic Disruption of Waste Packages and Other Integrated Subissues

- Total System Performance Assessment and Integration: Subissue 1—System Description and Demonstration of Multiple Barriers (NRC, 2000)

- Total System Performance Assessment and Integration: Subissue 2—Scenario Analysis and Event Probability (NRC, 2000)

- Total System Performance Assessment and Integration: Subissue 3—Model Abstraction (NRC, 2000)

- Total System Performance Assessment and Integration: Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards (NRC, 2000)

The key technical issue subissues formed the basis for the previous versions of the issue resolution status reports and also were the basis for technical exchanges with DOE where agreements were reached on the additional information DOE needed to provide to resolve the subissue. The resolution status of this integrated subissue is based on the resolution status of each of the contributing key technical issue subissues. The subsequent sections incorporate applicable portions of these key technical issue subissues, however, no effort has been made to explicitly identify each subissue.

3.3.10.3 Importance to Postclosure Performance

One aspect of risk-informing the NRC review was to determine how this integrated subissue is related to the DOE repository safety strategy. The Total System Performance Assessment–Site Recommendation reports no radiological risk in 10,000 years from the basecase repository (CRWMS M&O, 2000a). Postclosure volcanism has a maximum probability weighted risk of approximately 0.1 µSv/yr [0.01 mrem/yr], however, DOE has not classified it as a principal factor. Based on DOE analyses, intrusive igneous activity has a probability weighted risk of approximately 1 µSv/yr [0.1 mrem/yr], and DOE has classified it as a principal factor (CRWMS M&O, 2000i). Both these risk values increase by approximately one order of magnitude when probability values acceptable for prelicensing issue resolution are used (CRWMS M&O, 2000a).[1] Volcanism risks increase to approximately 1 µSv/yr [0.1 mrem/yr] in supplemental analyses presented in Bechtel SAIC Company, LLC (2001a). In contrast, risks from intrusive igneous activity decrease by approximately an order of magnitude to 0.1 µSv/yr [0.01 mrem/yr] in Bechtel SAIC Company, LLC (2001a). These levels of igneous risk clearly exceed calculated risks from other postclosure features, events, and processes in CRWMS M&O (2000a) or Bechtel SAIC Company, LLC (2001a,b).

Concerns have been raised with the technical bases DOE has used to evaluate both extrusive and intrusive igneous activity in the Total System Performance Assessment–Site

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[1]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (August 29–31, 2000)." Letter (October 23) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

Recommendation[2,3] (Hill and Connor, 2000). Analyses presented in, for example, NRC (1999) also demonstrate that probability-weighted risk from postclosure volcanism may be on the order of 10 µSv/yr [1 mrem/yr], with significant uncertainties associated with this value. DOE will need to continue resolving technical concerns with igneous processes, including those associated with postclosure volcanism, due to significant uncertainties with igneous risk calculations and the absence of other processes that lead to similar levels of risk in postclosure total system performance (i.e., CRWMS M&O, 2000a).

Processes of magma-repository interaction, which form the primary emphasis of the Volcanic Disruption of Waste Packages Integrated Subissue, affect the amount of radionuclides potentially released by both volcanic and groundwater pathways. Evaluation of the risks associated with these release pathways is predicated on a well-supported understanding of the magnitude of ascending basaltic magma that can interact with subsurface repository systems and of the possible rates of interaction. This interaction directly controls the amount and character of high-level waste potentially available for subsequent volcanic and hydrologic transports.

3.3.10.4 Technical Basis

As outlined in NRC (1999) and Hill and Connor (2000), previous DOE total system performance assessments have evaluated a limited range of effects from volcanic disruption of the proposed repository. In the Total System Performance Assessment–Viability Assessment, DOE relied on several critical assumptions to support the conclusion that there is no risk from volcanic disruption during a 10,000-year performance period (DOE, 1998). As discussed in Section 4.2 of Hill and Connor (2000), these assumptions were based on levels of information not adequate to substantiate waste package and waste form resiliences during igneous events.

Significant changes were made to DOE igneous activity models subsequent to the Total System Performance Assessment–Viability Assessment, as discussed in Section 4.2 of Hill and Connor (2000). These changes have addressed many technical concerns with key modeling assumptions previously made by DOE. Most importantly, DOE currently assumes waste packages fail on intersection by an erupting subvolcanic conduit and that all contained high-level waste is available for entrainment (CRWMS M&O, 2000d). In addition, the models now include a significant reduction in high-level waste particle size during volcanic disruption, and all eruptions have violent strombolian dispersal characteristics (CRWMS M&O, 2000e). The Total System Performance Assessment–Viability Assessment also assumed passive flow of magma from a dike segment that intersected subsurface drifts (DOE, 1998). Scoping calculations by Woods and Sparks (1998) indicated flow of magma would likely be more rapid and energetic than previously modeled. These calculations led to significant revisions of the

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[2]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (June 21–22, 2001)." Letter (June 27) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[3]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

DOE model abstraction (e.g., CRWMS M&O, 2000c). DOE also agreed to provide additional modeling support for magma-repository interactions, including evolution of potential magma flow paths through the duration of an igneous event. The Volcanic Disruption of Waste Packages Integrated Subissue currently is closed-pending, after agreements reached at the September 5, 2001, Technical Exchange on Igneous Activity.[4]

NRC developed a Yucca Mountain Review Plan (2002) that is consistent with the acceptance criteria and review methods found in previous issue resolution status reports. A review of DOE approaches for including volcanic disruption of waste packages in total system performance assessment abstractions is provided in the following subsections. The review is organized according to the five acceptance criteria identified in Section 1.5 as follows: (i) System Description and Model Integration Are Adequate, (ii) Data Are Sufficient for Model Justification, (iii) Data Uncertainty Is Characterized and Propagated Through the Model Abstraction, (iv) Model Uncertainty is Characterized and Propagated Through the Model Abstraction, and (v) Model Abstraction Output Is Supported by Objective Comparisons.

3.3.10.4.1 System Description and Model Integration Are Adequate

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.10.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess volcanic disruption of waste packages with respect to system description and model integration.

The DOE approach to evaluating volcanic disruption of waste packages involves several conceptual models.

- Ascending basaltic magma interacts with the subsurface rock surrounding the repository drifts. Based on calculations for an older high thermal-load repository design, CRWMS M&O (2000c) indicates that ascending magma may be deflected from repository drifts during the first 2,000 years of postclosure. This deflection is attributed to the rotation of rock-stress directions in response to heating of the rock by emplaced waste. The Total System Performance Assessment–Site Recommendation, however, apparently does not take credit for this magma deflection in current analyses (CRWMS M&O, 2000a).

- Ascending magma intersects the repository drifts. Because the magma is thought to be under lithostatic pressure {i.e., about 7.5 MPa [1,088 psi]}, volatiles in the magma expand upon entering the drift, and magma flows rapidly into the drift (CRWMS M&O, 2000c). The shock associated with this initial entry may be sufficient to wholly damage three waste packages on either side of the intersecting dike (CRWMS M&O, 2000c,d,f).

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[4] Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

- Magma continues to flow into a nonbackfilled drift at a rate sufficient to block the ends of drifts with an accumulation of drip shields, debris, and quenched magma fragments (CRWMS M&O, 2000c,d). For a backfilled repository, the extent of this flow is much more limited, with drifts thought to be wholly plugged within 15 m [49 ft] on either side of the intersecting dike (CRWMS M&O, 2000c).

- Magma fills the plugged drift until the pressure in the magma exceeds the force needed to hydraulically fracture the drift roof and propagate magma to the surface (CRWMS M&O, 2000c).

- A subvolcanic conduit forms 77 percent of the time at the point of dike intersection in the drift. The remaining 23 percent of the time, the conduit forms in the pillars, and no high-level waste is released through volcanism (CRWMS M&O, 2000c,d,g).

- All waste packages intersected by the conduit are assumed to fail from the adverse physical, thermal, and chemical conditions in the erupting conduit (CRWMS M&O, 2000d,f).

These models rely on several key assumptions that are not consistent with physical processes generally associated with igneous events. Most significant of these assumptions is that the pressure in an ascending basaltic magma at 300-m [984-ft] depth is equivalent to a 7.5-MPa [1,088-psi] lithostatic confining pressure (CRWMS M&O, 2000c). Magma likely ascends in the shallow crust by propagating a fracture that extends vertically for some distance above the dike tip (e.g., Delaney, et al., 1986; Rubin, 1993). Differences in horizontal deviatoric stress are relatively small in the Yucca Mountain region (e.g., Morris, et al., 1996). Thus, a hydraulic pressure greater than lithostatic is necessary to dilate a 300-m [984-ft]-deep fracture from approximately 0.1 cm [0.04 in] to typical dike widths of approximately 100 cm [39 in] (e.g., Rubin, 1993). Many authors calculate this pressure to be 1–10 MPa [145–1,450 psi] greater than lithostatic pressure for shallow dikes (Delaney, et al., 1986; Rogers and Bird, 1987; Baer and Reches, 1991; Rubin, 1993; Woods and Sparks, 1998). This amount of magmatic overpressure is important because it directly affects the potential rate of magma flow into the drift, which, in turn, determines the volume of ascending magma that can be captured by the intersected drift (e.g., Bokhove and Woods, 2000; Woods, et al., 2001). In addition, as the drift fills with magma and pressure reequilibrates with the pressure in the intersecting dike, the amount of magmatic overpressure affects when and where basaltic magma can break out of the drift roof and propagate to the surface (CRWMS M&O, 2000c; Woods, et al., 2001). In the Igneous Activity Key Technical Issue agreement 2.18, DOE agrees to evaluate how the presence of repository structures may affect magma ascent processes. This evaluation will use a range of physical conditions appropriate for the duration of igneous events.[5]

The pressure in the magma system likely affects the extent of magma flow into the subsurface drift system. The DOE volcanic disruption of waste packages model relies on debris plugs to

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[5]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001).” Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

form at the ends of the intersected drift (CRWMS M&O, 2000c,d). These plugs prevent magma from flowing into the drift network and from potentially damaging additional waste packages. In addition, these debris plugs must be stronger than the fracture strength of the roof rock along the drift to direct the repressurized magma vertically for the duration of the igneous event. Models to date have not demonstrated that these debris plugs will have a mechanical strength sufficient to withstand a 3–7 MPa [435–1,015 psi] repressurization of the magma-filled drift, throughout the duration of an igneous event. Although access drifts will likely be completely backfilled (Bechtel SAIC Company, LLC, 2001b), the absence of backfill could allow magma to break through the debris plugs and flow into drifts not directly intersected by a dike (Hill and Connor, 2000; Woods, et al., 2001). DOE agreed to evaluate the mechanical strength and durability of natural or engineered barriers that could restrict magma flow within intersected drifts, using an appropriate range of repository design options.[6]

DOE models for the deviation of ascending dikes caused by thermally altered stresses around proposed repository drifts have examined a limited range of processes and geologic couplings. Although these analyses are not currently used to reduce volcanism risk (CRWMS M&O, 2000a,d), the model implies that dike intersection with repository drifts may be unlikely during the first 2,000 years of postclosure when heat released from the waste packages may be greatest. Models that evaluate potential changes in rock stress because of thermal effects from the emplacement of waste will need to use consistent and appropriate design characteristics in the analyses; CRWMS M&O (2000c) uses wall-rock temperatures that are significantly higher than currently proposed repository designs. Stress models also will need to consider how stress induced through thermal expansion can be accommodated through strains along existing structures or through the propagation of new strain structures such as fractures. Topographic variations above the proposed repository horizon also can affect the maximum amount of stress that can accumulate from thermal expansion before a displacement strain occurs. Models also will need to evaluate the effects of differential thermal expansion because proposed waste-package loadings in drifts will not result in a uniform heat source (e.g., Bechtel SAIC Company, LLC, 2001b). Models for potential deviation of ascending magma away from proposed repository drifts will need to account for complex couplings between heterogeneous thermal stress and multiple strain accommodation processes. DOE agreed to evaluate the potential effects of topography and stress and the likely strain responses on existing or new geologic structures resulting from thermal loading of high-level waste, in future models of dike ascent.[7]

The processes that control the initial development of a subvolcanic conduit are poorly known. A common observation at basaltic cinder cone volcanoes is that a roughly 1-km [3,280-ft]-long fissure forms during the first 24 hours of an eruption, which supports a fire-fountain eruption. A central vent then localizes along the fissure, with the eruption becoming more energetic and forming a dispersive cinder cone volcano (e.g., Thorarinsson, et al., 1973; Fedotov, et al., 1984). One explanation for this process is that a preferred vertical-flow pathway develops in

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[6]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12,) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[7]Ibid.

the dike-fed fissure as a result of irregularities in dike width or fracture roughness. Magma in a typical shallow dike that is ascending slowly can solidify in several hours (Delaney and Pollard, 1982; Huppert and Sparks, 1985; Bruce and Huppert, 1989, 1990). Thus, any feature that favors vertical magma ascent should favor the localization of a subvolcanic conduit, because the conduit will not form in stagnated, solidifying basalt. Repository drifts represent one possible low-resistance flow path for vertically ascending magma, especially as calculations indicate magma will accelerate into the intersected drifts because of decompression effects (e.g., Woods and Sparks, 1998; Bokhove and Woods, 2000; CRWMS M&O, 2000c; Woods, et al., 2001). Streamlines for magma in the intersecting dike should focus on the drifts, with lower ascent velocities or possibly stagnation occurring in the areas between the drifts. The effect of focusing the vertical ascent of magma toward drifts may localize subsequent conduit formation in the drift. The potential effects of flow focusing on conduit formation, however, has not been evaluated. CRWMS M&O (2000h) asserts that the presence of repository drifts causes conduits to localize there only 50 percent of the time, and that conduits will form randomly in a drift an additional 27 percent of the time. No technical basis is supplied in CRWMS M&O (2000a–h), however, that evaluates potential magma flow processes in the presence of repository drifts. Such a technical basis is required to reduce resulting dose calculations by 23 percent (CRWMS M&O, 2000a,d), which is the credit taken for a conduit forming outside a repository drift. DOE agrees to evaluate how the presence of repository structures may affect conduit localization and evolution of the conduit system. This evaluation will include a range of physical conditions appropriate for the duration of basaltic igneous events.[8]

Current DOE total system performance assessment models (CRWMS M&O, 2000a,d,g) calculate the amount of high-level waste available for volcanic disruption by determining the number of waste packages that fall within the diameter of a circle centered on the point of dike-drift intersection. This modeling approach does not consider how the presence of repository structures may potentially affect igneous processes. Models presented in CRWMS M&O (2000c) conclude that magma may not ascend above the level of the drift until the drift is filled with magma at equilibrium pressure with the dike. There is no basis presented in CRWMS M&O (2000a–h), however, that demonstrates why magma should resume propagating vertically at the initial point of dike intersection rather than at some other location where the overlying lithostatic load is lower (e.g., Woods and Sparks, 1998; NRC, 1999; Woods, et al., 2001). Bedrock thicknesses overlying the proposed repository range from 200 to 300 m [656 to 984 ft]. Assuming that the overlying rock has an average density of 2,400 kg m^{-3} [150 lb/ft^3], results in a lithostatic load that ranges from approximately 4.7 MPa [682 psi] on the east to approximately 7.1 MPa [1,030 psi] beneath Yucca Crest. Subvertical breakout toward Solitario Canyon also may represent a potential pathway with lower lithostatic load than pathways to the east. Assuming a vertical fracture, the amount of horizontal force needed to dilate the fracture to 1 m [3.3 ft] is then controlled by the thickness of overlying rock, because other parameters essentially are equivalent along the drift length. Thus, a dike intersecting the western part of a drift has sufficient overpressure to dilate rock with a 7.1-MPa [1,030-psi]

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[8]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

lithostatic load during ascent. If the drift fills with magma and begins to repressurize, hydrofracturing and breakout through the drift roof are more likely to occur on the eastern part of the drift, or perhaps subvertically toward Solitario Canyon where the overlying rock is thinnest, and less fluid pressure is needed to dilate a fracture (e.g., CRWMS M&O, 2000c). In this situation, magma could flow horizontally through the drift between the initial intersection point and the final breakout point. Waste packages in this flow path would most likely fail because of the high thermal, physical, and chemical loads in the erupting volcanic conduit (CRWMS M&O, 2000d,f). Because the length of this horizontal flow path could readily exceed 150 m [492 ft] (i.e., the maximum diameter of the DOE subvolcanic conduit), more waste packages could be disrupted and entrained than currently calculated (CRWMS M&O, 2000a,d). DOE agreed to evaluate how magma flow paths may develop through time as a result of interactions with subsurface repository structures and the surrounding rock.[9]

Basaltic igneous events, like those that occurred in the geologic past in the Yucca Mountain region, can sustain weeks to years, and perhaps decades, of periodic activity. Subvolcanic conduits clearly evolve throughout the course of an eruption, because variations in mass flow result in wall-rock entrainment and conduit widening (Macedonio, et al., 1994; Valentine and Groves, 1996; Doubik and Hill, 1999). Models will need to evaluate the effects of sustained igneous activity on (i) changes in conduit geometry that could affect additional waste packages, (ii) strength of barriers restricting magma flow in drifts, and (iii) effects of sustained flow on waste package damage and waste entrainment (e.g., Woods, et al., 2001). In addition to the evolution of flow paths, DOE agreed to evaluate waste package response to a range of flow conditions that include pathways that may develop through drifts.[10]

The following is a summary evaluation for system description and model integration for the volcanic disruption of waste packages abstraction. The Disruptive Events Process Model Report and associated analysis and model reports (i.e., CRWMS M&O, 2000b–h) do not adequately consider the range of physical processes generally associated with igneous events. Ascending basaltic magmas must have a fluid pressure greater than lithostatic to dilate fractures significantly. DOE models that restrict magma flow to within a drift will need to evaluate the effects of this overpressure on structures or debris assumed to block drift ends. DOE models that propose stress reorientation and resulting dike deflection away from thermally loaded drifts will need to examine an appropriate range of stress-strain relationships rather than unbounded strain accumulation effects. The models presented to date do not adequately consider how the presence of subsurface repository structures may affect typical igneous processes. Repository drifts may localize magma flow significantly during the initial stages of a potential igneous event. This localization may force conduit development into drift areas and greatly restrict the ability of a conduit to form in the pillars. In addition, the continued path of magma ascent is poorly constrained and could range from the DOE model (e.g., CRWMS M&O, 2000h) to a model where magma is diverted down a number of drifts before resuming ascent (e.g., Woods and Sparks, 1998; Woods, et al., 2001). Magma

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[9]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[10]Ibid.

diversion could disrupt a larger number of waste packages than currently modeled. The initial flow of magma into a drift likely will be rapid (i.e., Woods and Sparks, 1998; CRWMS M&O, 2000c). Modeling initial and sustained flow of magma through repository systems is complex but appears necessary to support DOE risk assessments. Based on agreements reached at the September 5, 2001, Technical Exchange on Igneous Activity, DOE has a reasonable path forward to address staff questions and uncertainties regarding the system description and model integration for volcanic disruption of waste package processes. With the exceptions already described, DOE models appear generally consistent with the types of igneous activity likely to occur in possible future Yucca Mountain region basaltic igneous events. Model assumptions are generally consistent with available data, but interrelationships between important processes need to be better described and justified in agreed-on investigations by DOE. For example, conduit development is a dynamic process that occurs throughout an igneous event (e.g., Woods, et al., 2001). Current models only evaluate the initial stages of conduit development and do not consider how magma flow paths may change during the course of a basaltic eruption in response to flow within the intersected drift system. These comments are supported by previous reviews and analyses conducted for the Igneous Activity Key Technical Issue (e.g., NRC, 1999).

3.3.10.4.2 Data Are Sufficient for Model Justification

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.10.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess volcanic disruption of waste packages with respect to data being sufficient for model justification.

There are, however, few data used directly in the analysis of volcanic disruption of waste packages. Data for the physical and chemical characteristics of basaltic magmas used in CRWMS M&O (2000c,e) appear reasonable for evaluating volcanic disruption of waste package processes. The range and distribution of subvolcanic conduit dimensions in CRWMS M&O (2000e) also appear reasonable for basaltic cinder cone volcanoes. The number of conduits for each igneous event is derived from a generally reasonable interpretation of Yucca Mountain region volcano characteristics in CRWMS M&O (1996).

DOE has assembled sufficient information to support conclusions (CRWMS M&O, 2000d) that all waste package components fail when exposed to magma flowing in the subvolcanic conduit. Data are not available on the responses of proposed waste-package components to the physical and thermal conditions of an igneous event representative of the Yucca Mountain region. Internal pressurization analyses in CRWMS M&O (2000f) are combined with reasonable assumptions regarding dynamic stresses to conclude all waste packages in the subvolcanic conduit will fail (CRWMS M&O, 2000d). Scoping analyses presented in NRC (1999) also conclude that waste package failure is a reasonable assumption for the thermal, physical, and chemical conditions likely to occur in an erupting subvolcanic conduit. This assumption will not underestimate risk to public health and safety, and there are no alternative interpretations to available data that would indicate a greater level of risk. Staff recognize that a detailed engineering analysis has not been conducted for waste package performance during basaltic volcanic events. If the assumption of waste package failure during basaltic volcanic events is not used in future DOE models, DOE agreed to explicitly evaluate waste package response to stresses from thermal and mechanical effects associated with exposure to basaltic magma along relevant flow pathways.[11]

3.3.10.4.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.10.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess volcanic disruption of waste packages with respect to data uncertainty being characterized and propagated through abstraction.

The number of waste packages directly intersected by a basaltic subvolcanic is calculated using a range of conduit characteristics summarized in CRWMS M&O (2000e,h). Current total system performance assessment models sample a range of conduit diameters and the number of conduits per igneous event. These parameter ranges appear reasonably consistent with the underlying technical basis (CRWMS M&O, 1996, 2000e). Using simple geometric relationships, models then calculate the number of waste packages intersected by each sampled conduit. The range sampled (CRWMS M&O, 2000a) for the number of waste packages entrained in the erupting conduit is the simple product of the number of waste packages intersected per conduit diameter and the number of conduits that form in each sampled event.

DOE performed only one sensitivity calculation in the Total System Performance Assessment–Site Recommendation relative to volcanic disruption of waste packages (CRWMS M&O, 2000a). Variations in the number of waste packages entrained in the erupting subvolcanic conduit (i.e., 6 at 5th percentile, 16 at 95th percentile) had a factor of 1.5 variation in probability-weighted dose. Based on this sensitivity, one order of magnitude increase in dose appears likely for one order of magnitude increase in the number of waste packages entrained in the eruptive subvolcanic conduit. Thus, the physical dimensions of the subvolcanic conduit in the presence of repository drifts are a critical parameter in postclosure performance.

3.3.10.4.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.10.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess volcanic disruption of waste packages with respect to model uncertainty.

CRWMS M&O (2000c) presents a single conceptual model for the initial interaction between ascending magma and thermally loaded repository drifts. This model does not discuss how significant variations in repository design, including currently proposed design alternatives, can potentially affect the distribution of rock stress around repository drifts. Also not evaluated are

[11]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001).” Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

potentially significant variations in thermal loads from different waste-package inventories, which may produce significant differences in thermal expansion in affected subsurface rock. In addition, this model does not address uncertainties on how stress induced through differential thermal expansion may be accommodated through resulting strain on existing geologic structures or the formation of new strain accommodation structures. Each of these processes can affect how ascending magma interacts with the potentially disturbed zone of rock around repository drifts. CRWMS M&O (2000a) apparently does not take credit for dike deflection during the first 2000 years of postclosure (i.e., CRWMS M&O, 2000c). Acceptance of this model in future total system performance assessments requires DOE to address these significant model uncertainties. Agreed-on investigations by DOE should address these concerns with model uncertainties.[12]

CRWMS M&O (2000c) presents several alternative conceptual models for magma flow into open or backfilled drifts. The performance implications of these alternative models, however, are not discussed. For example, CRWMS M&O (2000c) discusses multiple flow modes that pyroclastic flows or liquid magma could follow, which result in different rates and extents of magma interactions within and between proposed repository drifts (i.e., Woods, et al., 2001). Only one of those models is evaluated within the Total System Performance Assessment–Site Recommendation: flow into and repressurization within each discretely intersected drift (CRWMS M&O, 2000a). A critical assumption for these flow models is that the ends of repository drifts are plugged by debris, which allows magmatic pressures to reestablish in the drifts (CRWMS M&O, 2000c). No technical basis is provided to demonstrate that debris plugs can withstand magmatic pressures of 3–7 MPa [435–1,015 psi] at representative magmatic temperatures, and alternatives to plugged drifts are not evaluated in CRWMS M&O (2000a,c,d). Although drifts may be plugged by debris immediately following initial flow of magma into a drift, debris plugs are not certain to form at the ends of drifts. In addition, the mechanical strength of anticipated debris plugs will need to be evaluated for the range of physical conditions associated with the duration of an igneous event. A reasonable alternative interpretation is that if debris plugs form, they may fail during repressurization of the magma-drift system in response to heating of the debris and the 3–7 MPa [435–1,015 psi] pressures within the magma system. Magma could then flow beyond directly intersected drifts, create additional locations where conduit formation may be favored, and affect a larger number of waste packages than are currently evaluated in CRWMS M&O (2000a,c,d,g). Agreed-on investigations by DOE, however, should resolve these concerns regarding alternative flow paths during potential igneous events.[13]

CRWMS M&O (2000c) concludes that debris-plugged drifts will fill with magma and reequilibrate with the pressure in the underlying magmatic system. Although some flow modes are thought to favor repropagation of the dike near the initial dike-drift intersection, other flow modes could establish vertical propagation anywhere along the drift roof where pressure in the magma system exceeds the pressure needed to fracture the roof rock (CRWMS M&O, 2000c;

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[12]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[13]Ibid.

Woods, et al., 2001). This process could create a magma ascent path initially horizontal along some distance in the drift. Horizontal flow paths could be significantly longer than 150 m [492 ft], which is the maximum diameter of subvolcanic conduits, and thus entrain more waste packages than currently modeled in CRWMS M&O (2000a,c,d). Analyses presented in NRC (1999), Hill and Connor (2000), and Woods, et al. (2001) demonstrate that magma could ascend away from the point of initial dike-drift intersection. Although alternative flow-path models are presented in CRWMS M&O (2000c), the performance implications of these models have not been evaluated by DOE (CRWMS M&O, 2000a,d).

In summary, alternative conceptual models that are consistent with available information are not evaluated within the context of total system performance. Uncertainties with existing conceptual models are not quantified nor discussed, and the potential effects of these uncertainties are not evaluated in the Total System Performance Assessment–Site Recommendation. The staff anticipate that these alternative models will be evaluated as part of agreed-on investigations by DOE.

3.3.10.4.5 Model Abstraction Output Is Supported by Objective Comparisons

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.10.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess volcanic disruption of waste packages with respect to model abstraction output being supported by objective comparisons.

Models relevant to volcanic disruption of waste packages in CRWMS M&O (2000a–h) have not been compared with detailed process-level models, appropriate laboratory or field tests, or natural analogs. Models for the flow of magma into repository drifts (CRWMS M&O, 2000c) are critically dependent on sustaining a debris plug at the end of each intersected drift. The abstracted models used to calculate pressures in the magma-drift system will need to be supported acceptably, in conjunction with an analysis of debris-plug strength, before magma flow can be modeled as wholly restricted to within an intersected drift. Models that presume the location and geometry of subvolcanic conduits are not significantly influenced by the presence of repository drifts (CRWMS M&O, 2000c,h) also will need support through detailed process-level models. Potential inconsistencies between the abstracted models and comparative data need to be explained and quantified, and the resulting uncertainties will need to be included in total system performance assessment model results.

3.3.10.5 Status and Path Forward

The Igneous Activity Key Technical Issue Consequences Subissue relating to Volcanic Disruption of Waste Packages is considered closed-pending at the staff level. Status of subissue closure is provided in Table 1.1-3. A consolidated list of all DOE and NRC agreements relevant to the Volcanic Disruption of Waste Packages Integrated Subissue is given in Table 3.3.10-1 and Appendix A. In summary, alternative interpretations of available data have potentially significant effects on postclosure risk calculations, and current DOE risk calculations (CRWMS M&O, 2000a; Bechtel SAIC Company, LLC, 2001a) likely underestimate the risk from volcanic igneous activity. Reports revised after the August 2000 Technical Exchange on Igneous Activity (CRWMS M&O, 2000c,d,g; Bechtel SAIC Company, LLC, 2001b) have not addressed staff concerns related to the processes of magma-repository interactions. Agreements reached at the September 2001 Technical Exchange on Igneous Activity present a reasonable path forward for DOE to obtain needed data and to conduct additional analyses that would meet current acceptance criteria prior to any potential license application. The Volcanic Disruption of Waste Packages Integrated Subissue thus is considered closed-pending.

The staff have discussed the technical basis for their concerns with DOE at an Appendix 7 meeting in Las Vegas, Nevada, on May 18, 2001, and at formal technical exchanges on June 21–22, 2001, and September 5, 2001. Movement of this issue from closed-pending to closed status will require completion of the agreed-on investigations by DOE and successful review of these investigations by NRC staff. DOE may also chose to address these agreements by using consistent, reasonably conservative assumptions in deterministic analyses for volcanic disruption of waste packages.

Table 3.3.10-1. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreements*
Igneous Activity	Subissue 1—Probability of Igneous Activity	Closed-pending	IA.1.01 IA.1.02
	Subissue 2—Consequences of Igneous Activity	Closed-pending	IA.2.05 IA.2.10 IA.2.18 IA.2.19 IA.2.20
Container Life and Source Term	Subissue 2—Mechanical Disruption of Waste Packages	Closed-pending	CLST.2.10 CLST.2.19
Total System Performance Assessment and Integration	Subissue 1—System Description and Demonstration of Multiple Barriers	Closed-pending	None
	Subissue 2—Scenario Analysis and Event Probability	Closed-pending	None
	Subissue 3—Model Abstraction	Closed-pending	TSPAI.2.02
	Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards	Closed-pending	None

*Related DOE and NRC agreements are associated with one or all five generic acceptance criteria.

NOTE: Key Technical Issue Agreement GEN. 1.01 pertains to multiple integrated subissues, as well as some specific issues related to this integrated subissue

3.3.10.6 References

Baer, G. and Z. Reches. “Mechanics of Emplacement and Tectonic Implications of the Ramon Dike Systems, Israel.” *Journal of Geophysical Research*. Vol. 96. No. B7. pp. 11,895–11,910. 1991.

Bechtel SAIC Company, LLC. “FY01 Supplemental Science and Performance Analyses.” Vol. 2: Performance Analyses. TDR–MGR–PA–000001. Revision 00. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2001a.

———. “FY01 Supplemental Science and Performance Analyses.” Vol. 1: Scientific Bases and Analyses. TDR–MGR–MD–000007. Revision 00 ICN 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2001b.

Bokhove, O. and A.W. Woods. “Explosive Magma-Air Interactions by Volatile-Rich Basaltic Melts in a Dike-Drift Geometry.” San Antonio, Texas: CNWRA. 2000.

Bruce, P.M. and H.E. Huppert. “Thermal Control of Basaltic Fissure Eruptions.” *Nature*. Vol. 342. pp. 665–667. 1989.

———. “Solidification and Melting Along Dykes by the Laminar Flow of Basaltic Magma.” *Magma Transport and Storage*. M.P. Ryan, ed. Chichester, England: John Wiley & Sons. pp. 87–101. 1990.

CRWMS M&O. “Probabilistic Volcanic Hazards Analysis for Yucca Mountain, Nevada.” BA0000000–1717–2200–00082. Revision 00. North Las Vegas, Nevada: DOE, Yucca Mountain Site Characterization Office. 1996.

———. “Total System Performance Assessment for the Site Recommendation.” TDR–WIS–PA–000001. Revision 00 ICN 01. North Las Vegas, Nevada: DOE, Yucca Mountain Site Characterization Office. 2000a.

———. “Disruptive Events Process Model Report.” TDR–NBS–MD–000002. Revision 00 ICN 01. North Las Vegas, Nevada: DOE, Yucca Mountain Site Characterization Office. 2000b.

———. “Dike Propagation Near Drifts.” ANL–WIS–MD–000015. Revision 00 ICN 01. North Las Vegas, Nevada: DOE, Yucca Mountain Site Characterization Office. 2000c.

———. “Igneous Consequence Modeling for the Total System Performance Assessment–Site Recommendation.” ANL–WIS–MD–000017. Revision 00 ICN 01. North Las Vegas, Nevada: DOE, Yucca Mountain Site Characterization Office. 2000d.

———. “Characterize Eruptive Processes at Yucca Mountain, Nevada.” ANL–MGR–GS–000002. Revision 00. North Las Vegas, Nevada: DOE, Yucca Mountain Site Characterization Office. 2000e.

———. "Waste Package Behavior in Magma." CAL–EBS–ME–000002. Revision 00. North Las Vegas, Nevada: DOE, Yucca Mountain Site Characterization Office. 2000f.

———. "Number of Waste Packages Hit by Igneous Intrusion." CAL–WIS–PA–000001. Revision 01. North Las Vegas, Nevada: DOE, Yucca Mountain Site Characterization Office. 2000g.

———. "Characterize Framework for Igneous Activity at Yucca Mountain, Nevada." ANL–MGR–GS–000001. Revision 00 ICN 01. North Las Vegas, Nevada: DOE, Yucca Mountain Site Characterization Office. 2000h.

———. "Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations." TDRL–WIS–RL–000001. Revision 04. Las Vegas, Nevada: CRWMS M&O. 2000i.

Delaney, P.T. and D.D. Pollard. "Solidification of Basaltic Magma During Flow in a Dike." *American Journal of Science*. Vol. 282. pp. 856–885. 1982.

Delaney, P.T., D.D. Pollard, J.I. Ziony, and E.H. McKee. "Field Relations Between Dikes and Joints: Emplacement Processes and Paleostress Analysis." *Journal of Geophysical Research*. Vol. 91. No. B5. pp. 4,920–4,938. 1986.

DOE. "Viability Assessment of a Repository at Yucca Mountain." Overview and all five volumes. DOE/RW–0508. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 1998.

Doubik, P. and B.E. Hill. "Magmatic and Hydromagmatic Conduit Development During the 1975 Tolbachik Eruption, Kamchatka, with Implications for Hazards Assessment at Yucca Mountain, Nevada." *Journal of Volcanology and Geothermal Research*. Vol. 91. pp. 43–64. 1999.

Fedotov, S.A., A.M. Chirkov, and A.A. Razina. "Northern Breakout." *Large Tolbachik Fissure Eruption, Kamchatka 1975–1976*. S.A. Fedotov, ed. Moscow, Russia: Science. pp. 11–74. 1984.

Hill, B.E. and C.B. Connor. "Technical Basis for Resolution of the Igneous Activity Key Technical Issue." San Antonio, Texas: CNWRA. 2000.

Huppert, H.E. and R.S.J. Sparks. "Cooling and Contamination of Mafic and Ultramafic Magmas During Ascent Through Continental Crust." *Earth and Planetary Science Letters*. Vol. 74. pp. 371–386. 1985.

Macedonio, G., F. Dobran, and A. Neri. "Erosion Processes in Volcanic Conduits and Application to the AD 79 Eruption of Vesuvius." *Earth and Planetary Science Letters*. Vol. 121. pp. 137–152. 1994.

Morris, A., D.A. Ferrill, and D.B. Henderson. "Slip-Tendency Analysis and Fault Reactivation." *Geology*. Vol. 24. No. 3. pp. 275–278. 1996.

NRC. "Issue Resolution Status Report, Key Technical Issue: Igneous Activity." Revision 2. Washington, DC: NRC. 1999.

———. "Issue Resolution Status Report, Key Technical Issue: Total System Performance Assessment and Integration." Revision 3. Washington, DC: NRC. 2000.

———. NUREG–1804, "Yucca Mountain Review Plan—Draft Report for Comment." Revision 2. Washington, DC: NRC. March 2002.

Rogers, R.D. and D.K. Bird. "Fracture Propagation Associated with Dike Emplacement at the Skaergaard Intrusion, East Greenland." *Journal of Structural Geology*. Vol. 9. No. 1. pp. 71–86. 1987.

Rubin, A.M. "Tensile Fracture of Rock at High Confining Pressure: Implication for Dike Propagation." *Journal of Geophysical Research*. Vol. 98. pp. 15,919–15,935. 1993.

Thorarinsson, S., S. Steinthórsson, Th. Einarsson, H. Kristmannsdóttir, and N. Oskarsson. "The Eruption on Heimaey, Iceland." *Nature*. Vol. 241. pp. 372–375. 1973.

Valentine, G.A. and K.R. Groves. "Entrainment of Country Rock During Basaltic Eruptions of the Lucero Volcanic Field, New Mexico." *Journal of Geology*. Vol. 104. pp. 71–90. 1996.

Woods, A.W. and R.S.J. Sparks. "Report on Scoping Calculations for Magma-Repository Interaction at Yucca Mountain, Nevada." San Antonio, Texas: CNWRA. 1998.

Woods, A.W., S. Sparks, O. Bokhove, A-M. LeJeune, C.B. Connor, and B.E. Hill. "Modeling the Explosive Eruption of Basaltic Magma into the Proposed High-Level Radioactive Waste Repository at Yucca Mountain, Nevada, U.S.A." San Antonio, Texas: CNWRA. 2001.

3.3.11 Airborne Transport of Radionuclides

3.3.11.1 Description of Issue

Basaltic volcanic eruptions produce volcanic ash plumes that can transport particulate matter tens to thousands of kilometers downwind from the erupting volcano (e.g., Blackburn, et al., 1976, Walker, 1993). In the event of a volcanic eruption through the proposed repository, high-level waste may also be transported in the volcanic ash plume. Deposition of radionuclides could occur at the reasonably maximally exposed individual location, either from direct sedimentation from the volcanic ash cloud, or from the remobilization of the radionuclides and volcanic ash after initial deposition by wind or surface water. Airborne transport and deposition of radionuclides in volcanic ash plumes should be modeled to estimate the dose consequences and risk associated with these phenomena. Radionuclide transport in volcanic plumes and subsequent deposition are the topics of this integrated subissue. The inputs on probability of volcanic activity disrupting the proposed Yucca Mountain repository and the consequences of this activity for waste package integrity are covered in five integrated subissues. These integrated subissues include Biosphere Characteristics, Volcanic Disruption of Waste Packages, Mechanical Disruption of Engineered Barriers, Airborne Transport of Radionuclides, and Radionuclide Redistribution in Soil. The relationship of this integrated subissue to other integrated subissues is depicted in Figure 3.3.11-1. The overall organization and identification of all the integrated subissues are depicted in Figure 1.1-2.

This section provides a review of the abstractions of airborne transport of radionuclides incorporated by DOE in its Total System Performance Assessment. The DOE description and technical basis for the airborne transport of radionuclides abstractions are primarily documented in CRWMS M&O (2000a). Results are used and documented in CRWMS M&O (2000b–d). Portions of additional analysis and model reports were reviewed if they contained data or analyses that supported the proposed total system performance assessment abstractions (CRWMS M&O, 2000e,f).

3.3.11.2 Relationship to Key Technical Issue Subissues

The Airborne Transport of Radionuclides Integrated Subissue incorporates subject matter previously captured in the following key technical issue subissues:

- Igneous Activity: Subissue 2—Consequences of Igneous Activity (NRC, 1999)
- Total System Performance Assessment and Integration: Subissue 1—System Description and Demonstration of Multiple Barriers (NRC, 2000)
- Total System Performance Assessment and Integration: Subissue 2—Scenario Analysis and Event Probability (NRC, 2000)

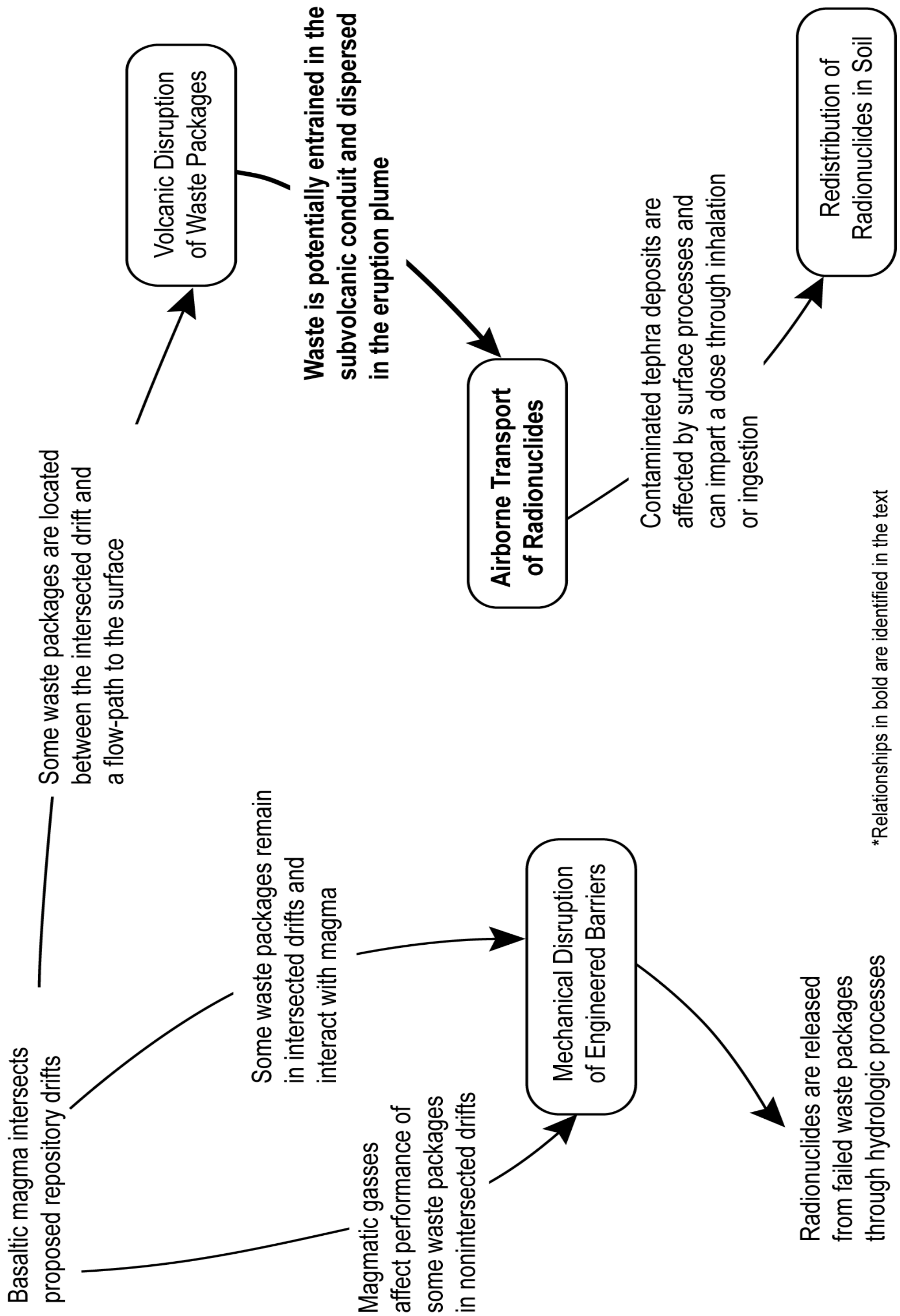

Figure 3.3.11-1. Diagram Illustrating the Relationship Between Airborne Transport of Radionuclides and Other Integrated Subissues

- Total System Performance Assessment and Integration: Subissue 3—Model Abstraction (NRC, 2000)

- Total System Performance Assessment and Integration: Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards (NRC, 2000)

The key technical issue subissues formed the basis for the previous versions of the issue resolution status reports and also were the basis for technical exchanges with DOE where agreements were reached on what additional information DOE needed to provide to resolve the subissue. The resolution status of this integrated subissue is based on the resolution status of each of the contributing key technical issues subissues. The subsequent sections incorporate applicable portions of these key technical issue subissues but no effort has been made to explicitly identify each subissue in the text.

3.3.11.3 Importance to Postclosure Performance

Eruption processes, such as diffusion and advection of tephra and radionuclides, form the primary emphasis of the Airborne Transport of Radionuclides Integrated Subissue. These processes directly affect the amount of radionuclides potentially deposited at the reasonably maximally exposed individual location by volcanic eruption through the repository. Igneous processes, partly evaluated in this integrated subissue, provide a mechanism for such rapid transport of radionuclides to a reasonably maximally exposed individual. The importance of this integrated subissue, as well as the integrated subissues of Volcanic Disruption and Mechanical Disruption of Engineered Barriers, are best documented in the DOE Total System Performance Assessment for the Site Recommendation and the Supplemental Science and Performance Analysis, Volumes 1 and 2 (CRWMS M&O, 2000h; Bechtel SAIC Company, LLC, 2001a,b). As is stated in Section 5.3 of Volume 2 of the Supplement Science and Performance Analysis (Bechtel SAIC Company, LLC, 2001b), "For the TSPA–SR [Total System Performance Assessment for the Site Recommendation] and the supplemental TSPA [Total System Performance Assessment] model, probability-weighted mean annual dose from igneous disruption determine the magnitude of the overall mean annual dose from nominal and disruptive performance during the first 10,000 years."

3.3.11.4 Technical Basis

NRC developed a plan (2002) consistent with the acceptance criteria and review methods found in previous issue resolution status reports. A review of the DOE approach for including airborne transport of radionuclides in total system performance assessment abstractions is provided in the following subsections. The review is organized according to the five acceptance criteria identified in Section 1.5 as follows: (i) System Description and Model Integration Are Adequate, (ii) Data Are Sufficient for Model Justification, (iii) Data Uncertainty Is Characterized and Propagated Through the Model Abstraction, (iv) Model Uncertainty Is Characterized and Propagated Through the Model Abstraction, and (v) Model Abstraction Output Is Supported by Objective Comparisons.

3.3.11.4.1 System Description and Model Integration Are Adequate

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.11.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess airborne transport of radionuclides with respect to system description and model integration.

Basaltic volcanic eruptions produce volcanic ash plumes that transport particulate matter tens to thousands of kilometers downwind from the erupting volcano. In the event of a volcanic eruption through the proposed repository, high-level waste may also be transported in the volcanic ash plume, with the potential deposition of radionuclides at the reasonably maximally exposed individual location, either from direct sedimentation from the volcanic ash cloud or from the remobilization of the radionuclides and volcanic ash after initial deposition by wind or surface water. Airborne transport and deposition of radionuclides in volcanic ash plumes must be modeled to estimate the dose consequences and risks associated with these phenomena.

ASHPLUME uses the Suzuki (1983) model to abstract the thermo-fluid dynamics of ash dispersion in the atmosphere,

$$X(x,y) = \int_{\varphi_{min}}^{\varphi_{max}} \int_{0}^{H} \frac{5 f_z(z) f_\Phi(\varphi) Q}{8\pi C (t+t_s)^{5/2}} \exp\left\{ - \frac{5\left[(x-ut)^2 + y^2\right]}{8C(t+t_s)^{5/2}} \right\} dz d\varphi \qquad (3.3.11\text{-}1)$$

where X is the mass of ash and radionuclides accumulated at geographic location $x, y,$ relative to the position of the volcanic vent; $f_Z(z)$ is a probability density function for diffusion of particles out of the eruption column, treated as a line source extending vertically from the vent to total column height, H; $f_\Phi(\varphi)$ is a probability density function for grain size, φ; Q is the total mass of material erupted; u is wind speed in the x-direction; t is the particle fall-time through the atmosphere; t_s is diffusion time of tephra and high-level-waste-laden tephra; and C is eddy diffusivity. Most of these parameters, in turn, depend on additional parameters that are estimated as part of performance assessments (Jarzemba, 1997; CRWMS M&O, 2000a,c; Connor, et al., 2001).

In ASHPLUME, the erupting column is treated as a line source reaching some maximum height governed by the energy and mass of the eruption. A linear decrease in the upward velocity of particles is assumed, resulting in segregation of ash or ash and waste particles in the ascending column by settling velocity, which is a function of grain size, shape, and density. Tephra and high-level waste particles are removed from the column based on their settling velocity, the decrease in upward velocity of the column as a function of height, and a probability density function [$f_z(z)$] that attempts to capture particle diffusion out of the column. These relationships are valid for particles larger than 15 μm [0.0006 in] in diameter, but do not capture the atmospheric dynamics of settling for smaller particle diameters (Suzuki, 1983). Dispersion of the tephra and high-level waste diffused out of the column is modeled for a uniform wind field and is governed by the diffusion-advection equation with vertical settling. Thus, results derived using this model depend heavily on assumptions about the shapes of the distributions $f_Z(z)$ and $f_\Phi(\varphi)$.

In CRWMS M&O (2000g), DOE demonstrated that the ASHPLUME code, as implemented by DOE, can reasonably represent an actual basaltic volcanic eruption. In addition, this document provides the parameters used in the analysis. In CRWMS M&O (2000c), DOE provided the cumulative distribution functions for both the mean ash particle diameter used in its models and the ash-dispersion controlling constant. These values appear reasonable and, therefore, NRC considers that DOE has the means to satisfactorily address this acceptance criterion.

3.3.11.4.2 Data Are Sufficient for Model Justification

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.11.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess airborne transport of radionuclides with respect to data being sufficient for model justification.

The ASHPLUME model itself was first developed for use in the high-level waste program by Jarzemba, et al. (1997) and later modified by DOE. Most of the parameters, with the notable exception of parameters related to the transport of high-level waste, used as input to ASHPLUME are derived from the volcanological literature (CRWMS M&O, 2000a,c). Because many of the volcanic processes important for consequence evaluation are not preserved in the Yucca Mountain region geologic record, proposed process-level consequence models should be verified with data from reasonably analogous small-volume basaltic volcanic systems to be acceptable. In CRWMS M&O (2000a), analogous eruptions, including but not limited to the 1975 Tolbachik, Russia; 1943–52 Parícutin, Mexico; and 1850–1999 Cerro Negro, Nicaragua, and violent strombolian eruptions are cited as the sources of acceptable parameter distributions for use in ASHPLUME. Staff agree these data and the volcanological processes evinced by these eruptions are reasonable analogs for potential volcanic eruptions in the Yucca Mountain region and ASHPLUME inputs.

Issues related to data sufficiency and model justification in the Airborne Transport of Radionuclides Integrated Subissue involve three topics: (i) the range of eruption energetics used by DOE in the ASHPLUME simulations, (ii) the method of incorporation of high-level waste into erupting tephra, and (iii) the use of a uniform windfield in ASHPLUME simulations of tephra and high-level waste dispersion using data derived from near-surface meteorological observations at the site. Each of these three topics is addressed in this section.

There has been extensive concurrent work on the nature of violent strombolian eruptions and application of numerical models of tephra dispersion in hazard assessments, simultaneous with the development of ASHPLUME (e.g., Woods, 1995; Sparks, et al., 1997; Hill, et al., 1998; Rosi, 1998; Connor, et al., 2001). The greatest relevance of this work is in bounding the energetics of potential future volcanic eruptions in the Yucca Mountain region. ASHPLUME Version 1.3 uses eruption power, volume, and conduit diameter [directly related to muzzle velocity at the vent (Wilson and Head, 1981)] to characterize the eruption. These parameters bound eruption energetics and are used to estimate steady-state eruption duration and column height, assuming that eruption column height, H (kilometers); eruption volume, V (cubic meter, dense rock equivalent); and duration of the violent strombolian phase of the eruption, T (seconds), are related by

$$\frac{\mathrm{d}V}{\mathrm{d}t} = \left(\frac{H}{1.67} \right)^{4} \qquad (3.3.11\text{-}2)$$

and

$$V = \frac{\mathrm{d}V}{\mathrm{d}t} T \qquad (3.3.11\text{-}3)$$

These relationships provide a check on input parameters. It is crucial for DOE to track also the mass flow rate together with the muzzle velocity at the vent for simulated eruptions in ASHPLUME to ensure that all eruptions used in the simulations have simple-to-super-buoyant plumes, as expected for the violent strombolian phase of cone-building eruptions (Woods and Bursik, 1991). Verification is needed in the model that mass flow and vent velocity regimes are sufficient to maintain such columns for all ASHPLUME simulations. Currently, it appears that some modeled events have mass flow rates and vent velocities that are too low to sustain such plumes (CRWMS M&O, 2000c). DOE has agreed to model the interaction of magma with the repository.[1] This model will provide a better understanding of flow velocities, mass rates, and other important properties of the eruption, which will be used to constrain the eruptive characteristics and, therefore, better justify the input parameters into ASHPLUME. As such, NRC has no questions related to this concern at this time.

CRWMS M&O (2000a) notes that the most difficult aspect of the ASHPLUME model abstraction involves quantifying high-level waste transport. Currently, the fuel fraction model developed by Jarzemba, et al. (1997) is used to abstract the complex process of high-level waste incorporation and transport. Waste particles are assumed to be incorporated into erupting pyroclasts following the rule

$$\rho_c = \log\left(\frac{d^{a}_{\mathrm{min}}}{d^{f}} \right) \qquad (3.3.11\text{-}4)$$

where d^f is the diameter of the waste particle to be incorporated and d^{a}_{min} is the minimum diameter of a pyroclast required to transport this particle. Motivation for this approach, detailed in Jarzemba, et al. (1997), was to bound the particle size and density distribution for estimating the dispersion of contaminated waste. Jarzemba, et al. (1997) arbitrarily chose a value of ρ_c = 0.3 to illustrate the application of the model. The assumption that ρ_c = 0.3 is propagated through the Total System Performance Assessment–Site Recommendation (CRWMS M&O, 2000c). That Jarzemba, et al. (1997) made this assumption about the incorporation ratio, as an

[1]Reamer,C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

example, is not a sufficient basis for DOE to make this assumption in a license application. Additional documentation will be required to justify assumptions about the incorporation of high-level waste. DOE agreed to describe the method of high-level waste incorporation used in the DOE models.[2]

Wind speed is a parameter that significantly affects tephra dispersion models for basaltic volcanoes (e.g., Hill, et al., 1998). The column from the next Yucca Mountain region eruption will likely reach altitudes of 2–6 km [1–4 mi] above ground level, as is observed for most violent-strombolian basaltic eruptions. Although near-ground-surface wind data are available for the proposed repository site, low-altitude winds will be affected significantly by surface topographic effects and, thus, have little relevance to modeling dispersal from 2–6-km [1–4-mi]-high eruption columns. For Total System Performance Assessment–Site Recommendation analyses, DOE used wind speeds and directions obtained from near-surface stations (CRWMS M&O, 2000a,c). It is much more appropriate to use data sets that extend to higher altitudes (e.g., data available from the Desert Rock Airstrip, Nevada) and to model the effects of stratified wind velocities and directions for eruptions (e.g., Glaze and Self, 1991). A stratified windfield is incorporated into ASHPLUME by specifying variation in the windfield as a function of height. A starting height, z_k, and windspeed and direction, u_k, are associated with each k stratum, within which wind speed and direction are held constant. With a windfield that varies with height, the site of particle deposition is controlled by the release height of the particle from the eruption column and the average windspeed and direction encountered during particle settling through the atmosphere. This average wind vector can be calculated using

$$u_{\text{avg}} = \frac{1}{Z} \sum_{k=0}^{N_k} u_k \Delta z_k \qquad (3.3.11\text{-}5)$$

where Z is the height above the ground from which the particle is released; N_k is the number of wind strata between Z and the ground; Δz_k is the thickness of the wind stratum, within which the windfield is assumed to be uniform; u_k is the wind vector in stratum k; and u_{avg} is the average resulting wind vector for particles released at height Z. This average wind vector for a specific height above the ground is independent of particle size. Therefore, the average wind vector experienced by all particles released from the eruption column at height Z need only be calculated once for a given eruption realization. DOE agreed to evaluate the wind speed data appropriate for the height of the eruptive columns being modeled.[3]

Staff conclude that the current version of ASHPLUME will be greatly improved (more realistic) if these three changes are incorporated. The resulting model will more accurately reflect outcomes of volcanic eruptions through the proposed repository. Furthermore, each of these

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[2]Reamer,C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[3]Schlueter, J. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (August 29–31, 2000)." Letter (October 23) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

changes could significantly affect estimates of dose and risk at the receptor location. Of course, their impact on risk cannot be evaluated until the changes are included in the model.

In summary, DOE agreed to provide the additional information necessary for model justification. For example, the agreements to model repository/magma interactions will result in the use of appropriate wind speeds for the height of the columns being modeled. The evaluation of the incorporation of high-level waste into the magma results also will result in a better justification of the methodology.

3.3.11.4.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.11.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess airborne transport of radionuclides with respect to data uncertainty being characterized and propagated through the model abstraction.

Parameter distributions for inputs into ASHPLUME are discussed in CRWMS M&O (2000a) and presented in detail in CRWMS M&O (2000c, Table 5). Most of these parameter distributions are well-documented and supported and, therefore, are not discussed further. In addition to the parameter distributions discussed in Section 3.3.11.4.2 (i.e., wind speed and direction, eruption velocity, and conduit diameter), the distribution function for distribution of tephra and high-level waste in the vertical eruption column, β, requires further attention.

In the ASHPLUME model, tephra is released from the eruption column for advective transport downwind at a height depending on grain size, total column height, and the parameter β. Essentially, a small value of β (e.g., 0.1) will result in a tendency for particles to be released low in the eruption column, with only very fine grained material reaching the top of the column. A large value of β (e.g., 1) results in most of the tephra reaching the top of the column. Large values of β (e.g., 10) result in a point source of tephra at height H in the atmosphere. Because particle advection downwind is strongly dependent on the height in the eruption column at which particles are released, β potentially has a strong influence on dose. In CRWMS M&O (2000c), β is limited to a range of 0.01 to 0.5, or a range that limits the ascent of particles, particularly large high-level waste bearing particles, in the tephra column. Hill, et al. (1998), however, found that $\beta = 10$ best fits the observed distribution of tephra at 20 km [12 mi] from the vent, using data from the 1995 Cerro Negro eruption. Further, in CRWMS M&O (2000g), a value of $\beta = 10$ was used by DOE to demonstrate that the ASHPLUME code can reasonably replicate a natural eruption (i.e., the 1995 Cerro Negro eruption).

In summary, CRWMS M&O (2000c) is one of the many reports that is scheduled for revision by the end of 2003 and DOE agreed that the discrepancies between CRWMS M&O (2000c) and CRWMS M&O (2000g) will be addressed at that time.

3.3.11.4.4 Model Uncertainty Is Characterized and Propagated through the Model Abstraction

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.11.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess airborne transport of radionuclides with respect to model uncertainty being characterized and propagated through the model abstraction.

DOE notes that there are uncertainties in the use of the ASHPLUME model, and this model cannot be used to capture the total range of eruption conditions that may occur in the Yucca Mountain region (CRWMS M&O, 2000a). This is correct: ASHPLUME can only model the violent strombolian phases of future Yucca Mountain region basaltic volcanic eruptions. One way to approach this limitation is to assume that only the violent strombolian phase of a cone-building eruption will result in a significant dose to the reasonably maximally exposed individual. This assumption is the current approach, and eruption durations are shortened appropriately (CRWMS M&O, 2000a).

Alternative models, such as PUFF and the Gas-Thrust models (CRWMS M&O, 2000c), are currently not implemented. This is a potential shortcoming in three respects. First, the input parameters most easily gleaned from the volcanological literature (e.g., initial volatile content and magma density) (CRWMS M&O, 2000a) are not directly input into ASHPLUME because it is not a physical abstraction; rather, ASHPLUME is empirical. This limitation means it is not possible to evaluate the effects of variation of some physical parameters (e.g., initial volatile content) directly to expected dose to the reasonably maximally exposed individual. As DOE has demonstrated that the ASHPLUME code can reasonable replicate analog eruptions (CRWMS M&O, 2000g), this concern has been generally alleviated. Second, because ASHPLUME is an empirical model, it is difficult to gain confidence in the manner that ASHPLUME treats high-level waste dispersion (CRWMS M&O, 2000a). Although it may be possible for DOE to bound this model uncertainty with sensitivity analyses, this has not yet been reported, although DOE agreed to conduct some sensitivity studies.[4] Third, there is potential that the repository engineered system may have substantial impact on the near-surface flow of magma. Magma flow through drifts, for example, may substantially change the mass flow and eruption velocity, resulting in altered airborne transport of high-level waste. The current version of ASHPLUME cannot account for these physical processes. DOE agreed to evaluate how the repository itself may modify flow conditions and, therefore, the eruptive characteristics.[5] Depending on the results of this analysis, it may be necessary to reevaluate, and possibly modify, the ASHPLUME code to account for these changes in physical processes.

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[4]Schlueter, J. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (August 29–31, 2000).” Letter (October 23) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[5]Reamer,C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001).” Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

The staff note that DOE conceptually evaluated the PUFF code based on descriptions in the scientific literature, but could not obtain a working version of the code from its originators. DOE concluded, however, that the code was not designed to model atmospheric transport and settling of waste and ash and, therefore, is not appropriate for current programmatic needs. (CRWMS M&O, 2000c).

The Suzuki (1983) model does not attempt to quantify the thermo-fluid dynamics of volcanic eruptions. The more recent class of models, pioneered by Woods (1988), concentrates on the bulk thermophysical properties of the column, defining a gas-thrust region near the vent and a convective region above, within which the thermal contrast between the atmosphere and the rising column results in the entrainment of air and buoyancy forces that loft particles upward. In contrast to Suzuki (1983), this class of models results in a highly nonlinear velocity profile within the ascending column. This difference can have a profound effect on the ascent height of high-level waste particles in an ascending eruption column and ensuing dispersion in the accessible environment (Hill and Connor, 2000). DOE considered the Gas-Thrust model, but concluded that the parameter β has a similar effect (CWRMS M&O, 2000c). If DOE continues to use a value of β similar to that used in its demonstration that the ASHPLUME code can replicate natural eruptions (CWRMS M&O, 2000g), this concern is generally alleviated.

Less energetic stages of a cinder-cone-forming eruption produce weak plumes that bend over as they rise because of wind advection. Sparks, et al. (1997) note that these weak plumes can remain highly organized as they are advected downwind. Such plumes can form convection cells or retain a puffy character with little entrainment and mixing with air. Thus, sedimentation out of these plumes may be slower than expected using the diffusion-advection equation. For example, although the 1995 eruption of Cerro Negro produced a relatively small volume of tephra {3×10^6 m^3 [1×10^8 ft^3]} in a column that rose to only 2–2.5 km [1.2–1.5 mi], ash-fall deposits 20 km [12 mi] downwind were 0.5 cm [0.2 in] (Hill, et al., 1998). Eruptions of this magnitude are capable of effecting peak annual total effective dose equivalents for individuals located 20 km [12 mi] from a repository-penetrating volcanic eruption (Hill and Connor, 2000). Clearly, realistic consequence analyses will be needed to evaluate dose from large, convective eruptions that ascend to atmospheric levels of neutral buoyancy as well as smaller eruptions with column ascent limited by prevailing winds. Finally, changes in the physics of the eruption caused by the development of complex near-surface magma flow in the repository can be incorporated in total system performance assessment.

In summary, both DOE and NRC can demonstrate that the ASHPLUME code, as implemented, can reasonably replicate a natural analog eruption (Hill et al., 1998, and CRWMS M&O, 2000g). It is recognized, however, that the changes in physics of an eruption, because of the interactions with the repository, may necessitate modifications to the code. This can not be determined until the analyses, being conducted under the Volcanic Disruption of the Waste Package Integrated Subissue have been completed.[6] Also, the basis for the incorporation ratio is the observation of incorporation of xenoliths in natural flows and eruptions; however, further

[6]Reamer,C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

work is needed by both DOE and NRC to evaluate if the incorporation ratio can be justified, and if not, which alternative method should be used as a substitute.[7] The accuracy of the air transport models, however, may not be that significant in evaluating total risk. The air and water transport of the ash and waste particles from the area of deposition to the area of the reasonably maximally exposed individual, with subsequent exposure of the reasonably maximally exposed individual, may overshadow the effect of uncertainty in the air transport during the eruption. Ash redistribution is being evaluated in the Radionuclide Redistribution in Soil Integrated Subissue.[8] Therefore, to get a reasonably accurate evaluation of the risk from a volcanic eruption, work for these three integrated subissues needs to be integrated and correlated. There are agreements in place in all three integrated subissues to cover these concerns as they relate to model uncertainty.

3.3.11.4.5 Model Abstraction Output Is Supported by Objective Comparisons

Overall, the current information, along with the agreements reached between DOE and NRC (Section 3.3.11.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess airborne transport of radionuclides with respect to model abstraction output being supported by objective comparisons.

Verification of ASHPLUME was provided, in part, by Hill, et al. (1998) in their analysis of the 1995 eruption of the Cerro Negro volcano in Nicaragua. DOE has performed a similar analysis. As demonstrated in Figure 6 of CRWMS M&O (2000g), the ASHPLUME code, as implemented by DOE, can also reasonable replicate the 1995 Cerro Negro eruption. NRC, therefore, considers this concern closed (Igneous Activity Agreement 2.04). In addition, DOE considers Cerro Negro as an analog for the eruption that could occur at the Yucca Mountain site and will document this in a revision to CRWMS M&O (2000a) (Igneous Activity Agreement 2.04).[9]

The questions remaining about the use of the ASHPLUME model are related to the incorporation and transport of high-level waste in the eruption column and dispersal in the volcanic plume. Uncertainty in this parameter distribution results from the lack of natural analogy in the geologic record. Basaltic eruptions that build cinder cones show dramatic variations in energy, duration, and style. Numerical models that quantify the physics of these eruptions have reached a stage of development that allows exploration of the parameters governing these variations. Thus, many of the nuances of observed eruption columns and their deposits can now be understood in terms of fundamental physical processes (e.g., Sparks, et al., 1997). Such an understanding is important for volcanic risk assessment related to the proposed Yucca Mountain repository because there are no observations

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[7]Reamer,C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001).” Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[8]Ibid.

[9]Schlueter, J. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (August 29–31, 2000).” Letter (October 23) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

analogous to the behavior of dense high-level waste particles in eruption columns, and no appropriate analogs have been identified. There also is considerable uncertainty in how to simulate the entrainment and dispersal of high-level waste in eruption columns. Physically accurate eruption column models provide an opportunity to extend our understanding of tephra plumes to encompass the distribution and deposition of dense high-level waste particles in tephra deposits. In these circumstances, application of physically accurate models is a fundamental step in estimating risk. DOE will need to present an acceptable level of analysis that captures essential details of volcanic ash-plume dispersion and the expected dose resulting from transport of high-level waste in volcanic ash plumes. DOE recognizes this concern and has agreed to describe the methodology it will be using in its models to account for waste incorporation, including possible particle aggregation.[10]

In summary, DOE has acceptably demonstrated that the ASHPLUME code, as implemented by DOE, can reasonable replicate a natural basaltic volcanic eruption, and has agreed to provide the necessary information on high-level waste incorporation to demonstrate that the code has a sound technical basis. It is recognized that there is no natural volcanic analog that can be used to demonstrate that this part of the model abstraction is supported by objective comparisons; therefore, accurate modeling of the physical process will be necessary.

3.3.11.5 Status and Path Forward

Table 3.3.11-1 provides the status of all key technical issue subisses, referenced in Section 3.3.11.2 for the Airborne Transport of Radionuclides Integrated Subissue. The table also provides the related DOE and NRC agreements pertaining to the Airborne Transport of Radionuclides Integrated Subissue. The agreements listed in the table are associated with one or all five generic acceptance criteria discussed in Section 3.3.11.4. Note that the status and the detailed agreements (or path forward) pertaining to all the key technical issue subissues are provided in Table 1.1-2 and Appendix A.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (through specified testing, analysis, and the like), acceptably addresses the NRC questions so that no information beyond that provided, or agreed to, will likely be required at the time of a potential license application.

[10]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001).” Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

Table 3.3.11-1. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreements*
Igneous Activity	Subissue 2—Consequences of Igneous Activity	Closed-pending	IA.2.01 IA.2.02 IA.2.03 IA.2.04 IA.2.09 IA.2.20
Total System Performance Assessment and Integration	Subissue 1—System Description and Demonstration of Multiple Barriers	Closed-pending	None
	Subissue 2—Scenario Analysis and Event Probability	Closed-pending	TSPAI.2.02
	Subissue 3—Model Abstraction	Closed-pending	None
	Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards	Closed-pending	None

*Related DOE and NRC agreements are associated with one or all five generic acceptance criteria.

3.3.11.6 References

Bechtel SAIC Company, LLC. "FY01 Supplemental Science and Performance Analyses." Vol. 1: Scientific Bases and Analyses. TDR–MGR–MD–000007. Revision 00 ICN 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2001a.

———. "FY01 Supplemental Science and Performance Analyses." Vol. 2: Performance Analyses. TDR–MGR–PA–000001. Revision 00. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2001b.

Blackburn, E.A., L. Wilson, and R.S.J. Sparks. "Mechanisms and Dynamics of Strombolian Activity." *Journal of the Geological Society of London*. Vol. 132. pp. 429–440. 1976.

Carey, S.N. "Modeling the Tephra Fallout from Explosive Volcanic Eruptions." *Monitoring and Mitigation of Volcanic Hazards*. R. Scarpa and R. Tilling, eds. Berlin, Germany: Springer. pp. 429–462. 1996.

CRWMS M&O. "Characterize Eruptive Processes at Yucca Mountain, Nevada." ANL–MGR–GS–000002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000a.

———. "Features, Events, and Processes: Screening for Disruptive Events." ANL–WIS–MD–000005. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000b.

———. “Igneous Consequence Modeling for Total System Performance Assessment–Site Recommendation.” ANL–WIS–MD–000017. Revision 00 ICN 01 Las Vegas, Nevada: CRWMS M&O. 2000c.

———. “Characterize Framework for Igneous Activity.” ANL–MGR–GS–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000d.

———. “Dike Propagation Near Drifts.” ANL–WIS–MD–000015. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000e.

———. “Miscellaneous Waste-Form Features, Events, and Processes.” ANL–WIS–MD–000009. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000f.

———. “Comparison of ASHPLUME Model Results to Representative Tephra Fall Deposits.” CAL–WIS–MD–000011. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000g.

———. “Total System Performance Assessment (TSPA) for the Site Recommendation.” TDR–WIS–PA–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000h.

Connor, C.B., J. Stamatakos, D. Ferrill, B. Hill, G. Ofoegbu, F.M. Conway, B. Sagar, and J. Trapp. “Geologic Factors Controlling Patterns of Small Volume Basaltic Volcanism: Application to a Volcanic Hazards Assessment at Yucca Mountain, Nevada.” *Journal of Geophysical Research*. Vol. 105. pp. 417–432. 2000.

Connor, C.B., B.E. Hill, B. Winfrey, N.M. Franklin, and P.C. La Femina. “Estimation of Volcanic Hazards from Tephra Fallout.” *Natural Hazards Review*. Vol. 2. No. 1. pp. 33–42. 2001.

Glaze, L. and S. Self. “Ashfall Dispersal for the 16 September 1986 Eruption of Lascar, Chile, Calculated by a Turbulent Diffusion Model.” *Geophysical Research Letters*. Vol. 18. pp. 1,237–1,240. 1991.

Hill, B.E. and C.B. Connor. “Technical Basis for Resolution of the Igneous Activity Key Technical Issue.” San Antonio, Texas: CNWRA. 2000.

Hill, B.E., C.B. Connor, M.S. Jarzemba, P.C. La Femina, M. Navarro, and W. Strauch. “1995 Eruptions of Cerro Negro Volcano, Nicaragua, and Risk Assessment for Future Eruptions.” *Geological Society of America Bulletin*. Vol. 110. pp. 1,231–1,241. 1998.

Jarzemba, M.S. “Stochastic Radionuclide Distributions After a Basaltic Eruption for Performance Assessments at Yucca Mountain.” *Nuclear Technology*. Vol. 118. pp. 132–141. 1997.

Jarzemba, M.S., P.A. LaPlante, and K.J. Poor. “ASHPLUME Version 1.0—A Code for Contaminated Ash Dispersal and Deposition.” CNWRA 97-004. San Antonio, Texas: CNWRA. 1997.

NRC. "Issue Resolution Status Report, Key Technical Issue: Igneous Activity." Revision 2. Washington, DC: NRC. 1999.

———. "Issue Resolution Status Report, Key Technical Issue: Total System Performance Assessment and Integration." Revision 3. Washington, DC: NRC. 2000.

———. NUREG–1804, "Yucca Mountain Review Plan—Draft Report for Comment." Revision 2. Washington, DC: NRC. March 2002.

Rosi, M. "Plinian Eruption Columns: Particle Transport and Fallout." *From Magma to Tephra*. A. Freundt and M. Rosi, eds. Amsterdam, Holland: Elsevier. pp. 139–172. 1998.

Sparks, R.S.J., M.I. Bursik, S.N. Carey, J.S. Gilbert, L.S. Glaze, H. Sigurdsson, and A.W. Woods. *Volcanic Plumes*. New York City, New York: John Wiley and Sons. 1997.

Suzuki, T. "A Theoretical Model for the Dispersion of Tephra." *Arc Volcanism: Physics and Tectonics*. D. Shimozuru and I. Yokiyama, eds. Tokyo, Japan: Terra Scientific Publishing Company. pp. 95–113. 1983.

Walker, G.P.L. "Basaltic-Volcano Systems, Magmatic Processes and Late Tectonics." *Geological Society Special Publications 76*. H.M. Prichard, T. Alabaster, N.B.W. Harris, and C.R. Neary, eds. London, United Kingdom: Geological Society. pp. 3–38. 1993.

Wilson, L. and J. Head. "Ascent and Eruption of Magma on the Earth and Moon." *Journal of Geophysical Research*. Vol. 86. pp. 2,971–3,001. 1981.

Woods, A.W. "The Dynamics of Explosive Volcanic Eruptions." *Reviews in Geophysics*. Vol. 33. pp. 495–530. 1995.

———. "The Dynamics and Thermodynamics of Eruption Columns." *Bulletin of Volcanology*. Vol. 50. pp. 169–191. 1988.

Woods, A.W. and M.I. Bursik. "Particle Fallout, Thermal Disequilibrium, and Volcanic Plumes." *Bulletin of Volcanology*. Vol. 53. pp. 559–570. 1991.

3.3.12 Representative Volume

3.3.12.1 Description of Issue

The Representative Volume Subissue addresses the effects of well pumping on the radionuclide concentrations in the extracted groundwater at the receptor location. Relationship of this integrated subissue to other integrated subissues is depicted in Figure 3.3.12-1. The overall organization and identification of all the integrated subissues are depicted in Figure 1.1-2. DOE description and technical bases for abstraction of dilution of radionuclides in groundwater due to well pumping are documented in CRWMS M&O (2000a,b). This section provides a review of the abstractions of dilution of radionuclides in groundwater due to well pumping that DOE incorporated in its total system performance assessment.

3.3.12.2 Relationship to Key Technical Issue Subissues

The Representative Volume Integrated Subissue incorporates subject matter previously captured in the following key technical issue subissues:

- Unsaturated and Saturated Flow Under Isothermal Conditions: Subissue 5—Saturated Zone Flow and Dilution Processes (NRC, 1999)

- Total System Performance Assessment and Integration: Subissue 1—System Description and Demonstration of Multiple Barriers (NRC, 2000)

- Total System Performance Assessment and Integration: Subissue 2—Scenario Analysis and Event Probability (NRC, 2000)

- Total System Performance Assessment and Integration: Subissue 3—Model Abstraction (NRC, 2000)

- Total System Performance Assessment and Integration: Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards (NRC, 2000)

The subissues of the key technical issue formed the bases for the previous versions of the issue resolution status reports and also were the bases for technical exchanges with DOE, where agreements were reached on what additional information DOE needed to provide to resolve the subissue. The resolution status of this integrated subissue is based on the resolution status of each of the contributing key technical issue subissues. The subsequent sections incorporate applicable portions of these key technical issue subissues, however, no effort was made to explicitly identify each subissue.

3.3.12.3 Importance to Postclosure Performance

One aspect of risk-informing the NRC review was to determine how this integrated subissue is related to the DOE repository safety strategy. In the postclosure section of

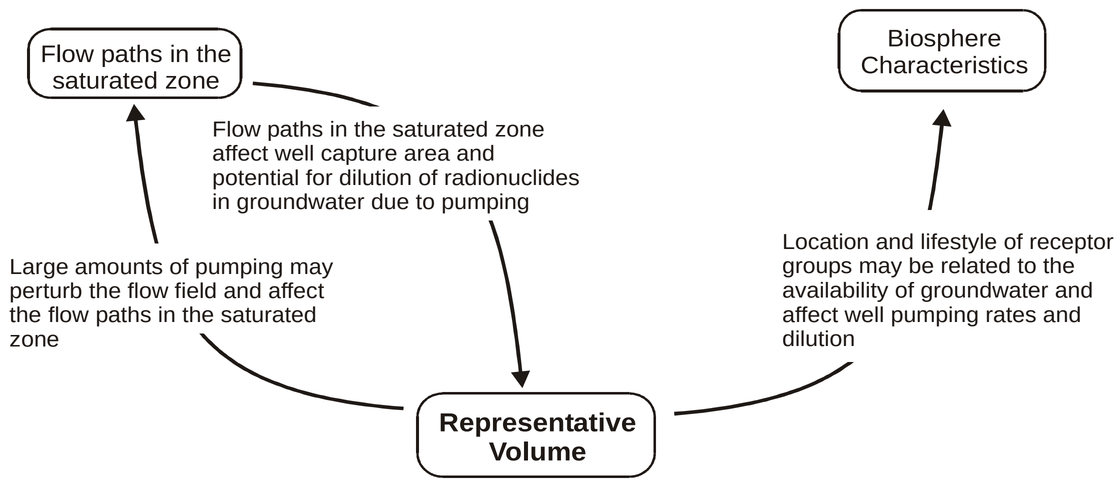

Figure 3.3.12-1. Diagram Illustrating the Relationship Between Representative Volume and Other Integrated Subissues

CRWMS M&O (2000c, Section 4.2.8), DOE concludes that its performance estimates were not very sensitive to dilution of radionuclides in groundwater due to well pumping. Based on that assessment, DOE did not consider dilution of radionuclides in groundwater due to well pumping to be a principal factor in its postclosure safety case. In the total system performance assessment model for site recommendation that DOE adopted, however, it is assumed that all radionuclides crossing the compliance boundary will be captured by pumping wells and diluted into the volume pumped (i.e., dilution volume). The sensitivity analyses DOE conducted (CRWMS M&O, 2000d, Figure 5.2-16) indicate that the calculated dose is directly affected by the pumping volume, and that increases or decreases in the pumping volume produce a proportional reduction or increase, respectively, in the calculated dose. Therefore, based on the approach DOE adopted in the total system performance assessment for site recommendation, NRC staff consider the dilution of radionuclides due to well pumping, including the pumping volume by the group containing the reasonably maximally exposed individual, important to the calculated dose.

3.3.12.4 Technical Basis

NRC developed a Yucca Mountain Review Plan (NRC, 2002) that is consistent with the acceptance criteria and review methods found in previous issue resolution status reports. A review of DOE approaches for including dilution of radionuclides in groundwater due to well pumping in total system performance assessment abstractions is provided in the following subsections. The review is organized according to the five acceptance criteria identified in Section 1.5: (i) System Description and Model Integration Are Adequate, (ii) Data Are Sufficient for Model Justification, (iii) Data Uncertainty Is Characterized and Propagated Through the Model Abstraction, (iv) Model Uncertainty Is Characterized and Propagated Through the Model Abstraction, and (v) Model Abstraction Output Is Supported by Objective Comparisons.

3.3.12.4.1 System Description and Model Integration Are Adequate

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.12.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess dilution of radionuclides in groundwater due to well pumping with respect to system description and model integration.

DOE treats dilution of radionuclides in groundwater due to well pumping as an included feature, event, and process in CRWMS M&O (2001, Subsection 6.2.23). To assess dilution of radionuclides in groundwater due to well pumping, DOE assumes the future population in the Yucca Mountain area is represented by a farming community located in the Amargosa Valley region, at and beyond the compliance boundary (CRWMS M&O, 2000b, Section 6.2.4). Radionuclide concentrations in the pumped groundwater are determined by dividing the radionuclide mass delivered to the biosphere per year (i.e., the radionuclide mass arriving at the compliance boundary assuming complete capture) by the groundwater volume extracted per year to meet water demand of the farming community (CRWMS M&O, 2000a, Section 3.6.3.3.4).

DOE assumes that all the radionuclide mass reaching the compliance boundary will be captured by the pumping wells, and the radionuclide mass is distributed uniformly in the total volume of groundwater used by the farming community. Although it is reasonable to expect variations in radionuclide concentrations among spatially distributed pumping wells in the community, redistribution of radionuclides along multiple pathways in the biosphere would lead to homogenization of dose to the reasonably maximally exposed individual. This assumption implies considerable sharing of produce and resources within the farming community.

The DOE approach is consistent with the provisions for disposal of high-level waste for Yucca Mountain (10 CFR Part 63). One of the criteria provided in the regulations to characterize the reasonably maximally exposed individual is that the individual uses well water with *average* concentrations of radionuclides [10 CFR 63.312(c)]. Furthermore, the water demand of the farming community is also *specified* in the regulations at 3.7×10^6 m^3 [3,000 acre-ft] [10 CFR 63.312(c)].

In summary, available information for the saturated zone, from the saturated zone process model report and supporting analysis and model reports, is sufficient to (i) characterize the dilution of radionuclides in groundwater due to well pumping and (ii) abstract dilution of radionuclides in groundwater due to well pumping in total system performance assessment analyses.

3.3.12.4.2 Data Are Sufficient for Model Justification

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.12.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess dilution of radionuclides in groundwater due to well pumping with respect to data being sufficient for model justification.

Because complete radionuclide mass capture at the compliance boundary is assumed, data to describe the spatial distribution of mass transport in the saturated zone are not used in the model abstraction for dilution by well pumping. Estimates of future groundwater pumping rates are based on a combination of data from a 1997 survey of groundwater pumping in Nye County, Nevada (State of Nevada, 1997), and 1990 census data (U.S. Census Bureau, 1999). These data were used to estimate a range of present-day, per-farm pumping rates. DOE chose the size of the hypothetical farming community assumed for the future to be reasonably consistent with 64 FR 8640, which indicates that the future farming community should be considered to contain approximately 100 people living on 15–25 farms. DOE interpreted 64 FR 8640 to mean consideration of either a farming community inhabited by 100 people or a farming community composed of 15–25 farms.

Notwithstanding that there may be variations in the calculated dose among different individual members of the farming community, the individual protection standard is based on the average water use. To the extent that the total system performance assessment will ultimately assume complete radionuclide capture, and a prespecified total water demand provided in the regulations {i.e., 3.7×10^6 m^3/yr [3,000 acre-ft/yr]}, the rates of water pumping by individual farms and the farm sizes are of no real consequence from a regulatory standpoint.

In summary, the NRC staff consider the available data adequate to support the DOE conceptual model for dilution of radionuclides in groundwater due to well pumping, and for model abstraction in performance assessment.

3.3.12.4.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.12.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess dilution of radionuclides in groundwater due to pumping with respect to data uncertainty being characterized and propagated through model abstraction.

Because DOE assumes total radionuclide mass capture at the compliance boundary, the only data uncertainties are associated with two parameters that control the volume of water extracted: the number of farms and the groundwater extraction rate per farm. To account for uncertainty, lower-limit, expected, and upper-limit water use rates were calculated for scenarios of 15, 20, and 25 farms in Amargosa Valley. The lower-limit, expected, and upper-limit rates were based on the 5^{th} percentile, mean, and 95^{th} percentile estimates of per-farm water use rate from the State of Nevada 1997 survey. This approach produced nine discrete pumping rates that are sampled stochastically in total system performance assessment calculations (CRWMS M&O, 2000b, Table 3-28).

To the extent that the total system performance assessment will ultimately assume complete radionuclide capture, and a prespecified total water demand provided in the regulations {i.e., 3.7×10^6 m^3/yr [3,000 acre-ft/yr]}, the rates of water pumping by individual farms and the farm sizes are of no real consequence from a regulatory standpoint.

In summary, the DOE approach—incorporating data uncertainty into total system performance assessment abstractions by sampling nine discrete pumping and dilution scenarios—is sufficient to provide information in a potential license application.

3.3.12.4.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.12.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess dilution of radionuclides in groundwater due to well pumping with respect to model uncertainty being characterized and propagated through model abstraction.

Groundwater data used to estimate the per-farm groundwater usage are for only a single year, 1997. A potential model uncertainty is that groundwater usage in Amargosa Valley may change (increase or decrease) considerably from that assumed for the model abstraction. Nye County representatives have stated, for example, that water demands in Amargosa Valley are expected to increase in the near future.[1] DOE did not explicitly consider changes in groundwater demand in the future in Amargosa Valley in the total system performance assessment.

NRC staff recognize that these uncertainties are potentially important and could result in doses that are different than the calculated dose. An increase in the total groundwater pumping in the Amargosa Valley area would result in a reduced expected dose, because of a greater water volume available for dilution of the radionuclide mass. A decrease in the pumping volume could lead to an increase in the dose. The NRC staff position is that the dose calculation for the safety case is based on the radionuclide capture and total groundwater pumping as defined by the regulations for the proposed high-level waste repository. As it has been stated, the total annual water demand used to evaluate the dose for individual members of the affected population is specified in the regulations to be 3.7×10^6 m^3 [3,000 acre-ft]. As for radionuclide capture, DOE assumes that all the radionuclide mass reaching the compliance boundary in the saturated zone will be captured. For a fixed water demand and radionuclide mass, the calculated dose required by the regulations is virtually unaffected by the groundwater pumping uncertainty.

In addition, it is noted that the regulations in 10 CFR Part 63 preclude projections of changes in society, biosphere (other than climate), human biology, or increases or decreases in human knowledge [10 CFR 63.305(b)].

In summary, the DOE approach is appropriate for inclusion in a potential license application.

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[1]Buqo, T. *Comments made at Nuclear Waste Technical Review Board meeting, January 31, 2001*. Amargosa Valley, Nevada. 2001.

3.3.12.4.5 Model Abstraction Output Is Supported by Objective Comparisons

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.12.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess dilution of radionuclides in groundwater due to pumping with respect to model abstraction output being supported by objective comparisons.

As indicated in Section 3.3.12.4.2, estimates of groundwater pumping rates are based on a survey of groundwater pumping in Nye County and census data. In addition, the saturated zone flow model provides some support for the abstraction in the sense that the estimated groundwater withdrawal rate can be sustained by the available inflow. Specifically, estimates of the expected annual withdrawal rate for a community of 25 farms total approximately 3.08×10^6 m^3/yr [2,500 acre-ft/yr] (CRWMS M&O, 2000b, Table 3-28), whereas the saturated zone flow model calculates a total groundwater flux of approximately 23.43×10^6 m^3/yr [19,000 acre-ft/yr] into Amargosa Valley (CRWMS M&O, 2000a).

In summary, the DOE approach for treatment of dilution of radionuclides in groundwater due to well pumping in the total system performance assessment considers available geologic, hydrologic, and geochemical data.

3.3.12.5 Status and Path Forward

Table 3.3.12-1 provides the status of all key technical issue subissues, referenced in Section 3.3.12.2, for the Representative Volume Integrated Subissue. The table also provides the related DOE and NRC agreements pertaining to the Representative Volume Integrated Subissue. The agreements listed in the table are associated with one or all five generic acceptance criteria discussed in Section 3.3.12.4. Note that the status and the detailed agreements (or path forward) pertaining to all the key technical issue subissues are provided in Table 1.1-3 and Appendix A.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (through specified testing, analyses, and the like), acceptably addresses the NRC questions so that no information beyond that provided, or agreed to, will likely be required at the time of a potential license application.

Table 3.3.12-1. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreements*
Unsaturated and Saturated Flow Under Isothermal Conditions	Subissue 5—Saturated Zone Flow and Dilution Processes	Closed-Pending	None
Total System Performance Assessment and Integration	Subissue 1—System Description and Demonstration of Multiple Barriers	Closed-Pending	None
	Subissue 2—Scenario Analysis and Event Probability	Closed-Pending	TSPAI.2.01 TSPAI.2.02 TSPAI.2.03

Table 3.3.12-1. Related Key Technical Issue Subissues and Agreements (continued)			
Key Technical Issue	**Subissue**	**Status**	**Related Agreements***
Total System Performance Assessment and Integration	Subissue 3—Model Abstraction	Closed-Pending	None
	Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards	Closed-Pending	None
*Related DOE and NRC agreements are associated with one or all five generic acceptance criteria.			

3.3.12.6 References

CRWMS M&O. “Saturated Zone Flow and Transport Process Model Report.” TDR–NBS–HS–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000a.

———. “Biosphere Process Model Report.” TDR–MGR–MD–000002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000b.

———. “Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations.” TDR–WIS–RL–000001. Revision 04. Las Vegas, Nevada: CRWMS M&O. 2000c.

———. “Total System Performance Assessment for the Site Recommendation.” TDR–WIS–PA–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000d.

———. “Features, Events, and Processes in Saturated Zone Flow and Transport.” ANL–NBS–MD–000002. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2001.

DOE. “Viability Assessment of a Repository at Yucca Mountain.” Overview and all five volumes. DOE/RW–0508. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 1998.

NRC. “Issue Resolution Status Report, Key Technical Issue: Unsaturated and Saturated Flow Under Isothermal Conditions.” Revision 2. Washington, DC: NRC. 1999.

———. “Issue Resolution Status Report, Key Technical Issue: Total System Performance Assessment and Integration.” Revision 3. Washington, DC: NRC. 2000.

———. NUREG–1804, "Yucca Mountain Review Plan—Draft Report for Comment." Revision 2. Washington, DC: NRC. March 2002.

State of Nevada. “Ground Water Pumpage Inventory of Amargosa Valley.” Number 230 1997. Carson City, Nevada: State of Nevada, Department of Conservation and Natural Resources. 1997.

U.S. Census Bureau. "1990 Census Database." C90STF3A. Summary Level: State–County. County Subdivision: Amargosa Valley Division. Washington, DC: U.S. Census Bureau. 1999. <http://venus.census.gov/cdrom/lookup>

3.3.13 Redistribution of Radionuclides in Soil

3.3.13.1 Description of Issue

The Redistribution of Radionuclides in Soil Integrated Subissue addresses the movement of radionuclides following deposition on the ground, either through surface application of groundwater or settling of volcanic ash following an eruption. Redistribution affects the quantity and concentrations of radionuclides accessible to receptors in the biosphere, and therefore, influences the dose from radionuclides deposited on the ground. The relationships between this integrated subissue and other integrated subissues are depicted in Figure 3.3.13-1. The overall organization and identification of all the integrated subissues are depicted in Figure 1.2-2.

The DOE description and technical basis for the redistribution of radionuclides in soil abstractions are documented in CRWMS M&O (2000a,b), and five supporting analysis and model reports, (CRWMS M&O, 2000c–g). Portions of additional analysis and model reports are reviewed to the extent they contain data or analyses that support the proposed total system performance assessment abstractions. This section provides a review of the abstraction of redistribution of radionuclides in soil incorporated by DOE in its Total System Performance Assessment.

3.3.13.2 Relationship to Key Technical Issue Subissues

The Redistribution of Radionuclides in Soil Integrated Subissue incorporates subject matter previously captured in the following five key technical issue subissues:

- Igneous Activity: Subissue 2—Consequences of Igneous Activity (NRC, 1999)
- Total System Performance Assessment and Integration: Subissue 1—System Description and Demonstration of Multiple Barriers (NRC, 2000)
- Total System Performance Assessment and Integration: Subissue 2—Scenario Analysis and Event Probability (NRC, 2000)
- Total System Performance Assessment and Integration: Subissue 3—Model Abstraction (NRC, 2000)
- Total System Performance Assessment and Integration: Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards (NRC, 2000)

The key technical issue subissues formed the basis for the previous versions of the issue resolution status reports and also were the basis for technical exchanges with DOE where agreements were reached on what additional information DOE needed to provide to resolve

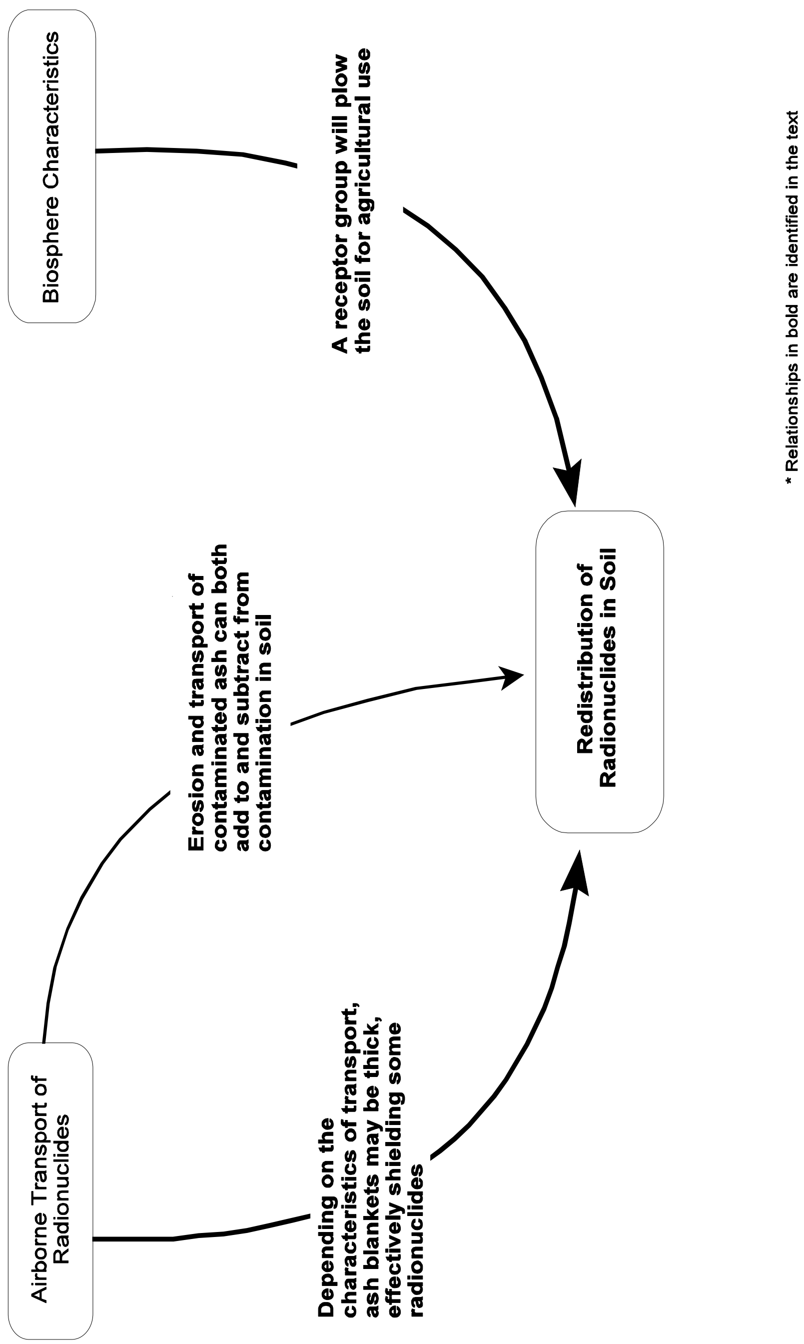

Figure 3.3.13-1. Diagram Illustrating the Relationship Between Redistribution of Radionuclides in Soil and Other Integrated Subissues

the subissue. The resolution status of this integrated subissue is based on the resolution status of each of the contributing key technical issue subissues.

The subsequent sections incorporate applicable portions of these key technical issue subissues, however, no effort was made to explicitly identify each subissue.

3.3.13.3 Importance to Postclosure Performance

The importance of appropriately assessing the effects of igneous activity on the repository system is illustrated in CRWMS M&O (2000a). This document indicates that igneous activity is the only natural process that can cause waste package failure and dose to the reasonably maximally exposed individual (called the receptor from here on) during the regulatory period of interest. Processes such as the redistribution of radionuclides in soil following an igneous event would be evaluated to ensure that models predicting the dose from igneous activity do not underestimate the risk associated with igneous activity.

Following an igneous event at the proposed repository location, a submillimeter-to-decimeter thick deposit will be deposited at the receptor location. For any future eruption through the proposed repository site, some amount of tephra will be deposited on slopes that are part of the Fortymile Wash drainage basin.

Through time, the high-level-waste-bearing tephra will be mobilized off these slopes through, and into, the Fortymile Wash drainage system. Sediment residence times in the confined channel of Fortymile Wash should be short relative to residence times on most hill slopes around Yucca Mountain. Bed-load sediments will move down the main Fortymile Wash drainage during periods of high water flow, with suspended-load sediments mobilized by relatively lower water flow. Just north of Highway 95, the main Fortymile Wash drainage changes from a steep-sided channel to a broad, braided fan system. This location represents the point below which significant long-term sediment deposition occurs within the Fortymile Wash drainage system. Sediment deposition and alluvial aggradation continue south into the Amargosa Desert and overlap the general area of the receptor location. This deposition of remobilized tephra could counteract the loss of radionuclides at the receptor location due to local erosion and lead to a net accumulation of radionuclides for some period of time following the event.

For the groundwater pathway, redistribution of radionuclides in soil affects the concentration of radionuclides in the surface soil. Irrigation of agricultural fields through multiple growing seasons can lead to a buildup of radionuclides in the soil. DOE assessments indicated that for most radionuclides, buildup of radionuclides in the soil has a minor effect on the calculated dose conversion factors (CRWMS M&O, 2000b), with the biosphere dose conversion factor increasing by less than a factor of two for most radionuclides, even for buildup times on the order of thousands of years.

3.3.13.4 Technical Basis

NRC has developed a Yucca Mountain Review Plan (NRC, 2002) that is consistent with the acceptance criteria and review methods found in the previous issue resolution status reports. A review of DOE approaches for including redistribution of radionuclides in soil in total system performance assessment abstractions is provided in the following subsections. The review is organized according to the five acceptance criteria identified in Section 1.5 as follows: (i) System Description and Model Integration Are Adequate, (ii) Data Are Sufficient for Model Justification, (iii) Data Uncertainty Is Characterized and Propagated Through the Model Abstraction, (iv) Model Uncertainty Is Characterized and Propagated Through the Model Abstraction, and (v) Model Abstraction Output Is Supported by Objective Comparisons.

In DOE (1998), analysis of redistribution of radionuclides in soil following an igneous event accounted for only the removal of radionuclides at the receptor location due to erosion, leaching, and radioactive decay of the deposited material. DOE has not developed a model to determine the effects of remobilization of radionuclides deposited upstream of the receptor following an igneous event. Instead, DOE makes several conservative assumptions about other processes and states that these conservative assumptions will bound the effects of remobilization of radionuclides.

3.3.13.4.1 System Description and Model Integration Are Adequate

Overall, the current information, along with agreements reached between DOE and NRC, (Section 3.3.13.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess redistribution of radionuclides in soil with respect to system description and model integration.

The features, events, and processes relevant to the Redistribution of Radionuclides in Soil Integrated Subissue (Dose 2) are listed in Section 3.2.1 of this report. The DOE technical bases for including or excluding the features, events, and processes related to redistribution of radionuclides in soil are provided in the analysis and model report (CRWMS M&O, 2000c). The following paragraphs provide a brief description of the conceptual and modeling approaches developed by DOE to integrate features, events, and processes that affect the redistribution of radionuclides in soil into the total system performance assessment abstraction.

The approach and technical basis for the methodology used to account for the effects of remobilization of radionuclides by aeolian and fluvial processes following deposition by an igneous event are documented in CRWMS M&O (2000a). DOE proposes to bound the potential effects of remobilization by (i) assuming that the wind blows toward the receptor throughout every modeled eruption; (ii) using transition-phase biosphere dose conversion factors for all time following the igneous event; and (iii) using biosphere dose conversion factors calculated for a thin [1-cm (0.39-in)] ash layer, neglecting any dilution of radionuclides in clean soil below the tephra deposit. The first assumption is considered conservative because it neglects variations in wind direction at the repository location, which could cause smaller quantities of radionuclides to be deposited initially at the receptor location. Scoping calculations in Hill and Connor (2000), however, suggest that the long-term accumulation of remobilized tephra may exceed original fallout thicknesses by a factor of 10. The second assumption overestimates doses by maintaining the relatively high airborne particle concentrations that would be expected for a number of years following an igneous eruption for all time after the volcanic event. This conservatively neglects the processes that could decrease the amount of resuspendable ash particles in the deposit, such as wind removal and rainwater infiltration. Offsetting that conservatism is the potential for a net influx of resuspendable ash through wind and water remobilization. Finally, the third assumption is conservative because it assumes that all radionuclides are concentrated in the upper centimeter of the deposit when calculating dose from the deposited radionuclides. This increases the dose from direct exposure and inhalation pathways for thicker deposits. Furthermore, when calculating removal due to erosion from these deposits, DOE assumes that deposited radionuclides are spread throughout a 15-cm [6-in] soil layer, to reduce the quantity of radionuclides removed each year. DOE agreed to provide further justifications, including supporting data, for all of its assumptions and modeling approaches.[1]

The approach for calculating the change in concentration of radionuclides in the soil following deposition is described in the analysis and model report (CRWMS M&O, 2000d) and in the process model report (CRWMS M&O, 2000b). The dynamics of the radionuclide concentration in the top layer of soil are governed by a conservation equation where the rate of change in radionuclide concentration in a volume of soil is equal to the quantity flowing in (from either irrigation or ash fall) minus the amount being removed. Mechanisms of potential radionuclide removal from the soil include radioactive decay, plant uptake, leaching into the deeper soil layer and physical loss of soil (i.e., erosion by wind and water). Countering the removal of radionuclides from the soil in the groundwater release scenario is the continual addition of radionuclides from irrigation. The igneous scenario assumes there is no input of radioactive material into the system following the initial deposition. Buildup of radionuclides in soil was modeled using the GENII-S Version 1.485 computer code (Napier, et al., 1988). GENII-S uses a K_d approach to calculate a leaching factor based on a formula derived in Baes and Sharp (1983), which determines the fraction of a given radionuclide that leaches out of the surface soil to deeper soil depths each year. GENII-S does not account for erosion of radionuclides as a removal mechanism for radionuclides in the surface soil. GENII-S modeling predicted that the buildup of most radionuclides in the soil due to multiple years of irrigation with contaminated water would increase the biosphere dose conversion factor by less than 15 percent at equilibrium (i.e., for an infinitely long buildup time). For those radionuclides, the GENII-S prediction of the biosphere dose conversion factor for the concentration of radionuclides in the soil, once equilibrium was reached, was used in the model (CRWMS M&O, 2000b). For the remaining five radionuclides, the equilibrium concentration was calculated outside of the GENII-S code, including the erosion of the soil, and this soil buildup factor was applied to the biosphere dose conversion factor prior to fitting a distribution to the range of possible biosphere dose conversion factor outputs for sampling in the total system performance assessment. Removal of radionuclides from the tephra deposit considered the loss due to leaching for only a single year of irrigation, but included removal by erosion for all years following the eruption.

[1]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (June 21–22, 2001)." Letter (June 29) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

The approach for calculating the concentration of radionuclides in the air following deposition in the soil is described in the Biosphere Process Model Report (CRWMS M&O, 2000b). The concentration of radionuclides in the air is calculated using the mass loading model in GENII-S (CRWMS M&O, 2000b). This model assumes the concentration of radionuclides on dust in the air [i.e., particle diameters $\leq$100 μm (0.004 in)] is equivalent to the concentration of radionuclides on the ground, and all dust in the air is contaminated. This model is appropriate for situations where the contamination consists of a relatively thick deposit and is widespread such that dust that blows into the area from upwind has a similar level of contamination as dust generated locally.

Output from the DOE redistribution of radionuclides in soil is used differently in calculating dose from the groundwater pathway and air pathway. For the groundwater pathway, the redistribution of radionuclides in soil analysis is performed outside of the total system performance assessment and provides the number of years of irrigation prior to the year for which the dose is being calculated as an input value to the GENII-S code (Leigh, 1993); GENII-S then calculates the concentration of radionuclides in the soil from a unit concentration of radionuclides in the water pumped from the ground to calculate a biosphere dose conversion factor. This biosphere dose conversion factor is multiplied by the calculated time-dependent concentration of radionuclides in the water pumped from the ground to calculate the time-dependent dose. For the air pathway, the redistribution of radionuclides in soil provides the time-dependent concentration of radionuclides in the soil following an igneous event. This concentration is multiplied by a biosphere dose conversion factor derived using GENII-S with a unit concentration of radionuclides in the soil to calculate the time-dependent dose from an igneous event.

Following is a summary of staff review regarding system description and model integration. CRWMS M&O (2000b) and supporting analysis and model reports provide sufficient information about the methodology used to incorporate the effects of the redistribution of radionuclides in soil and of the couplings between models for NRC to make a regulatory decision at the time of any future license application. DOE should demonstrate that the assumptions for bounding the effects of remobilization of radionuclides following an igneous event are appropriate. The use of the K_d approach in GENII-S (Leigh, 1993) to model the leaching of radionuclides out of the surface soil is reasonable for relatively low concentrations of radionuclides in the soil, as would likely be found in the groundwater discharge scenarios. For scenarios in which higher concentrations of radionuclides may be found on the ground surface, however, a check should be performed to ensure the concentration of radionuclides leaching out of the surface soil does not exceed the solubility limit of the radionuclide (Jarzemba and Manteufel, 1997). This check does not seem to have been performed in the DOE modeling. The models used to describe the removal of radionuclides from the soil due to erosion appear to include all significant processes as long as DOE can demonstrate that the current approach to account for remobilization of tephra by wind and water is reasonable. The use of a mass-loading model to predict the concentration of radionuclides in the air is appropriate for the scenarios analyzed in the total system performance assessment. DOE agreed to provide further justifications, including supporting data, for all of its assumptions and modeling approaches, including all of the issues above, prior to, or as part of, any potential license application.[2] NRC staff are satisfied, based on the agreements, that sufficient information will be available at the time of any potential license application review.

3.3.13.4.2 Data Are Sufficient for Model Justification

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.13.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess redistribution of radionuclides in soil with respect to data being sufficient for model justification.

Detailed descriptions of the data sets used to support the models of the redistribution of radionuclides in soil are found in these analysis and model reports (CRWMS M&O, 2000d–g).

As described in CRWMS M&O (2000a), no model has been developed to describe the remobilization of radionuclides due to aeolian and fluvial processes following an igneous event. Instead, three conservative assumptions have been made to bound the effects of remobilization.

Data used to model erosion and leaching out of the surface soil following deposition are described in two analysis and model reports (CRWMS M&O, 2000d,h). The best estimates of erosion rates used in the analysis are derived from the soil loss tolerance factor, which is defined as the maximum annual rate of soil erosion that can occur while still maintaining productivity indefinitely (Troeh, et al., 1980) for the different soil types found in the vicinity of Lathrop Wells, Nevada. The soil loss tolerance factor for each soil type is taken from Brady (1984), and data on the abundance and properties of the different soils found in the proposed receptor location are from the U.S. Department of Agriculture Natural Resource Conservation Service (CRWMS M&O, 1999). A bounding value for the analysis was assumed to be zero soil loss due to the potential for improved land management techniques to minimize the loss of soil from the farm. The leaching analysis used K_d values from Sheppard and Thibault (1990) for sandy soils, which are the types of soils found in Amargosa Valley (CRWMS M&O, 1999). Estimates of precipitation at the receptor location are taken from measurements of the annual precipitation at Lathrop Wells between 1986 and 1997. Values of evapotranspiration and irrigation were based on alfalfa production in Amargosa Valley (CRWMS M&O, 2000d).

Data used to support the mass loading values are described in the analysis and model report (CRWMS M&O, 2000e). For the nominal case, average annual outdoor concentrations of PM_{10} from similar arid farming communities were used to develop a distribution of concentrations representative of the reference biosphere farming community. Analog PM_{10} concentrations were obtained from the Office of Air Quality Planning and Standards AIRSData database (EPA, 2000), which contains air quality data collected by state and local agencies and reported to EPA to monitor compliance with Federal air quality standards. Analog sites were selected

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[2]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (June 21–22, 2001)." Letter (June 29) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

that were classified as a land use of agricultural and a location type of rural. These sites were further narrowed to limit selection to locations that had an arid climate similar to the Yucca Mountain region and little snowfall, which tends to decrease the level of airborne particulate matter. This process resulted in the selection of 5 analog sites, for which 19 measurements of annual average PM_{10} concentrations were available. The average of these measurements resulted in an annual average PM_{10} mass loading of 42 $\mu g/m^3$ [2.6×10^{-9} lb/ft^3]. This average value of PM_{10} mass loading was multiplied by the average value of the total suspended particulates to PM_{10} ratio, 2.5, which was measured in the Yucca Mountain region, to yield an annual average mass load of 105 $\mu g/m^3$ [6.5×10^{-9} lb/ft^3]. The use of this EPA database accounts for increases in airborne particulate concentration due to surface disturbing activities for time periods appropriate for a farming community. The mass load following an igneous event is higher than the nominal mass load. The annual average outdoor concentration of PM_{10} particles immediately following the eruption is assumed to be 1,000 $\mu g/m^3$ [6.2×10^{-8} lb/ft^3] based on comparison with measurements made of mass loads for different levels of disturbance (including walking/driving, outdoor play, and inside combines and farm trucks) following eruptions at Mount St. Helens and Montserrat, which had a higher concentration of fine material than would be expected from an eruption at Yucca Mountain. This mass loading value corresponds to a concentration that EPA characterizes as a level at which serious and widespread health effects occur to the general population (EPA, 1994). Data from Mount St. Helens indicate the ratio of total suspended particulates to PM_{10} is about 3.0 based on data for agricultural farming and within homes (Buist, et al., 1986). Based on data from three areas surrounding Mount St. Helens that indicated total suspended particulate values returned to preeruption values within a year of the eruption, DOE assumed that within 10 years of cessation of a volcanic eruption the concentration of resuspended particles decreases to background levels similar to that of a farming community (Bernstein, et al., 1986).

Following is a summary of staff review regarding data sufficiency and model justification. DOE needs to collect sufficient data to support the assertion that the conservative assumptions used to replace modeling of the remobilization process will bound the risk associated with igneous activity. If this assertion cannot be supported, additional data will be needed to model the remobilization process. The analysis would be strengthened by the use of site-specific K_d values instead of generic values from Sheppard and Thibault (1990) because these values can vary significantly due to variations in soil pH and other soil characteristics. Additional data are needed to support the assumption that the concentration of resuspended particles returns to background values within 10 years of cessation of an igneous event. As discussed in CRWMS M&O (2000e), data from Mount St. Helens on the rate at which particle concentrations return to nominal levels are not directly comparable to those postulated for a Yucca Mountain volcano because of (i) differences in the quantity of precipitation and snowfall between the two locations, (ii) higher concentrations of vegetation in Washington compared with the Lathrop Wells area, (iii) absence of coarser particles in the analog deposit to inhibit erosion, and (iv) lower initial mass loading values following the event at Mount St. Helens compared with estimates for Yucca Mountain and data in Hill and Connor (2000). It is not clear that increasing the observed reduction rate by a factor of 10 is sufficient to account for differences between the two locations. DOE will need to demonstrate that long-term input of fine particulates through wind and water remobilization would not significantly affect the proposed mass-load reduction factor. DOE agreed to provide further justifications, including supporting data, for all of its assumptions and modeling approaches, including all of the issues above, prior to, or as part of, any potential license application.[3] Based on the agreements, sufficient information will be available at the time of any potential license application review.

DOE has collected sufficient and appropriate pedological, hydrological, and geochemical data to adequately define relevant parameters necessary for developing the other portions of the abstraction of redistribution of radionuclides in the soil in the total system performance assessment. DOE has adequately described how data have been used and synthesized into parameters.

3.3.13.4.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.13.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess redistribution of radionuclides in soil with respect to data uncertainty being characterized and propagated through the model abstraction.

Detailed descriptions of the methodology used to characterize and propagate data uncertainty in the redistribution of radionuclides in soil abstraction are found in CRWMS M&O (2000a) and these four analysis and model reports (CRWMS M&O, 2000d–g).

DOE asserts that the conservative assumptions made in modeling the volcanism scenario bound the uncertainty associated with the process of remobilization of radionuclides deposited in areas other than the receptor location due to fluvial and aeolian processes. As such, uncertainty associated with this is not explicitly incorporated in the total system performance assessment. DOE should demonstrate that the assumptions made actually bound the effects of remobilization. DOE agreed to justify its assumptions.[4]

Uncertainty in data for the erosion and leaching of radionuclides out of the surface soil does not appear to have been appropriately incorporated into the total system performance assessment. GENII-S (Napier, et al., 1988) does not allow the user to sample the leaching factor and does not include erosion in the model. In the analysis and model report, (CRWMS M&O, 2000d), best estimate values and bounding values were developed for erosion rates and leaching factors. Best estimate values are based on the mean value of input parameters, while bounding values are based on the worst-case input values for all parameters. These values were used in two analysis and model reports (CRWMS M&O, 2000f,g), to develop a best estimate range and bounding value for the biosphere dose conversion factors. The best estimate range does not include any consideration of the variability in the K_d value, and CRWMS M&O (2000a) appears to only use this best estimate range in its modeling. Therefore, no uncertainty in the K_d value is incorporated into the results of the Total System Performance

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[3]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (June 21–22, 2001)." Letter (June 29) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[4]Ibid

Assessment–Site Recommendation. Similarly, the erosion rate developed is based on the maximum erosion rate that could be maintained and allow the field to continue to be used for agriculture. It is indicated, however, that current practice in agricultural communities is to manage soil resources to maintain soil erosion losses at levels well below the established tolerable soil loss rate, making it plausible to eliminate soil erosion entirely (CRWMS M&O, 2000d). Because higher erosion rates are less conservative in the analysis, using the maximum credible erosion rate without accounting for the potential for the erosion rate to be lower in the total system performance assessment is not appropriate. DOE agreed to address this issue.[5]

Data uncertainty for the mass loading above agricultural fields and tephra deposits is explicitly incorporated in the total system performance assessment. For the nominal scenario, the uncertainty in the mass-loading factor is derived from the variation in the measured values of the annual average mass loading at the analog agricultural sites. The method used to account for the variation in the mass-loading value for the extrusive volcanism scenario in the total system performance assessment is more complex. The concentration of particulates in the air following an igneous eruption will decrease through time. As described in the previous section, DOE estimated the annual average total suspended particulates concentration in the air immediately following an igneous event to be 3,000 $\mu g/m^3$ [1.9×10^{-7} lb/ft^3] for a thick tephra deposit that will drop to nominal levels {105 $\mu g/m^3$ [6.5×10^{-9} lb/ft^3]} within 10 years. The analysis mixes temporal variability with data uncertainty by using the mean value of a loguniform distribution {864 $\mu g/m^3$ [5.3×10^{-8} lb/ft^3]} to represent the average mass loading during the first 10 years following the eruption for a thick deposit. Further, DOE argues that because thin deposits will not cause the average mass loading during 10 years to increase significantly above the nominal value, and the distribution of tephra deposit thicknesses is approximately exponential, it is reasonable to sample the value of the mass-loading parameter from a loguniform distribution between the nominal value of 105 $\mu g/m^3$ [6.5×10^{-9} lb/ft^3] and the 10-year average value above a thick deposit of 864 $\mu g/m^3$ [5.3×10^{-8} lb/ft^3] (CRWMS M&O, 2000e). Sampling from a log-uniform distribution between the nominal mass load representing a thin deposit and the average mass load for a thick deposit assumes the average mass load during the first 10 years following an event is directly proportional to the thickness of the deposit. DOE has not provided sufficient technical basis for this assumption. It seems reasonable to assert that thin deposits (i.e., less than several millimeters) will be removed relatively quickly and will not significantly influence the average mass load during the 10 years following an eruption. Once a critical thickness of deposit is reached so that the deposit is thick enough to maintain a fines-depleted shield to protect lower levels of the deposit that contain significant quantities of fines, it is likely the mass load will increase rapidly beyond this critical level, reacting more like a step function. Additional comments on the reasonableness of the range of tephra thicknesses predicted to be deposited at the receptor location are located in Section 3.2.11.4. If the minimum thickness of tephra deposit is significantly greater than currently predicted in CRWMS M&O (2000a), this methodology of sampling may not be appropriate. Finally, this methodology does not account

[5]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (June 21–22, 2001).” Letter (June 29) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

for the remobilization of radionuclides causing deposits at the receptor location to thicken, which would also affect whether this methodology of sampling from a range with a lower value equivalent to the nominal mass load to take credit for varying deposit thicknesses is appropriate. Demonstration that the effects of remobilization are bounded by other conservative assumptions needs to ensure that this credit for thin deposits is accounted for.

Following is a summary of staff review regarding characterization and propagation of data uncertainty. DOE needs to demonstrate that the conservative assumptions used to replace modeling of the remobilization process are appropriate. DOE has failed to explicitly incorporate the uncertainties associated with leaching and erosional rates of radionuclides into the total system performance assessment and needs to include these uncertainties in the model or demonstrate that neglecting this uncertainty will not significantly affect the results of the calculation. The mixing of temporal variability and parameter uncertainty in development of the mass loading above a tephra deposit is confusing and will only provide correct results if other time-dependent processes do not result in a significant change in the concentration of radionuclides in the soil during the 10-year period for which temporal averaging is performed. Specifically, DOE needs to demonstrate that processes to remove radionuclides from the soil (i.e., decay, erosion, and leaching) do not cause a significant change in the concentration of radionuclides in the soil during the first 10 years following the igneous eruption. DOE needs to provide further justification that its sampling from a loguniform distribution between the nominal mass load representing a thin deposit and the average mass load for a thick deposit is reasonable or conservative, accounting for the remobilization of radionuclides causing deposits at the receptor location to potentially thicken. Demonstration that the effects of remobilization are bounded by other conservative assumptions needs to ensure that this credit for thin deposits is accounted for. DOE agreed to provide further justifications, including supporting data, for all its assumptions and modeling approaches, including all of the issues above, prior to, or as part of, any potential license application.[6] Based on the agreements, sufficient information will be available at the time of any potential license application review.

3.3.13.4.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.13.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess redistribution of radionuclides in soil with respect to model uncertainty being characterized and propagated through the model abstraction.

Detailed descriptions of the methodology used to characterize and propagate model uncertainty in the redistribution of radionuclides in soil abstraction are found in CRWMS M&O (2000a) and in the analysis and model report (CRWMS M&O, 2000e).

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[6]Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (June 21–22, 2001).” Letter (June 29) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

As described in CRWMS M&O (2000a), no model has been developed to describe the remobilization of radionuclides due to aeolian and fluvial processes following an igneous event. Instead, three conservative assumptions have been made to bound the effects of remobilization. DOE needs to provide justification that the methodology used to bound the effects of remobilization does not underestimate the risk from igneous activity. DOE agreed to provide the justification.[7]

The methodology used to model erosional removal of radionuclides is reasonable and sufficient, given an appropriate data set. Therefore, NRC staff agree that no alternative modeling is necessary for the erosional model. The International Atomic Energy Agency (2001) indicated that radionuclide experiments in recent years have indicated that migration of radionuclides in soil is dominated by radionuclides bound to small particles and that the sorption/desorption process only contributes to a minor extent, especially for radionuclides with a high K_d value. DOE may wish to investigate this alternative model of leaching to determine how it would affect the results of the total system performance assessment.

An alternative model was considered for modeling the resuspension process to determine the concentration of radionuclides in the air in the analysis and model report (CRWMS M&O, 2000e). This report considered the use of a resuspension model, which correlates the concentration of radionuclides in the air to concentration of radionuclides on the ground through use of a resuspension factor. For the widespread area of contamination and the relatively thick deposits of contamination associated with an igneous event, DOE argued that a mass-loading model is more appropriate. Additionally, data supporting mass-loading values are more readily available to support the model than the resuspension factor needed for the resuspension model. This approach to assessing model uncertainty is sufficient for inclusion in a potential license application.

Following is a summary of staff review regarding characterization and propagation of model uncertainty. DOE needs to provide justification that the methodology used to bound the effects of remobilization does not underestimate the risk from igneous activity. The DOE has agreed to provide further justifications, including supporting data, for all of its assumptions and modeling approaches, including all of the issues above, prior to, or as part of, any potential license application.[8] NRC staff are satisfied, based on the agreements, that sufficient information will be available at the time of any potential license application review. DOE has adequately considered appropriate alternative conceptual models for other processes in the redistribution of radionuclides in soil abstraction and has provided sufficient justification for the selection of preferred models.

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[7]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (June 21–22, 2001)." Letter (June 29) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[8]Ibid

3.3.13.4.5 Model Abstraction Output Is Supported by Objective Comparisons

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.13.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess redistribution of radionuclides in soil with respect to model abstraction output being supported by objective comparisons.

As described in CRWMS M&O (2000a), no model has been developed to describe the remobilization of radionuclides due to aeolian and fluvial processes following an igneous event. Instead, three conservative assumptions have been made to bound the effects of remobilization. Therefore, verification of the accuracy of the model is not possible. DOE needs to provide justification that the methodology used to bound the effects of remobilization does not underestimate the risk from igneous activity.

The model for tracking concentration of radionuclides in soil, which includes processes such as deposition by irrigation and losses by erosion, leaching, and decay, is a simple box model for which the input data control the results. Provided that the computer code is verified to be performing the mathematics correctly and the input data are determined to be appropriate for the materials and activities being modeled, support for the model used is not necessary.

The mass-loading model that estimates the concentration of radioactive material in the air relies on the assumption that the concentration of radioactive material in the air is the same as the concentration of radioactive material on the ground. No attempt has been made in the DOE analysis and model reports to compare the results of this model with field data.

Following is a summary of staff review regarding verification of the redistribution of radionuclides in soil abstraction. DOE needs to provide justification that the methodology used to bound the effects of remobilization does not underestimate the risk from igneous activity. DOE should compare the results of the mass-loading model to field data to demonstrate that use of the mass-loading model does not underestimate the concentration of radionuclides in the air compared with the concentration of radionuclides on the ground. DOE agreed to provide further justifications, including supporting data, for all of its assumptions and modeling approaches, including all of the issues above, prior to, or as part of, any potential license application.[9] NRC staff are satisfied, based on the agreements, that sufficient information will be available at the time of any potential license application review.

3.3.13.5 Status and Path Forward

Table 3.3.13-1 provides the status of all key technical issue subissues, referenced in Section 3.3.13.2, for the Redistribution of Radionuclides in Soil. The table also provides the related DOE and NRC agreements to the Redistribution of Radionuclides in Soil. The agreements listed in the table are associated with one or all five generic acceptance criteria

[9] Reamer, C.W. “U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (June 21–22, 2001).” Letter (June 29) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

discussed in Section 3.3.13.4. Note that the status and the detailed agreements (or path forward) pertaining to all the key technical issue subissues are provided in Table 1.1-3 and Appendix A.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (through specified testing, analyses, and the like), acceptably addresses the NRC questions so that no information beyond that provided, or agreed to, will likely be required at the time of a potential license application.

Table 3.3.13-1. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreement*
Igneous Activity	Subissue 2—Consequences of Igneous Activity	Closed-Pending	IA.2.06 IA.2.07 IA.2.08 IA.2.11 through IA.2.17
Total System Performance Assessment and Integration	Subissue 1—System Description and Demonstration of Multiple Barriers	Closed-Pending	None
	Subissue 2—Scenario Analysis and Event Probability	Closed-Pending	TSPAI.2.01 TSPAI.2.02 TSPAI.2.03
	Subissue 3—Model Abstraction	Closed-Pending	TSPAI.3.33
	Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards	Closed-Pending	None

*Related DOE and NRC agreements are associated with one or all five generic acceptance criteria.

3.3.13.6 References

Baes, III, C.F., and R.D. Sharp. “A Proposal for Estimation of Soil Leaching Constants for Use in Assessment Models.” *Journal of Environmental Quality*. Vol. 12. pp. 17–28. 1983.

Bernstein, R.S., P.J. Baxter, H. Falk, R. Ing, L. Foster, and F. Frast. “Chapter 3: Immediate Public Health Concerns and Actions in Volcanic Eruptions: Lessons from the Mount St. Helens Eruptions, May 18–October 18, 1980.” *Health Effects of Volcanoes: An Approach to Evaluating the Health Effects of an Environmental Hazard*. R.S. Bernstein and A.S. Buist, eds. *American Journal of Public Health*. Vol. 76, Supplement. Washington, DC: American Public Health Association. 1986.

Brady, N.C. *The Nature and Properties of Soil*. 9th Edition. New York City, New York: Macmillan Publishing Co. 1984.

Buist, A.S., T.R. Martin, J.H. Shore, J. Butler, and J.A. Lybarger. "Chapter 4: The Development of a Multidisciplinary Plan for Evaluation of the Long-Term Health Effects of the Mount St. Helens Eruptions." *Health Effects of Volcanoes: An Approach to Evaluating the Health Effects of an Environmental Hazard*. R.S. Bernstein and A.S. Buist, eds. *American Journal of Public Health*. Vol. 76, Supplement. Washington, DC: American Public Health Association. 1986.

CRWMS M&O. "Evaluation of Soils in the Northern Amargosa Valley." B0000000–01717–5705–00084. Revision 00. Las Vegas, Nevada: CRWMS M&O. 1999.

———. "Total System Performance Assessment for the Site Recommendation." TDR–WIS–PA–000001. Revision 00 ICN 01. North Las Vegas, Nevada: DOE, Yucca Mountain Site Characterization Office. 2000a.

———. "Biosphere Process Model Report." TDR–MGR–MD–000002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000b.

———. "Evaluation of the Applicability of Biosphere-Related Features, Events, and Processes." ANL–MGR–MD–000011. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000c.

———. "Evaluate Soil/Radionuclide Removal by Erosion and Leaching." ANL–NBS–MD–000009. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000d.

———. "Input Parameter Values for External and Inhalation Radiation Exposure Analysis." ANL–MGR–MD–000001. Revision 01 ICN 00. Las Vegas, Nevada: CRWMS M&O. 2000e.

———. "GENII-S Disruptive Event Biosphere Dose Conversion Factor." ANL–MGR–MD–000003. Revision 00B. Las Vegas, Nevada: CRWMS M&O. 2000f.

———. "GENII-S Non-Disruptive Event Biosphere Dose Conversion Factor." ANL–MGR–MD–000009. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000g.

———. "Identification of Suggestion Exposure Parameters." ANL–MGR–MD–000006. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000h.

DOE. "Viability Assessment of a Repository at Yucca Mountain." Overview and all five volumes. DOE/RW–0508. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 1998.

EPA. "Measuring Air Quality: The Pollutant Standards Index." EPA 451/K–94–001. Washington, DC: EPA. 1994.

———. “AIRSData, Monitor All Columns Report, United States Air Quality Monitors for Particulate Matter (All Years).” Research Triangle Park, North Carolina: EPA, Office of Air Quality Planning and Standards. 2000. <http://www.epa.gov/airsdata/moncols.htm> (16 June 2000).

Hill, B.E. and C.B. Connor. “Technical Basis for Resolution of the Igneous Activity Key Technical Issue.” San Antonio, Texas: CNWRA. 2000.

International Atomic Energy Agency. “An International Peer Review of the Biosphere Modeling Programme of the DOE’s Yucca Mountain Site Characterization Project.” Report of the International Atomic Energy Agency International Review Team. Vienna, Austria: International Atomic Energy Agency. 2001.

Jarzemba, M.S. and R.D. Manteufel. “Leaching-Chain Decay of Radionuclides From Soil.” *Health Physics*. Vol. 73, No. 6. pp. 919–927. 1997.

Leigh, C.D., B.M. Thompson, J.E. Campbell, D.E. Longsine, R.A. Kennedy, and B.A. Napier. “User’s Guide for GENII-S: A Code for Statistical and Deterministic Simulation of Radiation Doses to Humans from Radionuclides in the Environment.” SAND91–0561. Albuquerque, New Mexico: Sandia National Laboratories. 1993.

Napier, B.A., R.A. Peloquin, D.L. Strenge, and J.V. Ramsdell. “Volume 1: Conceptual Representation. GENII—The Hanford Environmental Radiation Dosimetry System.” PNL–6584. Richland, Washington: Pacific Northwest Laboratory. 1988.

NRC. “Issue Resolution Status Report, Key Technical Issue: Igneous Activity.” Revision 2. Washington, DC: NRC. 1999.

———. “Issue Resolution Status Report, Key Technical Issue: Total System Performance Assessment and Integration.” Revision 3. Washington, DC: NRC. 2000.

———. NUREG–1804, "Yucca Mountain Review Plan—Draft Report for Comment." Revision 2. Washington, DC: NRC. March 2002.

Sheppard, M.I. and D.H. Thibault. “Default Soil Solid/Liquid Partition Coefficients, K_ds, for Four Major Soil Types: A Compendium.” *Health Physics*. Vol. 59, No. 4. pp. 471–482. 1990.

Troeh, F.R., J.A. Hobbs, and R.L. Donahue. *Soil and Water Conservation for Productivity and Environmental Protection*. Englewood Cliffs, New Jersey: Prentice-Hall, Inc. 1980.

3.3.14 Biosphere Characteristics

3.3.14.1 Description of Issue

The Biosphere Characteristics Integrated Subissue encompasses technical and regulatory issues regarding development and implementation of total system performance assessment models to convert concentration estimates of radionuclides in soil and groundwater to human dose estimates that can be used to assess compliance with 10 CFR Part 63 dose limits. Model development is based on a combination of site-specific and relevant technical information and scientific principles applied within the regulatory policy framework established in 10 CFR Part 63. The Biosphere Characteristic Integrated Subissue includes the features, events, and processes that impact fate and transport of radioactive contamination in the biosphere and subsequent exposure of the dose receptor (i.e., the reasonably maximally exposed individual). The dose receptor is a hypothetical individual defined by regulation (for dose modeling) in 10 CFR Part 63 to be protective of the vast majority of the potentially exposed population (i.e., an individual based on characteristics derived from local populations that live in the accessible environment directly above the area of highest radionuclide concentration in the groundwater plume). The reference biosphere is defined also by regulation in 10 CFR Part 63 and represents (for dose modeling) the local environment of the dose receptor. Radioactive releases from a potential repository can enter the biosphere through transport processes, such as saturated zone flow, following a postulated groundwater release and airborne fallout resulting from a postulated volcanic event. The DOE description and technical basis for biosphere dose modeling are documented in CRWMS M&O (2000a) and various supporting analysis and model reports. Only Revision 00 reports were reviewed to support this status report. Revisions to CRWMS M&O (2000a) or any of the analysis and model reports will be reviewed as they become available, and results will be documented in future reports or meetings.

3.3.14.2 Relationship to Key Technical Issue Subissues

The Biosphere Characteristic Integrated Subissue is derived from the dose calculation component of the biosphere subsystem (Figure 1.1-2). The relationships between biosphere characteristic and other integrated subissues are illustrated in Figure 3.3.14-1. The overall organization and identification of all the integrated subissues are depicted in Figure 1.1-2. The Biosphere Characteristics Integrated Subissue incorporates subject matter previously captured in the following key technical subissues:

- Radionuclide Transport: Subissue 3—Radionuclide Transport Through Fractured Rock (NRC, 2000a)
- Igneous Activity: Subissue 2—Consequences of Igneous Activity (NRC, 1999a)
- Unsaturated and Saturated Flow Under Isothermal Conditions:
Subissue 1—Climate Change (NRC, 1999b)

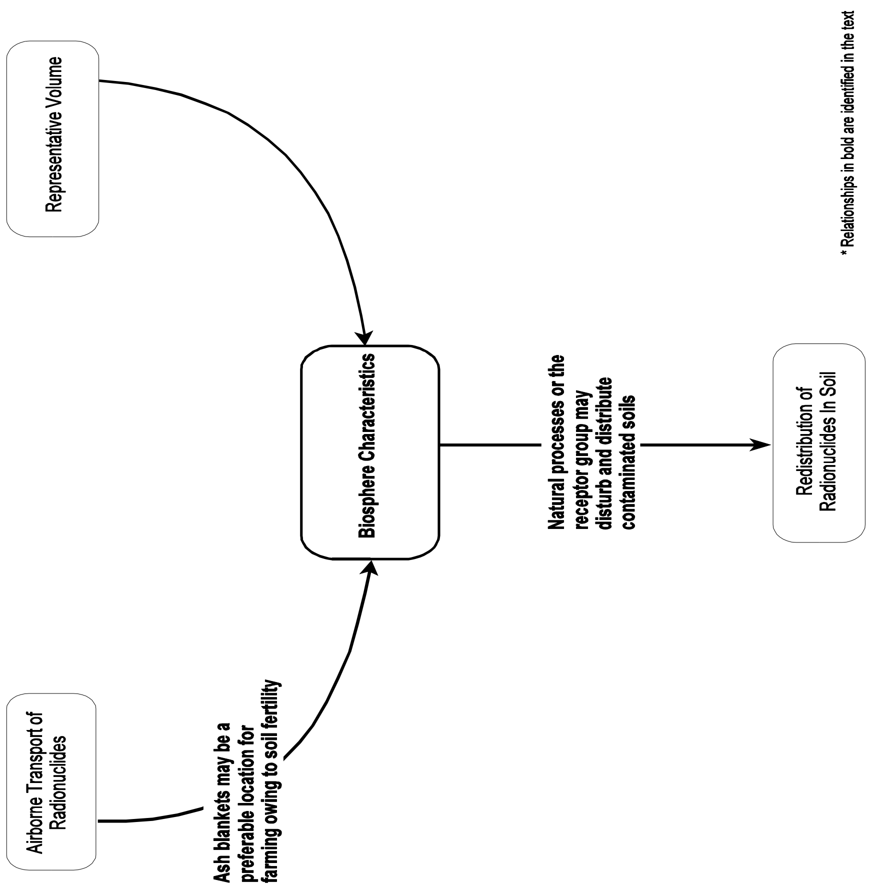

Figure 3.3.14-1. Diagram Illustrating the Relationship Between Biosphere Characteristics and Other Integrated Subissues

- Unsaturated and Saturated Flow Under Isothermal Conditions: Subissue 2—Hydrologic Effects of Climate Change (NRC, 1999b)

- Unsaturated and Saturated Flow Under Isothermal Conditions: Subissue 3—Present Day Shallow Groundwater Infiltration (NRC, 1999b)

- Unsaturated and Saturated Flow Under Isothermal Conditions: Subissue 5—Saturated Zone Ambient Flow Conditions and Dilution Processes (NRC, 1999b)

- Total System Performance Assessment and Integration: Subissue 1—System Description and Demonstration of Multiple Barriers (NRC, 2000b)

- Total System Performance Assessment and Integration: Subissue 2—Scenario Analysis and Event Probability (NRC, 2000b)

- Total System Performance Assessment and Integration: Subissue 3—Model Abstraction (NRC, 2000b)

- Total System Performance Assessment and Integration: Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards (NRC, 2000b)

The key technical issue subissues formed the bases for the previous versions of the issue resolution status reports and were also the bases for technical exchanges with DOE where agreements were reached on what additional information DOE needed to provide to resolve the subissue. The resolution status of this integrated subissue is based on the resolution status of each of the contributing key technical issue subissues. The subsequent sections incorporate applicable portions of these key technical issue subissues.

3.3.14.3 Importance to Postclosure Performance

One aspect of risk informing the NRC review was to determine how this integrated subissue is related to the DOE repository safety strategy. DOE initially determined that the biosphere dose conversion factors were important parameters in the total system performance assessment calculations (DOE, 1998), but later demonstrated diminished importance of the biosphere in sensitivity studies in CRWMS M&O (2000b). This change in significance was attributed to the small variation DOE propagated in the biosphere dose conversion factor distributions after parameter changes to mean values for parameters now specified by regulation. Staff propagate a slightly larger, yet relatively small, amount of variation in biosphere dose conversion factors in the TPA Version 4.0 code (Mohanty, et al., 2002) calculations for radionuclides important to performance. Nonetheless, staff sensitivity analyses have identified a few important biosphere parameters in system-level sensitivity analyses. Staff expect the small amount of variation in the biosphere dose conversion factors places the biosphere at a borderline level of importance. Furthermore, staff sensitivity analyses are based on the total amount of uncertainty and variation propagated in biosphere dose conversion factors, whereas DOE sensitivity analyses truncated the distribution at the 5^{th} and 95^{th} percentiles. Because stochastic biosphere dose conversion factor results for important radionuclides approximate lognormal distributions

(CRWMS M&O, 2000a), DOE truncation at the 95th percentile significantly reduces the range of values. This truncation directly impacts the results of the DOE perturbation type of sensitivity analysis because that analysis method is sensitive to the range of the parameter being analyzed. If DOE changes parameter ranges in the process of resolving existing agreements, or if the magnitude of radionuclide concentrations change in the total system performance assessment calculations, the importance of biosphere dose conversion factor distributions could change, and the sensitivity analyses may need to be updated. As a result, staff will continue to monitor DOE updates of the biosphere dose modeling abstraction.

3.3.14.4 Technical Basis

NRC developed a plan (2002) consistent with the acceptance criteria and review methods found in previous issue resolution status reports. A review of DOE approaches for including biosphere characteristics in total system performance assessment abstractions is provided in the following subsections. The review is organized according to the five acceptance criteria identified in Section 1.5: (i) System Description and Model Integration Are Adequate, (ii) Data Are Sufficient for Model Justification, (iii) Data Uncertainty Is Characterized and Propagated Through the Model Abstraction, (iv) Model Uncertainty Is Characterized and Propagated Through the Model Abstraction, and (v) Model Abstraction Output Is Supported by Objective Comparisons. Review methods have been formulated to focus on those aspects of the abstraction that prior sensitivity studies have shown are important to performance (LaPlante, et al., 1995; LaPlante and Poor, 1997) and relevant to the NRC requirements for 10 CFR Part 63. A review of the DOE approach to biosphere characteristics for each acceptance criterion is contained in the sections that follow.

3.3.14.4.1 System Description and Model Integration Are Adequate

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.14.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the biosphere characteristics with respect to system description and model integration.

The system description for biosphere characteristics supports identification, screening, and integration of features, events, and processes to aid development, selection, and integration of conceptual and mathematical models. Identification and screening of features, events, and processes related to the biosphere are included in Section 3.2.1 of this report. Therefore, this section will concentrate on the adequacy of the DOE overall system description supporting conceptual model development, selection, and integration.

The reference biosphere and dose receptor must be developed and implemented within the regulatory framework provided by the 10 CFR Part 63 requirements. Some important characteristics of the biosphere and dose receptor have been explicitly defined in 10 CFR Part 63 requirements to avoid unnecessary speculation. Although DOE is not required to justify characteristics of the biosphere and dose receptor explicitly defined in the regulation (e.g., drinking water consumption rate, and location of the dose receptor), supporting information is needed to define characteristics not explicitly defined in 10 CFR Part 63 (e.g., irrigation rates, food consumption, and outdoor activity).

DOE developed various documents that describe the biosphere and dose receptor at various levels of detail. A general description of the biosphere and dose receptor is provided in CRWMS M&O (2000b). More detailed technical information is provided in a series of analysis and model reports that cover specific aspects of the biosphere and dose receptor. In general, these reports provide a system description that is adequate for understanding the bases for selection of exposure scenarios, identification of exposure pathways, and selection or development of models for biosphere dose modeling. Staff concerns were identified during the review; however, most of the concerns relate to transparency and traceability which are covered by existing agreements.

The following discussion will focus on the status of various important aspects of the biosphere system description and model integration that staff reviewed. For discussion purposes, these aspects include the general system description that supports the overall conceptual dose model exposure scenarios and pathway information. A more detailed discussion of specific technical areas, including support for establishing the habits of the dose receptor, support for modeling processes related to fate and transport of radioactive materials in the biosphere, and documentation of the bases for the implementation of biosphere dose modeling in total system performance assessment calculations, is also included.

In defining the dose receptor, 10 CFR 63.312(b) requires the diet and living style to be representative of the people who now reside in the town of Amargosa Valley, Nevada. The regulation also requires DOE to use projections based on surveys of the people residing in the town of Amargosa Valley, Nevada, to determine living styles and use mean values for the performance assessment calculations. Staff review of DOE documentation (CRWMS M&O, 2000a,b) indicates demographic surveys of Amargosa Valley have been completed and documented, and the results are incorporated as mean value parameters into the biosphere dose modeling. 10 CFR 63.312(e) also requires the dose receptor to be an adult with metabolic and physiological considerations consistent with present knowledge of adults. In CRWMS M&O (1999), DOE documents the use of adult dosimetry in its application of dose coefficients from existing EPA Federal Guidance reports (1988, 1993) that NRC commonly uses and accepts for dose modeling. DOE also indicates the location of the dose receptor will be 18 km [11 mi] south of Yucca Mountain as required by 10 CFR Part 63.

The general description of the biosphere dose modeling provided in CRWMS M&O (2000b) includes a dose receptor and biosphere intended to be consistent with regulations proposed by EPA and NRC. The receptor is described as a member of a hypothetical farming community presumed to be exposed to radionuclide releases to groundwater (nominal scenario) and air (for the disruptive volcanic event scenario). The reference biosphere is based on characteristics of Amargosa Valley which includes a climate characterized as arid to semiarid (considering potential future climate evolution). Alfalfa production and dairy farming are noted as primary agricultural activities in the area. Water for all uses in the area comes predominantly from local wells. Census data and results of a survey of local residents provide information on the lifestyle characteristics of people in the region. Information on biosphere characteristics is adequate for inclusion in a potential license application.

The DOE conceptual model of the biosphere includes a scenario (i.e., nominal case) where radionuclides presumed to leach from the repository are transported to the location of the dose receptor where wells pump the contaminated water to the surface. The community where the dose receptor resides then uses the pumped water. The nominal scenario provides one mechanism for transporting radioactive materials to the biosphere. A separate disruptive event scenario involves a volcanic eruption that transports airborne particles of ash contaminated with radionuclides to the biosphere location for deposition and contamination of surface soil. DOE used its understanding of these mechanisms of biosphere contamination, along with a detailed analysis of biosphere features, events, and processes, to refine the conceptual model of the biosphere and identify potential exposure pathways that should be included in the biosphere dose modeling.

The biosphere conceptual model emphasizes aspects of the biosphere that can directly contribute to exposure of the human dose receptor. This model includes transfer of radionuclides to soil, the atmosphere, and flora and fauna (CRWMS M&O, 2000b). The conceptual model for movement of material within the biosphere is consistent with commonly known fate and transport models including deposition of radionuclides from water to soil through irrigation, from soil to air through resuspension, and from air to soil through deposition. Subsequent movement of material occurs from air and soil to plants and from water and plants to livestock. Human exposure to radioactive material from inhalation, ingestion, and external exposure pathways result from contact with contaminated air, water, food products (both plant and animal), and soil. Staff identified an additional transport mechanism for the volcanic scenario involving redistribution of contaminated ash deposits during a review of the DOE model. Redistribution in the biosphere is covered by another integrated subissue (Redistribution of Radionuclides in Soil) and is addressed by an existing agreement (Section 3.3.13), which may result in collection of additional information to support the conceptual model. The remainder of the DOE biosphere conceptual model appears to be well supported by the existing information. Results of the staff review of the DOE features, events, and processes analysis for the biosphere have identified concerns predominantly related to transparency and traceability, which have been incorporated into existing agreements.[1]

Integration with related integrated subissues was evident from reviews of the DOE biosphere abstraction. A number of biosphere modeling issues related to the igneous activity scenario are receiving technical input from the Igneous Activity Subissue 2 (e.g., redistribution and mass-loading). DOE conducted analyses into the effects of natural climate change on biosphere dose conversion factors in Bechtel SAIC Company, LLC (2001), but decided not to use the revised biosphere dose conversion factors because climate had the effect of lowering the dose (nonconservative). DOE has also developed biosphere dose conversion factors for those radionuclides expected to transport through the saturated zone (or be transported by an igneous event). The issues regarding transport of radioactive material in the saturated zone and the atmosphere (from igneous events) are sufficiently understood to translate the relevant modeling concepts to dose calculations. Resolutions of some issues from the Igneous Activity Integrated Subissue will provide input to further improve the technical bases for biosphere dose modeling in the future (e.g., redistribution and mass-loading). Overall, the staff did not identify

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[1]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

any major integrated subissue integration issues impacting the biosphere dose modeling when they reviewed the DOE reports.

In summary, the system description DOE provided is based on local surveys and other available information appropriate for supporting the conceptual model of the biosphere and receptor group. The DOE conceptual model is consistent with a detailed features, events, and processes analysis that is found to be generally comprehensive for the biosphere, which DOE is updating to address a current agreement regarding transparency and traceability issues.[2] At the general conceptual model level, it is unlikely that any additional features, events, or processes significant to the dose calculation will be identified after resolution of existing agreements. Resolving agreements related to including redistribution of volcanic ash may add complexity to the present conceptual model. At a more detailed submodel level, some models may be optimized or updated, but these modifications are not expected to significantly change the overall conceptual model of the biosphere.

3.3.14.4.2 Data Are Sufficient for Model Justification

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.14.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the biosphere dose modeling with respect to data being sufficient for model justification.

DOE selected a series of mathematical models for the biosphere dose modeling consistent with the needs of the biosphere conceptual model. The mathematical models are contained within the GENII-S dose modeling software program (Leigh, et al., 1993). DOE selected GENII-S because it has the flexibility to model the features, events, and processes that have been included in the biosphere conceptual model for Yucca Mountain. NRC has not identified any major problems with the code selection or justification; however, the DOE resolution of some existing agreements may result in the use of additional models for specific biosphere processes (e.g., redistribution and leaching) (Section 3.3.13.5).

The DOE implementation of the biosphere dose modeling in total system performance assessment calculations uses lookup tables of biosphere dose conversion factors that convert groundwater and soil concentrations into human doses. DOE uses the GENII-S modeling to generate the tables for the total system performance assessment calculations. Insufficient justification was provided for this implementation approach in the DOE documents. As a result, an existing agreement[3] requests DOE to provide further justification for the selection of this implementation approach to demonstrate it does not significantly bias the original GENII-S results.

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[2]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[3]Ibid.

The DOE biosphere dose conversion factor calculations using GENII-S require a large number of parameter selections. The input parameters for the biosphere dose conversion factor calculations are documented in various analysis and model reports that the NRC staff has reviewed. The review effort focused on those parameters found to be important for the GENII-S dose modeling. Both DOE (CRWMS M&O, 2000c,d) and staff (LaPlante and Poor, 1997) conducted sensitivity analyses at the process model level that identified a similar set of important input parameters. These parameters include consumption rates (e.g., water, vegetables, and milk), animal and plant uptake factors, a resuspension factor, and crop interception fraction.

Both DOE (CRWMS M&O, 2000c) and staff (LaPlante and Poor, 1997) have shown that consumption rates, which have a direct impact on the magnitude of modeled doses, are important at the process model level. Because 10 CFR 63.312(b) requires behavioral input parameters to be based on mean values (e.g., no variation propagated in the performance calculation), the parameter cannot be included in system-level sensitivity analyses without violating the requirement. Nonetheless, importance at the process level suggests the magnitude of the selected mean values used for total system performance assessment calculations can directly impact dose results, and the mean values should, therefore, be adequately justified. Furthermore, 10 CFR 63.312(b) requires the consumption rates to be based on local survey, data and such survey data will serve to address the regulatory requirement as well as address the technical need for justification.

The DOE mean value consumption rates are supported by results of a stratified random sample survey of the local population that the University of Las Vegas Cannon Center for Survey Research conducted (CRWMS M&O, 2000e). The survey included the population residing within 84 km [52 mi] of Yucca Mountain including the communities of Amargosa Valley, Beatty, Indian Springs, and Pahrump. Information was collected on the frequency of locally produced food and water consumption, which was then converted into amounts consumed by applying average intake information from a national survey. Intakes were not measured directly because recall of specific intake amounts is less reliable than frequency information. Staff found descriptions of the survey methodology, execution, and analysis of results in CRWMS M&O (2000e) provide sound bases for the consumption rate parameter information. Staff continue to await DOE publication of the detailed documentation for the Amargosa Valley survey.

Animal and plant uptake factors are important parameters for the process-level biosphere modeling (CRWMS M&O, 2000c; LaPlante and Poor, 1997). Preliminary system-level sensitivity results conducted by staff suggest plant uptake can be important in the total system performance calculations. These factors are used in the plant and animal uptake models to transfer contaminants from soil to plants and from feed to livestock (Napier, et al., 1988).
The DOE technical basis for selection of plant and animal uptake factors is provided in CRWMS M&O (2000f). Because DOE indicates no site-specific information is available, it has used available information from the technical literature to select values for Yucca Mountain. Although this is likely to be the case, staff informed DOE of radionuclide transfer studies EPA conducted at the Nevada Test Site that could be applicable to Yucca Mountain. DOE agreed to

investigate the information and provide a technical basis for transfer factor information.[4] The staff also questioned the DOE method for selecting values from the available technical literature. The selection method is partially based on the frequency of occurrence of the same parameter values in the literature rather than the applicability of the values to conditions at Yucca Mountain. Because transfer coefficient research is limited, some references in the technical documents that DOE reviewed refer to the same source data. Although staff do not have technical concerns with the source data, DOE has been encouraged to use original references to source data and to consider using a more technically sound basis for selecting transfer coefficients relevant to the site conditions at Yucca Mountain (rather than selecting data because they are most frequently used by others). Again, DOE agreed to provide a technical basis for radionuclide specific parameters important to biosphere dose conversion factors including transfer coefficients.[5]

Crop interception fraction is the fraction of the contaminants in irrigation water deposited on the plant surface. DOE (CRWMS M&O, 2000c) and the NRC staff indicate that the crop interception fraction (LaPlante and Poor, 1997) is important to dose modeling at the process level. The parameter has been found to be moderately important in some staff system-level sensitivity checks. The importance of the parameter in dose calculations is influenced by the width of the parameter probability distribution. DOE discusses the crop interception fraction in CRWMS M&O (2000g). DOE adopts a calculation for the interception fraction from Hoffman (1989) that is based on comparisons with experimental data from two radionuclides. DOE calculations result in a normal distribution for the parameter from approximately 0.044 to 0.47 at the 99.9 percent confidence interval, with a mean of 0.26. For comparison, prior staff biosphere calculations used a triangular distribution with a conservative range (0.06 to 1.0) and mode of 0.40 based on a technical expert calculation after a review of 20 studies (of varying applicability and quality) in the technical literature by Anspaugh (1987). The studies reviewed by Anspaugh (1987) involve 10 radionuclides; however, neither DOE nor staff identified any of the radionuclides considered as important contributions.

Staff raised a concern that the two radionuclides forming the basis for the DOE approach may not be representative of the entire suite of radionuclides considered important in the total system performance assessment calculation, and DOE has agreed to conduct an analysis of the applicability of the assumed crop interception fraction to all important radionuclides in the total system performance assessment. The following discussion provides further justification for the original comment[6] regarding the applicability of the DOE crop interception fraction values to other radionuclides.

Anspaugh (1987) considered a variety of studies conducted in controlled laboratory conditions with fine sprays as well as field studies involving natural rainfall and radioactive fallout. The

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[4]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[5]Ibid.

[6]Ibid.

majority of values presented in Anspaugh (1987) fall within the DOE range. Some values are higher than the DOE upper limit value, but DOE indicates the studies are based on fine sprays of small amounts of water not applicable to agricultural field irrigation conditions at Yucca Mountain (e.g., where large amounts of water must reach the soil) (CRWMS M&O, 2000g). Staff review of the data also found that many of the elevated results reported in field studies are compromised by the potential effects of atmospheric deposition on rain collectors and imprecision leading to values much greater than one. Because the overall weight of the evidence reviewed suggests values within the DOE range, and the calculation appears technically sound, staff do not have major concerns that would invalidate the present DOE approach. Nonetheless, there are field results outside the DOE range not explained by the fine spray or low-volume argument used in the analysis and model report. Values have been recorded between 0.65 and 0.80 for individual rainfall events of 0.5 to 2 cm [0.2 to 0.8 in] volume that is consistent with the DOE stated daily irrigation application rate of 1 cm [0.4 in]. Staff did agree with DOE that the prior total system performance assessment value with a maximum of 1.0 is probably an overestimate for an irrigation scenario where the objective is to water the crop roots in soil. As a result, staff will continue to conduct confirmatory calculations and sensitivity studies using a range wider than the DOE range [i.e., consistent with the Anspaugh (1987) data] but narrower than the previously used range (e.g., 0.06 to 0.80) until DOE provides additional information to address the agreement regarding the applicability of the crop interception fraction to all important radionuclides.

The mass-loading factor combines several biosphere processes into a coefficient used to determine the concentration of radionuclides in air from known soil concentrations. Resuspension is important for the inhalation pathway that dominates the biosphere dose calculations for the igneous activity disruptive event scenario (CRWMS M&O, 2000d). The staff has raised various issues (leading to agreements) regarding the DOE basis for the mass-loading factor. Although the mass-loading factor is used in the biosphere dose modeling, the issues are discussed in the Redistribution of Radionuclides in Soil Integrated Subissue (Section 3.3.13).

Specific agreements were developed for issues where initial DOE responses to staff concerns were incomplete. Some initial DOE responses to staff concerns that were initially adequate included DOE action items to be completed in the future. These action items include (i) update the radionuclide inventory analysis and model report to account for biological transport in radionuclide screening, (ii) improve documentation of the assumptions in a future revision to the environmental transport analysis and model report, (iii) update the analysis and model report entitled Transfer Coefficient Analysis to include methods for combining data based on individual crops to food groups and a clarified definition of conservatism, and (iv) complete additional model validation for the GENII-S code (Leigh, et al., 1993). These items will be checked by staff when the revised analysis and model reports are available.

In summary, DOE parameter choices for biosphere dose conversion factor calculations to support the biosphere dose modeling abstraction are, in general, consistent with available data and adequately justified except for a few exceptions where DOE agreed to provide additional information to resolve issues. Consumption rates are adequately documented. Other parameters, such as transfer coefficients, the crop interception fraction, and the mass-loading factor need additional justification.

3.3.14.4.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.14.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the biosphere characteristics with respect to data uncertainty characterization and propagation through the model abstraction.

This section discusses the status of issues related to uncertainty propagation in the GENII-S (Leigh, et al., 1993) biosphere dose conversion factor calculations as well as in the implementation of the abstraction. As described in CRWMS M&O (2000b), DOE propagates biosphere dose modeling input parameter uncertainty by executing the GENII-S code stochastically using input parameter distributions to generate a biosphere dose conversion factor distribution for each radionuclide. The parameter uncertainty is propagated through the biosphere abstraction by sampling from the biosphere dose conversion factor distributions for each realization of the total system performance assessment code. Uncertainty propagation of biosphere dose conversion factors is potentially important because of the impact on sensitivity results. DOE concluded the biosphere dose conversion factors are not important in the total system performance assessment whereas staff analyses suggest the importance of a few biosphere parameters at the total system level. Staff believe the limited range of biosphere dose conversion factors used in the total system performance assessment calculations produce a borderline level of importance. Changes to the total system performance assessment to resolve existing agreements, however, could impact the level of importance of biosphere parameters, and, therefore, staff will continue to monitor biosphere uncertainty propagation in future reviews.

As noted before, the NRC regulations in 10 CFR Part 63 limit the propagation of parameter uncertainty by requiring the use of mean values for behavioral input parameters (i.e., diet and living style) such as consumption rates and exposure times. Consumption rates have been shown to be important parameters in process model level sensitivity analyses conducted by both DOE (CRWMS M&O, 2000c) and staff (LaPlante and Poor, 1997). Fixing consumption rates at the mean values eliminates a substantial portion of the uncertainty propagated to the biosphere dose conversion factors as DOE indicated in CRWMS M&O (2000b); however, nonbehavioral parameters contribute variability to the biosphere dose conversion factors to the extent that the total range for most biosphere dose conversion factor distributions are approximately order of magnitude.

Propagated uncertainty and variation in DOE biosphere dose conversion factors for most radionuclides are similar to staff-generated results, when geometric means and standard deviations are compared. Staff-generated distributions are somewhat wider, but the magnitude is not considered significant (possible impacts on sensitivity analysis conclusions are discussed in Section 3.3.14.3). The difference can be explained partly by differences in the ranges used for the crop interception fraction (Section 3.3.14.2). The difference in total system performance assessment results using both the DOE and NRC ranges for crop interception fraction produces similar geometric mean biosphere dose conversion factors with only moderate differences in geometric standard deviations.

Regarding the DOE implementation of the biosphere dose modeling abstraction, an issue was raised about the need for DOE to address the potential for the abstraction to introduce bias into the original GENII-S modeling results. Because DOE samples from radionuclide-specific biosphere dose conversion factor distributions created by the GENII-S code, the resulting suite of selected biosphere dose conversion factors for any particular realization of the total system performance assessment is unlikely to be based on the same set of input parameters for all radionuclides (i.e., the original GENII-S output vectors have been disrupted by the sampling). It is more likely that each radionuclide-specific biosphere dose conversion factor would be based on a suite of sampled input parameters different from any other radionuclide-specific biosphere dose conversion factor sampled for that realization. Such conditions are physically impossible when the conceptual model suggests the radionuclides exist in the same biosphere at the same time. DOE has also correlated the sampling of biosphere dose conversion factor distributions for all radionuclides to the sampling for one radionuclide (Np-237). This method of sampling is inconsistent with the GENII-S modeling results that indicate the magnitude of each biosphere dose conversion factor is affected by radionuclide-dependent factors that vary in effect on dose. For example, a high plant transfer coefficient scale factor may greatly increase the magnitude of biosphere dose conversion factors for radionuclides where plant uptake is high but have little impact on the biosphere dose conversion factor for radionuclides with low plant transfer coefficients. To ensure such deviations from the original process level modeling do not impact results, DOE agreed to conduct a quantitative analysis to demonstrate its selected abstraction approach does not significantly bias the total system performance assessment results.[7]

In summary, propagation of uncertainty is limited in biosphere dose modeling by regulations specifying the use of mean values for behavioral parameters. This regulatory specification reduces propagated uncertainty to levels that lead to low or borderline significance of the biosphere in sensitivity studies. The range of uncertainty propagated in the biosphere, however, could be impacted by resolution of existing agreements; therefore, staff will continue to monitor the issues. In general, the range of biosphere dose conversion factor distributions span no more than one order of magnitude, which is low relative to the uncertainty propagated in other total system performance assessment abstractions. The DOE implementation of the biosphere dose modeling involves correlated sampling of process model output that may generate results different from the original process modeling. DOE agreed to conduct a quantitative analysis to test the potential for the approach to bias results.

[7]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

3.3.14.4.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the biosphere dose modeling with respect to characterization and propagation of model uncertainty through the model abstraction.

Biosphere dose modeling is a highly abstracted and idealized type of modeling that lacks precision. Many available models for biosphere dose calculations are based on similar conceptual models and mathematical representations. Available models, such as those in the GENII-S code (Leigh, et al., 1993) are designed to be inherently conservative to avoid underestimation of doses. Furthermore, many of the conceptual models, and some mathematical models (e.g., dosimetry) are sufficiently (and intentionally) constrained by regulation such that DOE is not free to choose alternative models under the present regulation. As a result, staff believe the characterization and propagation of model uncertainty are unnecessary for biosphere dose modeling. The emphasis on propagation of parameter uncertainty is more appropriate for the type of modeling conducted for the biosphere. Nonetheless, because the biosphere dose model represents a compilation of a variety of submodels that represent specific features, events, or processes in the biosphere, some of these submodels may have specific, known limitations that could benefit by a comparison with alternative modeling approaches. Such modeling approaches could be integrated into the biosphere dose modeling in the future, if necessary. Examples include special submodels to account for redistribution of radionuclides in the biosphere, mass-loading (Section 3.3.13), and inhalation calculations. Other than those issues addressed by related integrated subissues, staff have not identified any parts of the biosphere dose modeling where model uncertainty comparisons would help inform the review of the DOE safety case.

In summary, staff believe the abstracted nature of biosphere models precludes the usefulness of model uncertainty comparisons.

3.3.14.4.5 Model Abstraction Output Is Supported by Objective Comparisons

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.14.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the biosphere dose modeling with respect to system description and model integration.

The DOE biosphere dose modeling abstraction consists of the biosphere dose conversion factor distributions, the approach for sampling these factors for each realization, and the routine that multiplies estimated soil and groundwater radionuclide concentrations by the sampled factors to calculate dose. The biosphere dose conversion factor distributions are generated from process modeling using the GENII-S code (Leigh, et al., 1993). DOE has made comparisons to improve confidence that the modeling in the abstraction is being performed correctly, the biosphere dose conversion factor distributions can be verified against the GENII-S modeling results easily with a simple check. DOE has also compared its GENII-S results with other results from the same process model to provide confidence that the code was executed correctly (CRWMS M&O, 2000a). A DOE-sponsored independent technical review of the conceptual model of the biosphere and its implementation using GENII-S also has been completed (CRWMS M&O, 2000a). In response to a concern regarding the potential for the abstraction approach to bias original process model results, DOE agreed to conduct an additional quantitative analysis to check for bias (TSPAI.3.37) as discussed in the previous section. Staff also expect additional documentation to be provided from DOE to resolve a general comment regarding the validation of codes used in the total system performance assessment.

In summary, the nature of the abstraction (look-up table of code results) provides a basis for simple comparisons with process model results. DOE conducted some reasonable comparisons; however, additional comparisons likely will be done to resolve existing agreements regarding potential for bias in the abstraction approach and model validation.

3.3.14.5 Status and Path Forward

Table 3.3.14-1 provides the status of all key technical issue subissues referenced in Section 3.3.14.2 for the Biosphere Characteristics Subissue. The table also provides the related DOE and NRC agreements pertaining to the biosphere dose modeling subissue. The agreements listed in the table are associated with one or all five generic acceptance criteria discussed in Section 3.3.14.4. Note that the status and the detailed agreements (or path forward) pertaining to all the key technical issue subissues are provided in Table 1.1-3 and Appendix A.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (through specified testing, analyses, and the like), acceptably addresses the NRC questions so that no information beyond that provided, or agreed to, will likely be required at the time of a potential license application.

Table 3.3.14-1. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreements*
Radionuclide Transport	Subissue 3—Radionuclide Transport Through Fractured Rock	Closed-Pending	None
Igneous Activity	Subissue 2—Consequences of Igneous Activity	Closed-Pending	IA.2.06 IA.2.07 IA.2.08 IA.2.11 through IA.2.17
Unsaturated and Saturated Flow Under Isothermal Conditions	Subissue 1—Climate Change	Closed-Pending	None

Table 3.3.14-1. Related Key Technical Issue Subissues and Agreements (continued)			
Key Technical Issue	**Subissue**	**Status**	**Related Agreements***
Unsaturated and Saturated Flow Under Isothermal Conditions	Subissue 2—Hydrologic Effects of Climate Change	Closed-Pending	None
	Subissue 3—Shallow Infiltration	Closed-Pending	None
	Subissue 5—Saturated Zone	Closed-Pending	None
Total System Performance Assessment and Integration	Subissue 1—System Description and Demonstration of Multiple Barriers	Closed-pending	None
	Subissue 2—Scenario Analysis and Event Probability	Closed-Pending	TSPAI.2.01 through TSPAI.2.04
	Subissue 3—Model Abstraction	Closed-Pending	TSPAI.3.34 through TSPAI.3.37
	Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards	Closed-Pending	None
*Related DOE and NRC agreements are associated with one or all five generic acceptance criteria.			

3.3.14.6 References

Anspaugh, L. R. “Retention by Vegetation of Radionuclides Deposited in Rainfall—A Literature Summary.” UCRL–53810. Livermore, California: Lawrence Livermore National Laboratory. 1987.

Bechtel SAIC Company, LLC. “FY01 Supplemental Science and Performance Analyses.” Vol. 1: Scientific Bases and Analyses. TDR–MGR–MD–000007. Revision 00 ICN 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2001.

CRWMS M&O. “Dose Conversion Factor Analysis: Evaluation of GENII-S Dose Assessment Methods.” ANL–MGR–MD–000002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 1999.

———. “Biosphere Process Model Report.” TDR–MGR–MD–000002. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000a.

———. “Total System Performance Assessment for the Site Recommendation.” TDR–WIS–PA–000001. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000b.

———. “Non-Disruptive Biosphere Dose Conversion Factor Sensitivity Analysis.” ANL–MGR–MD–000010. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000c.

———. “Disruptive Biosphere Dose Conversion Factor Sensitivity Analysis.” ANL–MGR–MD–000004. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000d.

———. “Identification of the Critical Group (Consumption of Locally Produced Food and Tap Water).” ANL–MGR–MD–000005. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000e.

———. “Transfer Coefficient Analysis.” ANL–MGR–MD–000008. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000f.

———. “Identification of Ingestion Exposure Parameters.” ANL–MGR–MD–000006. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000g.

DOE. “Viability Assessment of a Repository at Yucca Mountain. Vol. 3: Total System Performance Assessment.” DOE/RW–0508/V3. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 1998.

EPA. “Limiting Values of Radionuclide Intakes and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion: Federal Guidance Report No. 11.” EPA 520/1–88–020. Washington, DC: EPA. 1988.

———. “External Exposure to Radionuclides in Air, Water, and Soil.” EPA 402–R–93–081. Washington, DC: EPA. 1993.

Hoffman, F.O., M.L. Frank, B.G. Blaylock, R.D. von Bermuth, E.J. Deming, R.V. Graham, D.A. Mohrbacher, and A.E. Waters. “Pasture Grass Interception and Retention of 131I, 7Be, and Insoluble Microspheres Deposited in Rain.” ORNL–6542. Oak Ridge, Tennessee: Oak Ridge National Laboratory. 1989.

LaPlante, P.A. and K. Poor. “Information and Analyses to Support Selection of Critical Groups and Reference Biospheres for Yucca Mountain Exposure Scenarios.” CNWRA 97-009. San Antonio, Texas: CNWRA. 1997.

LaPlante, P.A., S.J. Maheras, and M.S. Jarzemba. “Initial Analysis of Selected Site-Specific Dose Assessment Parameters and Exposure Pathways Applicable to a Groundwater Release Scenario at Yucca Mountain.” CNWRA 95-018. San Antonio, Texas: CNWRA. 1995.

Leigh, C.D., B.M. Thompson, J.E. Campbell, D.E. Longsine, R.A. Kennedy, and B.A. Napier. “User Guide for GENII-S: A Code for Statistical and Deterministic Simulation of Radiation Doses to Humans from Radionuclides in the Environment.” SAND91–0561. Albuquerque, New Mexico: Sandia National Laboratories. 1993.

Mohanty, S., T.J. McCartin, and D.W. Esh. “Total System Performance Assessment (TPA) Version 4.0 Code: Module Descriptions and User’s Guide.” San Antonio, Texas: CNWRA. 2002.

Napier, B.A., R.A. Peloquin, D.L. Strenge, and J.V. Ramsdell. “GENII: The Hanford Environmental Radiation Dosimetry Software System, Volume 1: Conceptual Representation.” PNL–6584. Richland, Washington: Pacific Northwest Laboratory. 1988.

NRC. “Issue Resolution Status Report, Key Technical Issue: Igneous Activity.” Revision 2. Washington, DC: NRC. 1999a.

———. “Issue Resolution Status Report, Key Technical Issue: Unsaturated and Saturated Flow Under Isothermal Conditions.” Revision 2. Washington, DC: NRC. 1999b.

———. “Issue Resolution Status Report, Key Technical Issue: Radionuclide Transport.” Revision 2. Washington, DC: NRC. 2000a.

———. “Issue Resolution Status Report, Key Technical Issue: Total System Performance Assessment and Integration.” Revision 3. Washington, DC: NRC. 2000b.

———. NUREG–1804, "Yucca Mountain Review Plan—Draft Report for Comment." Revision 2. Washington, DC: NRC. March 2002.

3.4 Demonstration of Compliance with the Postclosure Public Health and Environmental Standards

3.4.1 Demonstration of Compliance with the Postclosure Individual Protection Standard

3.4.1.1 Description of Issue

The analysis of repository performance that demonstrates compliance with the postclosure individual protection standard at 10 CFR 63.311 is necessary to ensure DOE has presented an acceptable analysis demonstrating the safety of the repository system. The analysis of repository performance that demonstrates compliance with the postclosure individual protection standard includes the following parts: (i) appropriate incorporation of scenarios into the DOE total system performance assessment results, (ii) calculation of the annual total effective dose equivalent from the repository system, and (iii) credibility of the DOE total system performance assessment results.

This section provides a review of the methodologies used by DOE to demonstrate that the repository system will meet the postclosure individual protection standard requirements in 10 CFR 63.113(b). The DOE description and technical basis for the analysis of repository performance that demonstrates compliance with the postclosure individual protection standard are documented in CRWMS M&O (1999, 2000a,b)

3.4.1.2 Relationship to Key Technical Issue Subissues

The analysis of repository performance that demonstrates compliance with the postclosure individual protection standard is related to appropriately incorporating scenarios into the total system performance assessment, demonstrating that the DOE total system performance assessment has been conducted correctly, and the results have been appropriately combined for comparison with regulatory limits. This subissue is related to all key technical issue subissues because proper conduct of the DOE total system performance assessment requires identification and incorporation of scenarios and data analysis for conceptual model development and validation, which are the focal points of these key technical issues. The reviews in the past were previously captured (NRC, 2000) within the framework of the following nine key technical issues:

- Igneous Activity
- Structural Deformation and Seismicity
- Evolution of the Near-Field Environment
- Container Life and Source Term
- Thermal Effects on Flow

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- Repository Design and Thermal-Mechanical Effects

- Total System Performance Assessment and Integration

- Unsaturated and Saturated Flow Under Isothermal Conditions

- Radionuclide Transport

The key technical issue subissues formed the bases for the previous versions of the issue resolution status report and also were the bases for technical exchanges with DOE where agreements were reached about the additional information DOE needed to provide to resolve the subissue.

3.4.1.3 Importance to Postclosure Performance

This issue relates to the methodology used to calculate the performance of the proposed Yucca Mountain repository system and to compare the results of the DOE total system performance assessment with the regulatory requirements. Therefore, this issue is directly related to the determination of postclosure safety of the repository.

In addition to calculating the performance at Yucca Mountain during the most likely scenarios, it is important to ensure DOE is appropriately including the consequences of disruptive events in calculating total effective dose equivalent from the repository for comparison against the 0.15 mSv/yr [15 mrem/yr] all pathways dose standard in 10 CFR Part 63. 10 CFR 63.2 indicates in the definition of performance assessment that estimates of dose from disruptive events should be weighted by their probability of occurrence when included in the calculation of dose to the reasonably maximally exposed individual.

3.4.1.4 Technical Basis

NRC developed a plan (2002) consistent with the acceptance criteria and review methods found in previous issue resolution status reports. A review of DOE approaches for analyzing repository performance that demonstrates compliance with the postclosure individual protection standard is provided in the following subsections.

3.4.1.4.1 Appropriate Incorporation of Scenarios into the Total System Performance Assessment Results

Overall, the current information, along with agreements reached between DOE and NRC, is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the incorporation of scenarios into the DOE total system performance assessment results.

The approach and technical basis for the appropriate incorporation of scenarios into the DOE total system performance assessment results are documented by DOE in CRWMS M&O (2000a). Based on the results of the features, events, and processes analysis, DOE concludes there are two disruptive event classes that could significantly affect the repository performance, igneous activity, and seismically induced cladding failure. The probability of extrusive volcanism is incorporated into the DOE total system performance assessment results by multiplying the sampled annual probability of occurrence of extrusive volcanism by the timestep size and the dose from the igneous event assuming an eruptive igneous event occurred before that time for each timestep in the realization. The mean value of these probability-weighted realizations is then calculated for each timestep. The probability of intrusive volcanism is incorporated into the DOE total system performance assessment results by multiplying the sampled probability that an intrusive igneous event has occurred at any time during the simulation by the dose from the event at all timesteps in the realization. The mean value of these probability-weighted realizations is then calculated for each timestep. Both methodologies are acceptable and result in an appropriate estimate of the probability-weighted dose to be compared with the 0.15 mSv/yr [15 mrem/yr] all pathways dose standard in 10 CFR Part 63. DOE does not calculate the nominal dose from the unaffected parts of the repository after an igneous event. The calculation of dose from the nominal case, however, is not weighted by the probability of the nominal scenario class, which is slightly less than one, because the volcanism event class is excluded. The mean probability-weighted dose curve from the disruptive events is added to the conditional nominal case dose to calculate the total effective dose equivalent from the repository. The only concern with combining the results of the nominal case and the igneous scenario is that the same waste packages involved in the igneous event are also counted in the nominal case; however, double counting is acceptable because it increases the doses, a conservative outcome.

The current approach adopted by DOE for incorporating seismically induced cladding failure into its total system performance assessment may not adequately characterize the variability of the consequences. To address this concern, DOE agreed[1] to modify the approach used in its total system performance assessment to estimate the risk caused by seismically induced cladding failure so that the full range of variability in the consequence is accounted for.

3.4.1.4.2 Calculation of the Total Effective Dose Equivalent from the Repository System

Overall, the current information, along with agreements reached between DOE and NRC, is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess calculation of the total effective dose equivalent from the repository system.

The approach and technical basis for the calculation of the total effective dose equivalent from the repository system are documented by DOE in CRWMS M&O (2000a). DOE demonstrates the stability of its total system performance assessment results by plotting the results of the time history of the dose curve from the repository system for different numbers of realizations. NRC staff have concerns that this approach is too qualitative and difficult to conclude that the results are stable, especially when the dose histories are plotted on a logarithmic scale. NRC

[1]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Structural Deformation and Seismicity (October 11–12, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

staff found no indication that similar tests were performed for models that provided results to the total system performance assessment. For example, the biosphere model provides distributions of biosphere dose conversion factors to the total system performance assessment model, but stability checks for these results were not documented. Another example is the saturated zone transport model, which provides 100 transfer functions to be used in the total system performance assessment model. Additional realizations of the total system performance assessment model will not increase the variance in the results of the saturated zone transport model. Again, no stability check was included to show that 100 transfer functions were sufficient to properly represent uncertainty in the saturated zone transport model. To address these concerns, DOE agreed[2] to document the method to be used to demonstrate that the overall results of its total system performance assessment are stable. NRC staff also had concerns that DOE did not provide a methodology to demonstrate the results of its total system performance assessment were stable with respect to discretization of the model in Total System Performance Assessment–Site Recommendation (CRWMS M&O, 2000a). To address this concern, DOE agreed[3] to conduct analyses and provide documentation demonstrating that the results of the performance assessment are stable with respect to discretization. The documentation will include a description of the statistical measures that will be used to support the argument of stability.

Based on the intermediate outputs available in CRWMS M&O (2000c), it appears that sufficient information about intermediate outputs in the DOE total system performance assessment will be available to allow NRC staff to understand how individual components or subsystems contribute to system performance. Concerns about the consistency between the modeling of individual components or subsystems have been documented in all 14 subsections of Section 3.3.0 of this Integrated Issue Resolution Status Report. The results of the analysis in CRWMS M&O (2000a) seem to be consistent with the performance of individual subsystems or components.

3.4.1.4.3 Credibility of the Total System Performance Assessment Results

Overall, the current information, along with agreements reached between DOE and NRC, is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess credibility of the DOE total system performance assessment results.

The approach and technical basis for credibility of the DOE total system performance assessment results are documented by DOE in CRWMS M&O (1999, 2000a,b). Concerns about the consistency between assumptions in different individual modules of the performance assessment code have been documented in all 14 subsections of Section 3.3.0 of this

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[2]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Structural Deformation and Seismicity (October 11–12, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[3]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

Integrated Issue Resolution Status Report. DOE indicated that its TSPA Code will be verified using a two-phase process. The first phase will assure the input construction is in complete accord with the conceptual models of the different processes as developed in a series of relevant and applicable analysis and model reports. This verification will be accomplished by using an independent review process to check a tabular form that lists the different elements of the conceptual models and records their manner of incorporation in the DOE total system performance assessment. The second phase of verification is designed to ensure the GoldSim model (Golder Associates, 2000) provides the correct output for a given input model embodying the full-scale complexity of the Yucca Mountain site. This verification is beyond what has been conducted by Golder Associates for GoldSim and is specifically related to the Yucca Mountain model. This phase consists of three stages. The first stage consists of performing hand calculations at selected times to verify the results of models that rely on the output from another model to produce results. These hand calculations use the output from the upstream model to verify the results of the dependent model. The second stage verifies all the inputs, including both data files and GoldSim arguments, and stand-alone codes that are incorporated into GoldSim as a dynamically linked library. The third stage consists of verifying that transfers of information between dynamically linked libraries are performed correctly when the full-scale Total System Performance Assessment–Site Recommendation model is implemented. This verification includes writing the time-dependent inputs to a dynamically linked library to an output file and comparing these inputs to the correct values as output from the upstream dynamically linked library.

NRC staff have concerns about the validation performed on the DOE TSPA Code. The verification process should demonstrate that (i) the models used have been adequately tested for calculational correctness with all relevant data together with associated uncertainties, (ii) a well-defined and rational assessment procedure has been followed, and (iii) results have been fully disclosed and subjected to quality assurance and review procedures. The verification process should encompass both tests that provide evidence of correct and successful implementation of algorithms and bench-marking or comparative testing against results from other software for cases where accuracy of the code cannot be judged otherwise. DOE has the elements of verification in its Total System Performance Assessment–Site Recommendation and supporting documents. Rigorous verification of the modules and the full code, however, was either not conducted or was not adequately reported. A specific verification plan was not found, and the verification was not uniform across Total System Performance Assessment–Site Recommendation (CRWMS M&O, 2000a). NRC review of CRWMS M&O (2000c) found errors in verification of hand calculations and abstractions in the performance assessment that were operating outside of their intended ranges.[4] Verification was performed only on a median input value run without rationale to justify that this verification was sufficient for a probabilistic model. Verification of CRWMS M&O (2000c) included various levels of analyses to demonstrate the verification of selected aspects of the performance assessment model but did not carry the calculations forward to step through different parts of the model in larger segments. DOE

[4]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Conference Call Regarding Quality Assurance and Performance Assessment Issues." Letter (May 17) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

agreed[5] to document the process used to develop confidence in the total system performance assessment models, such as described in NRC (1999) and to document compliance with the improved process in the verification documentation required by AP-SI.1Q (DOE, 2001).

DOE indicated that models used within the total system performance assessment will be validated in accordance with AP-3.10Q (CRWMS M&O, 2000c). This procedure requires comparing analysis results against data acquired from the laboratory, field experiments, natural and humanmade analog studies, or other relevant observations to validate models used in the total system performance assessment. It also requires that existing engineering-type models be validated using accepted engineering practices. The criteria used to evaluate the appropriateness and adequacy of the model for its intended use may be qualitative or quantitative but must be justified in the model documentation. If data are not available to support validation of the model, DOE AP-3.10Q requires the use and documentation of an alternative approach. Alternative approaches may include one or more of the following activities: (i) peer review or review by international collaborations; (ii) technical review through publication in the open literature; (iii) review of model calibration parameters for reasonableness, or consistency in explanation of all relevant data; (iv) comparison of analysis results with the results from alternative conceptual models, including supporting information to establish a basis for confidence in the selected model; (v) calibration and corroboration within experimental data sets; or (vi) comparison of analysis results with data attained during performance confirmation studies.

NRC staff have concerns about the steps DOE performed to build confidence in its total system performance assessment models. Confidence building in models should include demonstrating that (i) the processes are properly formulated mathematically and correctly parameterized following accepted theories (or tested theories if a new theory is used), (ii) numerical schemes used have acceptable convergence properties, and (iii) space and time dimensionality is appropriate. DOE has the elements of model validation in its documents supporting Total System Performance Assessment–Site Recommendation (CRWMS M&O, 2000a). A model validation plan does not appear to exist, however. Rigorous model validation at the system level has either not been conducted or has not been adequately reported. For example, the discussion of validation of the mathematical model of the biosphere (GENII-S) (Leigh, et al., 1993) includes only aspects of software verification. DOE has collected field and laboratory data to support detailed hydrologic calculations from which abstractions were made when representing the data in tabular form. This document does not consistently document whether the data that support the original model also support the abstracted model (in the form of tabular data). Also, objective comparisons have not been made for all the constituent models, such as validating the colloidal transport model with data from the C-Wells Testing Complex. DOE audits of the Total System Performance Assessment Program have identified problems with the validation of models, and DOE has issued Corrective Action Report BSC–01–C–001

[5]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

(Bechtel SAIC Company, LLC, 2001) to address these problems. DOE has also agreed[6] to document the implementation of the process for model confidence building and demonstrate compliance with model confidence criteria in accordance with the applicable procedures.

The treatment of scenario and parameter uncertainty described in CRWMS M&O (2000a) appears to be appropriate. The approach outlined in CRWMS M&O (2000b) for determining the effect of alternative conceptual models on performance using sensitivity studies by weighting the results of the alternative conceptual models, based on the probability of the model being correct, or by demonstrating that one model is more conservative and using that one in the analysis is acceptable to NRC staff. NRC staff have concerns, however, that DOE has weighted the results of the alternative conceptual models based on the probability of the model being correct in Total System Performance Assessment–Site Recommendation (CRWMS M&O, 2000a) without an appropriate technical basis for assigning the weights to the alternative conceptual models. Additionally, it is not clear to NRC staff if DOE will analyze the effect of alternative conceptual models for more than one process at a time that may interact with each other and potentially have a greater effect on the results than either alternative conceptual model individually. The aforementioned approach (completing essentially a one-off replacement of conceptual model with an alternative model) leads to difficulties in determining which alternative conceptual models significantly impact risk and which ones do not. When many alternative conceptual models exist, the number of permutations for combinations of alternative conceptual models becomes large. To address these concerns, DOE agreed[7] to document the methodology used to incorporate alternative conceptual models into the performance assessment in such a manner that risk is not underestimated including the guidance given to process-level experts for treating alternative models.

The methodology outlined by DOE in CRWMS M&O (1999) for sampling parameter uncertainty seems to be reasonable. This use of Latin hypercube sampling permits parameters to be sampled across their ranges of uncertainty. This sampling is acceptable as long as a sufficient number of realizations is conducted to ensure the intervals, in which the range of uncertainty is divided, are not excessively large.

3.4.1.5 Status and Path Forward

Table 3.4.1-1 provides related DOE and NRC agreements pertaining to the analysis of the repository performance that demonstrate compliance with the postclosure individual protection standard. The status and the detailed agreements (or path forward) pertaining to all the key technical issue subissues are provided in Table 1.1-3 and Appendix A.

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[6]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[7]Ibid.

Repository Safety After Permanent Closure

The Total System Performance Assessment and Integration Key Technical Issue Subissue pertaining to the demonstration of the postclosure individual protection standard is considered closed-pending. Following is a summary of issues that DOE needs to resolve before this subissue can be closed.

Table 3.4.1-1. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreements*
Igneous Activity	—	—	All Agreements
Structural Deformation and Seismicity	—	—	All Agreements
Evolution of Near-Field Environment	—	—	All Agreements
Container Life and Source Term	—	—	All Agreements
Thermal Effects on Flow	—	—	All Agreements
Repository Design and Thermal-Mechanical Effects	—	—	All Agreements
Unsaturated and Saturated Flow Under Isothermal Conditions	—	—	All Agreements
Radionuclide Transport	—	—	All Agreements
Total System Performance Assessment and Integration	Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards	Closed-Pending	TSPAI.4.01 TSPAI.4.03 through TSPAI.4.07

*Related DOE and NRC agreements are associated with one or all five generic acceptance criteria.

3.4.1.6 References

Bechtel SAIC Company, LLC. “Root Cause Analysis Report for CAR BSC–01–C–001 and CAR BSC–01–C–002.” Revision 1. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2001.

CRWMS M&O. “Total System Performance Assessment–Site Recommendation Methods and Assumptions.” TDR–MGR–MD–000001. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 1999.

———. “Total System Performance Assessment for the Site Recommendation.” TDR–WIS–PA–000001. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000a.

———. “Repository Safety Strategy: Plan to Prepare the Postclosure Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations.” TDR–WIS–RL–000001. Revision 03. North Las Vegas, Nevada: DOE, Yucca Mountain Site Characterization Office. 2000b.

———. “Total System Performance Assessment Model for Site Recommendation.” MDL–WIS–PA–000002. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000c.

DOE. “Software Management.” Procedure AP-SI.1Q. Revision 03 ICN 01. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 2001.

Golder Associates. “Software Code: GoldSim.” 6.04.007. Redmond, Washington: Golder Associates. 2000.

Leigh, C.D., B.M. Thompson, J.E. Campbell, D.E. Longsine, R.A. Kennedy, and B.A. Napier. “User’s Guide for GENII-S: A Code for Statistical and Deterministic Simulation of Radiation Doses to Humans from Radionuclides in the Environment.” SAND91–056. Albuquerque, New Mexico: Sandia National Laboratories. 1993.

NRC. NUREG–1636, “Regulatory Perspectives on Model Validation in High-Level Radioactive Waste Management Programs: A Joint NRC/SKI White Paper.” Washington, DC: NRC. March 1999.

———. “Issue Resolution Status Report, Key Technical Issue: Total System Performance Assessment and Integration.” Washington, DC: NRC. 2000.

———. “NUREG–1804, “Yucca Mountain Review Plan—Draft Report for Comment.” Revision 2. Washington, DC: NRC. March 2002.

3.4.2 Demonstration of Compliance with the Human Intrusion Standard

3.4.2.1 Description of Issue

The Demonstration of Compliance with the Human Intrusion Standard section addresses the DOE approach for conducting a total system performance assessment of the effects of limited human intrusion on the repository system and, if necessary, demonstrates that the repository system is not substantially degraded as a result. Limited human intrusion, as detailed in 10 CFR 63.322, describes an event for which (i) a single groundwater exploration borehole is drilled through a degraded waste package and continues to the saturated zone, (ii) the borehole is not properly sealed and is assumed to degrade naturally, (iii) no waste material falls into the borehole, (iv) only exposure to radionuclides transported to the saturated zone by water is considered, and (v) unlikely natural processes and events are not considered. The overall organization and identification of all the integrated subissues are depicted in Figure 1.1-2. The DOE description and technical basis for analyzing performance in case of limited human intrusion are documented in the total system performance assessment and model reports for the site recommendation (CRWMS M&O, 2000a,b) and numerous supporting analysis and model reports. This chapter reviews the analysis of performance, in case of limited human intrusion, DOE incorporated in its total system performance assessment.

3.4.2.2 Relationship to Key Technical Issue Subissues

The Demonstration of Compliance with the Human Intrusion Standard section incorporates subject matter previously captured in the following key technical issue subissues:

- Total System Performance Assessment and Integration: Subissue 3—Model Abstraction (NRC, 2000)

- Total System Performance Assessment and Integration: Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards (NRC, 2000)

These key technical issue subissues formed the bases for the previous versions of the issue resolution status reports and also were the bases for technical exchanges with DOE where agreements were reached on the additional information DOE needed to provide to resolve the subissue. The resolution status of this integrated subissue is based on the resolution status of each of the contributing key technical issue subissues. The subsequent sections incorporate applicable portions of these key technical issue subissues, however, no effort was made to explicitly identify each subissue.

3.4.2.3 Importance to Postclosure Performance

One aspect of risk-informing the NRC review was to determine how this issue is related to the DOE repository safety strategy. Repository performance in case of limited human intrusion at Yucca Mountain is directly related to three of the principal factors DOE identified in the repository safety strategy (CRWMS M&O, 2000c)—seepage into emplacement drifts, radionuclide concentration limits in water, and radionuclide delay through the saturated zone. The DOE analyses indicate that the peak dose rate for human intrusion is most affected by the amount of seepage contacting the waste intersected by the borehole, radionuclide concentrations in this seepage, delay of radionuclide migration through the saturated zone, dilution of the radionuclide concentrations during pumping, and biosphere dose conversion factors for the groundwater related pathway (CRWMS M&O, 2000c). Note that 10 CFR 63.332 specifies the amount of water that can be pumped per year and, therefore, fixes the dilution rate of radionuclides.

3.4.2.4 Technical Basis

NRC used the acceptance criteria and review methods found in previous issue resolution status reports to develop the Yucca Mountain Review Plan (NRC, 2002). This section documents the review of DOE approaches for including analysis of performance in case of limited human intrusion in total system performance assessment abstractions. The review is organized according to three acceptance criteria: (i) Time of the Earliest Intrusion Event Is Technically Supported, (ii) Evaluation of an Intrusion Event Demonstrates the Annual Dose to the Reasonably Maximally Exposed Individual in Any Year During the Compliance Period Is Acceptable, and (iii) The Total System Performance Assessment Code Provides a Credible Representation of the Intrusion Event.

3.4.2.4.1 Time of the Earliest Intrusion Event Is Technically Supported

Overall, the current information is sufficient to conclude that the necessary information will be available at the time of a potential license application to evaluate the earliest time of an intrusion event.

Staff found the method for estimating the time of earliest intrusion presented in CRWMS M&O, (2000a) was generally satisfactory. The individual protection standard for human intrusion in 10 CFR 63.321 is a two-step process. The first step requires DOE to provide the analyses and technical bases used to determine the earliest time after disposal that the waste package would degrade sufficiently that a human intrusion could occur without recognition by the drillers. The second step, which will be covered in more detail in Section 3.4.2.4.2, requires that an assessment be performed if a waste package is projected to be penetrated at or before 10,000 years after disposal. The DOE approach presented in the Total System Performance Assessment–Site Recommendation assumed that the human intrusion occurred 100 years after closure of the repository. DOE stated that 100 years was used "   because it was considered to be conservative and because it was difficult to defensibly quantify a later intrusion time   ." Staff found that assuming the human intrusion event occurs 100 years after closure of the repository is conservative and acceptable. It should be noted, however, if DOE elects to modify this approach by using a different time of occurrence for the human intrusion event, DOE must provide, as required by 10 CFR 63.321, the analyses and technical bases used to justify the new time of occurrence.

3.4.2.4.2 Evaluation of an Intrusion Event Demonstrates the Annual Dose to the Reasonably Maximally Exposed Individual in Any Year During the Compliance Period Is Acceptable

Overall, the current information is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the adequacy of DOE demonstration that the annual dose to the reasonably maximally exposed individual in any year during the compliance period because of a human intrusion event is acceptable.

The methods presented in CRWMS M&O (2000a) for evaluating the annual dose to the reasonably maximally exposed individual in any year during the compliance period resulting from human intrusion were generally acceptable to allow information in a potential license application. DOE assumed the human intrusion event occurs 100 years after closure of the repository. Because the event is assumed to occur at or before 10,000 years after disposal, DOE is required by 10 CFR 63.321 to demonstrate there is a reasonable expectation that the reasonably maximally exposed individual receives no more than an annual dose of 0.15 mSv [15 mrem] as a result of human intrusion during the 10,000-year compliance period. DOE used its TSPA Code for this demonstration in CRWMS M&O (2000a).

3.4.2.4.3 The Total System Performance Assessment Code Provides a Credible Representation of the Intrusion Event

Overall, the current information, along with agreements reached between DOE and NRC, is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess whether the DOE TSPA Code provides a credible representation of the intrusion event. The methods presented in CRWMS M&O (2000b) for performing a total system performance assessment should provide a credible representation of the human intrusion event.

Any parameter and scenario description choices DOE made in developing an approach for human intrusion analysis must be justified. A few examples of scenario specifications that still must be justified include, but are not limited to water infiltration rates in the borehole, assumption of no gain or loss of water from or to the unsaturated zone, borehole dimensions, treatment of early-time vaporization, in-package temperature and chemistry, and credit for sorption in the unsaturated fault pathway. Other examples of where assumptions made in the analysis of the effects of human intrusion do not appear to be justified or appropriate, based on 10 CFR Part 63, were raised at the Total System Performance Assessment and Integration Technical Exchange[1] and follow:

- Volume and chemistry of drilling fluids are ignored in analysis.
- Rate of infiltration is unaffected by the presence of the borehole.

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[1]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

- Cladding in the penetrated waste package is perforated because of the event, but not completely failed.

- The properties of the rubblized borehole (porosity, fluid saturation, and dispersivity) are represented by the matrix properties of an unsaturated zone fault.

DOE responded that human intrusion inputs will be reevaluated after promulgation of final EPA, DOE, and NRC rules. This response is acceptable, and NRC expects the approach DOE selects for analysis of the limited human intrusion scenario will conform to 10 CFR Part 63. No specific agreement was generated for this comment.

DOE should ensure the results of the human intrusion analyses are consistent with other models in the DOE TSPA Code. The following apparent inconsistency was raised at the Total System Performance Assessment and Integration Technical Exchange.[2]

The peak expected dose resulting from human intrusion is shown to occur approximately 200 years after the single waste package is breached by drilling. This result suggests that the travel time in the saturated zone is extraordinarily short. Elsewhere in the Total System Performance Assessment–Site Recommendation technical document, it appears the three-dimensional saturated zone model predicts a median travel time for unretarded C-14 of approximately 600 years, whereas for slightly retarded Tc-99, the median travel time is around 1,000 to 1,500 years. These findings seem inconsistent.

DOE responded that the apparent inconsistency may be caused by comparison of time for mean peak dose from the calculation of human intrusion from the DOE TSPA Code in CRWMS M&O (2000a, Figure 4.4-11) to breakthrough times calculated using median inputs to the three-dimensional saturated zone model in CRWMS M&O (2000a, Figure 3.8-18). DOE noted the mean human intrusion dose is strongly dominated by the early breakthroughs, and the DOE TSPA Code median human intrusion dose peaks after 10,000 years, consistent with retardation of neptunium and plutonium. This response is acceptable, and NRC expects results of the human intrusion analyses are consistent with other models in the DOE TSPA Code. NRC further recommends that explanations be provided for cases where results do not appear consistent. No specific agreement was generated for this comment.

DOE should ensure human intrusion calculations are stable with respect to the number of realizations and timestepping used. This comment was raised at the Total System Performance Assessment and Integration Technical Exchange, August 6–10, 2001.[3]
DOE responded that 300 realizations have been conducted for human intrusion calculations. The calculations result in lower peak dose during the 10,000-year timeframe when compared with results using 100 realizations. Results using both 300 and 100 realizations are well below the current regulatory limit of 0.15 mSv [15 mrem]. DOE agreed the supporting

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[2]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[3]Ibid

basis for the number of realizations will be documented in the Total System Performance Assessment–License Application Technical Report and the rationale for timestepping in the Total System Performance Assessment–License Application Model Report. This response is acceptable, and NRC expects that technical bases will be provided to demonstrate the results are stable for the number of realizations and timestepping used. This comment is addressed by agreements TSPAI.4.03 and TSPAI.4.04, which deal with stability for the number of realizations and spatial and temporal discretization.

3.4.2.5 Status and Path Forward

Table 3.4.2-1 provides the status of all key technical issue subissues referenced in Section 3.4.2.2 for analysis of performance in case of limited human intrusion. The table also provides the related DOE and NRC agreements. The agreements listed in the table are associated with one or all of the acceptance criteria discussed in Section 3.4.2.4. Note that the status and the detailed agreements (or path forward) pertaining to all the key technical issue subissues are provided in Table 1.1-3 and Appendix A.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (through specified testing, analyses, and the like), acceptably addresses the NRC questions so that no information beyond that provided, or agreed to, will likely be required at the time of a potential license application.

The final approach DOE selected for analysis of the limited human intrusion scenario must conform to 10 CFR Part 63, and all scenario-specific assumptions will be justified. To meet the acceptance criteria, the DOE human intrusion analysis must (i) adequately support the selection of time of occurrence of the earliest human intrusion; (ii) be performed separately from the overall code used to conduct a total system performance assessment but be generally consistent with the code used to conduct a total system performance assessment; (iii) demonstrate that the calculations are stable; (iv) use calculations based on appropriate conceptual models and produce results that are reasonable and consistent with the available conceptual models and data; (v) show that the repository system meets NRC performance objectives; and (vi) ensure the code used to conduct a total system performance assessment provides a credible representation of the intrusion event with respect to consistent assumptions, code verification, estimate of uncertainty, and proper sampling methods.

Table 3.4.2-1. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreements*
Total System Performance Assessment and Integration	Subissue 3—Model Abstraction	Closed-Pending	None
	Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards	Closed-Pending	TSPAI.4.03 TSPAI.4.04

*Related DOE and NRC agreements are associated with one or all acceptance criteria.

3.4.2.6 References

CRWMS M&O. "Total System Performance Assessment for the Site Recommendation." TDR–WIS–PA–000001. Revision 00. Las Vegas, Nevada: TRW Environmental Safety Systems, Inc. 2000a.

———. "Total System Performance Assessment Model for Site Recommendation." MDL–WIS–PA–000002. Revision 00. Las Vegas, Nevada: TRW Environmental Safety Systems, Inc. 2000b.

———. "Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations." TDR–WIS–RL–000001. Revision 04. Las Vegas, Nevada: TRW Environmental Safety Systems, Inc. 2000c.

NRC. "Issue Resolution Status Report, Key Technical Issue: Total System Performance Assessment and Integration, Revision 3." Washington, DC: NRC. 2000.

———. NUREG–1804, "Yucca Mountain Review Plan—Draft Report for Comments." Revision 2. Washington, DC: NRC. March 2002.

3.4.3 Analysis of Repository Performance That Demonstrates Compliance with Separate Groundwater Protection Standards

Text in this section will be provided at a later date.

3.5 Status of Postclosure Issue Resolution and Path Forward

This section summarizes the status of postclosure issue resolution at the staff level. These results do not constitute a licensing review, and none of the agreements summarized here should be used to draw conclusions about whether the proposed Yucca Mountain site is likely to meet applicable NRC regulatory requirements for postclosure performance. The DOE and NRC agreements describe the information DOE agreed to provide and is needed to support an NRC licensing review. As previously noted, if DOE were to adopt a lower temperature operating mode or the approach used in Bechtel SAIC Company, LLC (2001a,b), NRC believes that more information would be needed for a potential license application.

The organization of the postclosure section follows the report structure previously used in NRC (2000), which consists of four parts: (i) System Description and Demonstration of Multiple Barriers, (ii) Scenario Analysis and Event Probability, (iii) Model Abstraction, and (iv) Demonstration of Compliance with the Postclosure Public Health and Environmental Standards. As described in Chapter 1, this approach was adopted to streamline the postclosure performance assessment review process and focus on those areas important to repository performance after permanent closure. The total repository system is divided into 14 integrated subissues or model abstractions (Figure 1.1-2), each of which is evaluated against 5 generic acceptance criteria. Historically, issue resolution activities have been conducted and documented on the basis of nine key technical issues.

In the issue resolution status reports for individual key technical issues, issue resolution was documented subissue by subissue. The nine key technical issues represent major processes and related staff concerns regarding the postclosure safety of a geologic repository. Some processes were shared among key technical issues, making discussion and resolution cumbersome. As the NRC and the CNWRA staffs conducted independent performance assessment exercises over the years and reviewed similar exercises by the U.S. Department of Energy Yucca Mountain Project, Electric Power Research Institute, the U.S. Department of Energy Waste Isolation Pilot Project, and other international programs, it became clear that a more integrated and transparent issue structure was needed.

To clarify the issue structure, charts were constructed to depict the components of a safety review (Figure 1.1-1) and the relationships among various components of a postclosure performance assessment for the proposed repository at Yucca Mountain (Figure 1.1-2). These charts showed that an efficient way to review the DOE postclosure safety case and its associated performance assessment is to follow the partitioning depicted in Figure 1.1-2. This partitioning is primarily based on the natural progress of potential radionuclide release and transport to a receptor group at the Yucca Mountain site. The topics at the most detailed level of decomposition (14 in all) in Figure 1.1-2 are called integrated subissues or model abstractions, mainly because each integrated subissue draws information from multiple key technical issues. The integrated subissues represent an interdisciplinary and logical approach to reviewing the DOE total system performance assessment. The integrated subissue format and the interdisciplinary questions posed for each of the integrated subissues assist the staff in more formally integrating the contribution of the key technical issue subissues. Therefore, it was decided to adopt this structure in developing the postclosure portions of the Yucca

Mountain Review Plan (NRC, 2002). NRC (2002) documents guidance to the staff for the review of any license application submitted by DOE. To create traceability and transparency through better correlation of current reviews with future reviews of the potential license application, the same structure is also followed for the postclosure portion of this document.

System Description and Demonstration of Multiple Barriers

The Total System Performance Assessment and Integration Key Technical Issue Subissue pertaining to System Description and Demonstration of Multiple Barriers was categorized as closed-pending at the staff level as a result of agreements[1] reached at the August 2001 technical exchange. When information identified in the agreements is adequate, DOE will have provided sufficient information for the staff to conduct a detailed review of the DOE license application with respect to its demonstration that the repository has multiple barriers. DOE has to provide information in response to two multiple-barrier-related agreements. DOE agreed to enhance the description of its approach for presenting and describing the capabilities of barriers, which NRC anticipates will include how DOE will use its performance assessment model to support assertions of barrier performance. The eventual approach that DOE decides to use when describing the capabilities of particular barriers will influence the amount of effort used to complete the multiple-barrier-related agreements. Satisfying the agreements would not require DOE to conduct further site studies, however. The staff understanding is that DOE will be extending its Total System Performance Assessment–Site Recommendation analyses to address these agreements and, consequently, the staff anticipate that fulfilling these agreements may involve a minor level of effort.

Scenario Analysis and Event Probability

The Total System Performance Assessment and Integration Key Technical Issue Subissue pertaining to Scenario Analysis and Event Probability was categorized as closed-pending at the staff level as a result of agreements[2] reached at the August 2001 technical exchange between DOE and NRC. Presently, it appears DOE will have sufficient information on (i) the features, events, and processes considered for the total system performance assessment; (ii) the technical basis for including or excluding each feature, event, or process in the dose assessment; (iii) the formation and screening of scenario classes; and (iv) the treatment of events with a probability greater than one chance in 10,000 in 10,000 years for NRC to make a regulatory decision on receipt of any potential license application. DOE agreed to revise its process of defining, describing, and screening features, events, and processes to ensure that the process is comprehensive and that NRC can audit it. DOE has flexibility in how it addresses the NRC staff questions raised during the May and August 2001 technical exchanges. The effort that may be needed to provide the information described in the agreements will depend on the DOE approach. Changes in the scope of a feature, event, or process could affect the documentation of the technical basis used to include or exclude it from

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[1] Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[2] Ibid.

the dose assessment. In some instances, additional analyses or data may be necessary to provide a technical basis for excluding a feature, event, or process from the dose assessment. In addition, information and analyses used to support these technical bases will be developed in response to other, linked, agreements. Consequently, the staff anticipate a moderate level of effort will be needed for DOE to provide the information necessary to fulfill the scenario-analysis-related agreements.

Model Abstraction

The Total System Performance Assessment and Integration Key Technical Issue Subissue pertaining to Model Abstraction was categorized as closed-pending at the staff level as a result of agreements[3] on performance assessment methods reached at the August 2001 technical exchange, as well as agreements reached at previous technical exchanges related to the other key technical issues. Presently, it appears DOE will have sufficient information to demonstrate (i) each model abstraction is adequately described and properly integrated with other model abstractions, (ii) the data are sufficient to justify each model abstraction, (iii) data uncertainty is properly characterized and propagated through each model abstraction, (iv) model uncertainty is properly characterized and propagated through each model abstraction, and (v) the output from each model abstraction is supported by objective comparison to confirmatory data. The information DOE agreed to provide to meet criteria (i)–(v) for each model abstraction is described in Sections 3.3.1–3.3.14.

Demonstration of Compliance with the Postclosure Public Health and Environmental Standards

The Total System Performance Assessment and Integration Key Technical Issue Subissue pertaining to Demonstration of Compliance with the Postclosure Public Health and Environmental Standards was categorized as closed-pending at the staff level as a result of agreements[4] reached at the August 2001 technical exchange. Presently, it appears DOE will have enough information about the methods to compute an accurate and stable estimate of the peak mean dose for 10,000 years, thus enabling NRC to make a regulatory decision on receipt of any potential license application. DOE must satisfy the terms of seven agreements to completely close this subissue. Although none of these seven agreements explicitly requires DOE to collect additional data, fulfilling the terms of the requirements, which include conducting new stability analysis, should require a moderate level of effort.

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[3] Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[4] Ibid.

3.5.1 References

Bechtel SAIC Company, LLC. "FY01 Supplemental Science and Performance Analyses." Vol. 1: Scientific Bases and Analyses. TDR–MGR–MD–000007. Revision 00 ICN 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2001a.

Bechtel SAIC Company, LLC. "FY01 Supplemental Science and Performance Analyses." Vol. 2: Performance Analyses. TDR–MGR–PA–000001. Revision 00. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2001b.

NRC. "Issue Resolution Status Report, Key Technical Issue: Total System Performance Assessment and Integration." Revision 3. Washington, DC: NRC. 2000.

———. NUREG–1804, "Yucca Mountain Review Plan—Draft Report for Comment." Revision 2. Washington, DC: NRC. March 2002.

4 PERFORMANCE CONFIRMATION

4.1 Research and Development Program to Resolve Safety Questions

4.1.1 Description of Issue

Requirements for the content of the license application at 10 CFR 63.21(c)(10) specify that DOE identifies those structures, systems, and components of the geologic repository, both surface and subsurface, that require research and development to confirm the adequacy of design. This requirement also specifies that for structures, systems, and components important to safety and for the engineered and natural barriers important to waste isolation, DOE shall provide a detailed description of the programs designed to resolve safety questions, including a schedule indicating when these questions would be resolved.

DOE cannot provide schedules and detailed descriptions of research and development programs to resolve safety questions for either structures, systems, and components important to safety or engineered and natural barriers important to waste isolation until the safety questions have been identified. Unresolved safety questions are likely to be associated with other topics discussed in this Integrated Issue Resolution Status Report. It is premature to identify these questions until DOE has presented its safety case in a license application for construction authorization.

NRC staff will evaluate any safety questions, and the schedules and descriptions of the research and development programs to resolve them, using review methods and acceptance criteria in NRC (2002. This review, and staff knowledge of the status of open item issue resolution, could result in identification of additional safety questions. These additional safety questions would require DOE to define additional acceptable research and development programs before NRC could approve a construction authorization.

Because assessment of safety questions is premature as of the writing of this report, no specific concerns have been defined.

4.1.2 Relationship to Key Technical Issue Subissues

Specific topics for the research and development programs to resolve safety questions will not be identified until DOE has completed its safety case to support the license application for construction authorization. NRC staff expect that any such safety issues are likely to derive from existing integrated subissues that may not be adequately resolved at the time of a license application. It is also possible that safety questions that have not yet been identified will evolve before submission of a license application.

4.1.3 Importance to Safety and Postclosure Performance

Any safety question, by definition, is important to safety or to waste isolation. The degree of significance of any specific safety question will be evaluated on the basis of risk insights and information gained throughout the prelicensing consultation period. The degree of safety significance also will be considered in determining the adequacy of any proposed research and development program. The integrated safety significance of all safety questions must be taken into account when the staff decide whether it is appropriate to approve a construction authorization.

4.1.4 Technical Basis

Because safety questions and their associated research and development programs have not yet been presented in a license application, there is no technical basis to evaluate. A generic approach for the review of any such concerns and programs will be provided in NRC (2002).

4.1.5 Status and Path Forward

No safety questions have yet been identified. Consequently, the associated research and development programs have not been developed.

When the license application for construction authorization is submitted, the NRC staff will evaluate the research and development programs for any safety questions using an approach that will be included in NRC (2002).

4.1.6 Reference

NRC. NUREG–1804, "Yucca Mountain Review Plan—Draft Report for Comment." Revision 2. Washington, DC: NRC. March 2002.

4.2 Performance Confirmation Program

4.2.1 Description of Issue

Performance confirmation is the program of tests, experiments, and analyses to evaluate the adequacy of the information used to determine that the performance objectives for the facility will be met. The Performance Confirmation Program begins during site characterization and continues until permanent closure of the repository. DOE will conduct a Performance Confirmation Program to confirm the assumptions, data, and analyses that support the performance assessment and any findings, based thereon, that permitted construction of the repository and subsequent emplacement of the wastes. Key geologic, hydrologic, geomechanical, and other physical parameters will be monitored to detect any significant changes in the conditions assumed in the performance assessment that may affect compliance with the performance objectives.

4.2.2 Importance to Safety and Postclosure Performance

The DOE Performance Confirmation Program is intended to address the full range of safety issues described elsewhere in this report. Many of those safety issues have substantial uncertainties, especially those issues related to meeting long-term system performance objectives. The responses of the engineered and natural system barriers to activities conducted during waste emplacement and as a result of waste emplacement are to be evaluated using the Performance Confirmation Program, during an extended operating period, to discover any negative effects on the safety of the repository. Conduct of the Performance Confirmation Program is therefore an important part of the DOE repository safety case. Specifically, performance confirmation is identified in Revision 4 of the DOE Repository Safety Strategy as one of five elements of the planned DOE postclosure safety case (CRWMS M&O, 2000a).

4.2.3 Status and Path Forward

DOE published CRWMS M&O (2000b), presenting its current plans for test and evaluation activities, including predicting test outcomes, conducting *in-situ* and laboratory tests, analyzing test data, and modeling and evaluating test results. The staff understand that DOE will update this Performance Confirmation Plan when new information becomes available. DOE activities conducted to date as part of site characterization have begun to establish baseline information against which future repository performance can be evaluated. DOE anticipates that the transition from baseline development to monitoring and modeling the performance effects of changes from baseline conditions will occur after submittal of the site recommendation report and before emplacement of waste in the repository. The staff will review in detail the DOE Performance Confirmation Plan subsequent to the DOE completion of its planned revision of the Performance Confirmation Plan.

Performance Confirmation

DOE plans these steps to accomplish the Performance Confirmation Program (CRWMS M&O, 2000b):

(1) Identify performance confirmation factors and parameters: Identify the factors (processes) and related parameters important to postclosure safety that should be monitored as part of performance confirmation.

(2) Establish the performance confirmation database and predict performance: Establish the database from site characterization efforts and identify the analytical process models and performance assessment models to be used to predict and evaluate performance. Using this basis, predict the expected preclosure values and variations of these values.

(3) Establish tolerances and bounds: Establish tolerances or acceptable limits (screening levels) of deviations from predicted performance, including acceptable ranges of key parameter values, regulatory limits, and model validity or credibility limits. Analyses are to address expected changes as a result of construction, operations, and waste emplacement.

(4) Establish completion criteria and guidelines for corrective actions: Establish criteria and guidelines for completing an activity and for evaluating conditions outside of tolerance, as well as identify and recommend corrective actions to be taken in these cases.

(5) Plan and set up the performance confirmation test and monitoring program: Conduct detailed planning, construct the test/monitoring facilities, and set up instrumentation necessary for the Performance Confirmation Program, including establishment of the ambient baseline, if necessary.

(6) Monitor, test, and collect data: Perform the testing and monitoring activities necessary to collect data in accordance with applicable regulations and quality assurance requirements.

(7) Analyze, evaluate, and assess data: Analyze and evaluate performance confirmation data against the performance confirmation baseline, including conducting statistical tests and trend analyses. When changes occur in the predicted construction and operation sequencing, total system performance assessments will be conducted as necessary to assess the impact of these changes on the activity baseline.

(8) Recommend and implement corrective actions (if required): Identify, recommend, and (if necessary) implement corrective action if data or data trends exceed (or are expected to exceed) the prescribed bounds. If data stay within prescribed bounds, continue to perform periodic evaluations against completion criteria to determine whether to continue the test operation or stop the monitoring.

The current version of CRWMS M&O (2000b) is based on Revision 3 of CRWMS M&O (2000c). DOE is expected to update the principal safety factors for CRWMS M&O (2000b) when future versions of the CRWMS M&O (2000c) are produced.

4.2.4 References

CRWMS M&O. "Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations." DRL–WIS–RL–000001. Revision 04. Las Vegas, Nevada: CRWMS M&O. 2000a.

———. "Performance Confirmation Plan." TDR–PCS–SE–000001. Revision 01 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000b.

———. "Repository Safety Strategy: Plan to Prepare the Postclosure Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations." TDR–WIS–RL–000001. Revision 03. Las Vegas, Nevada: CRWMS M&O. 2000c.

5 ADMINISTRATIVE AND PROGRAMMATIC REQUIREMENTS

5.1 Quality Assurance Program

The following is based on the status of the DOE quality assurance program.

5.1.1 Background

In late 1998 and early 1999, DOE identified significant deficiencies in the implementation of its quality assurance program in the following areas: (i) procurement (qualification of suppliers and the use of unqualified sources), (ii) model development (inadequate technical review and collection of data and documentation of data collection in scientific notebooks), and (iii) software development (inadequate identification and implementation of software controls). As a result of these deficiencies, DOE implemented a corrective action plan. The two major elements of this corrective action plan required that the quality of all data and software developed before June 1999 be reverified and that procedures controlling the areas where deficiencies were identified be revised to provide adequate controls to ensure the quality assurance program is effectively implemented. Further, all personnel supporting site characterization activities received extensive training in the regulatory and licensing processes.

During fiscal years 2000 and 2001, staff reviewed the implementation of the DOE corrective action plan, including data and software qualification, by (i) observing several DOE performance-based audits; (ii) using daily overviews performed by the NRC onsite representatives assigned to the Yucca Mountain Project Office in Las Vegas, Nevada; and (iii) addressing concerns and progress with DOE during technical exchanges and management meetings.

DOE also stated that it will submit a comprehensive corrective action plan to address the causes of problems. This plan will consider and address items such as (i) results of DOE reviews of the documents supporting the site recommendation and a potential license application; (ii) root cause analysis for the various quality assurance problems; (iii) lessons learned from past corrective action plans; (iv) accountability; (v) performance measures; (vi) upgrading and enhancing procedures; and (vii) audits, surveillances, self assessments, and management oversight to confirm that the corrective actions are being implemented and are effective.

NRC reviewed DOE (2002), which is intended to address the items described in the previous paragraph. This document did not meet NRC expectations and the DOE committed to revise the document to address NRC concerns.

The following paragraphs provide additional information on the progress DOE has made in implementing its corrective action plan and addressing quality assurance issues relating to DOE documentation supporting the site recommendation and a potential license application.

5.1.2 Staff Oversight of the DOE Quality Assurance Program

Before May 2000, the staff observed several DOE performance-based audits of analysis and model reports and related process model reports. In February 2000, the DOE Office of Quality

Assurance suspended audits of analysis and model reports and process model reports because (i) deficiencies were repeatedly identified in the areas of procedure compliance and software control, (ii) recommendations and lessons learned from the audits were not being effectively communicated to and implemented by the preparers of the analysis and model reports, and (iii) scheduled completion dates for some of the analysis and model reports were being postponed. Staff considered the actions of the DOE Office of Quality Assurance to delay the remaining audits appropriate.

In July 2000, DOE resumed auditing of analysis and model reports and process model reports. Some of the audits yielded no significant findings and indicated improvement in the technical quality and completeness of analysis and model reports and process model reports. Other audits, however, revealed that problems continued in the area of procedure compliance and some analysis and model reports and process model reports contained insufficient detail for documenting the bases for certain assumptions, inputs, and equations. During the April 17, 2001, DOE and NRC Quarterly Quality Assurance Breakout Session Meeting, DOE reported that it was evaluating the results of the recurring problems identified during the audits and determined that improvements were needed to clearly document model validation and the qualification of software routines and macros.[1,2] Further, DOE stated that it was investigating whether there was a significant condition adverse to quality regarding traceability and transparency of documentation supporting analysis and model reports and process model reports.

During prelicensing interactions in 2001, DOE discussed the results of its reviews to verify the quality of the documents supporting the site recommendation, including the Yucca Mountain Science and Engineering Report, the Total System Performance Assessment for the Site Recommendation, and the FY01 Supplemental Science and Performance Analyses. DOE performed vertical, horizontal, and technical reviews of these documents using, in some cases, personnel independent of the Yucca Mountain project. DOE also used independent personnel to perform an analysis for determining the root causes of the errors found in these documents. Although the NRC staff have not independently verified them, the staff believes that the performance of the reviews by DOE was necessary and appropriate to verify the quality of the documents supporting the site recommendation. It appears to the NRC staff that the reviews did not reveal any significant errors or problems that would impact the conclusions in the Total System Performance Assessment for the Site Recommendation portion of the site recommendation.

Although DOE has not yet fully qualified data and software used in the CRWMS M&O (2000) portion of the site recommendation, it has a reasonable approach to do so. Further, DOE indicated that if the information contained in Bechtel SAIC Company, LLC (2001a,b) is used to support or be a part of a potential license application, the information would be fully qualified and subjected to the same qualification controls as used for CRWMS M&O (2000). The staff

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[1]Reamer, C.W. "Minutes of the April 17, 2001, Quality Assurance and Key Technical Issue Status Management Meeting." Letter (August 20) to R. Clark, DOE. Washington, DC: NRC. 2001.

[2]Reamer, C.W. "Minutes of the April 18, 2001, Management Meeting." Letter (June 20) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

accept the DOE commitment to fully qualify all data, software, and models if used in a potential license application.

5.1.3 Implementation of Corrective Action

DOE has made significant progress in implementing appropriate corrective action to address its quality assurance problems. DOE responded to and completed the required corrective action requests documenting the 1998 and 1999 significant conditions adverse to quality except for the corrective action to confirm the adequacy of data and software qualified before June 1999.

In September 1999, DOE notified NRC of its goal to have 80 percent of all data fully qualified by mid-January 2001. To meet this goal, a graded approach was applied to the reverification of data collected before June 1999. This graded approach was based on the risk significance of the data. At that time, DOE also committed to have 100 percent of all data and software fully qualified by the time of a potential license application. DOE met its mid-January 2001 data qualification goal of 80 percent. As of September 6, 2001, DOE had qualified 94 percent of the data and 98 percent of the software supporting a potential license application. During the April 17, 2001, DOE and NRC Quarterly Quality Assurance Breakout Session Meeting, DOE reported that its goal was to have all data fully qualified by the time of the site recommendation and all software fully qualified by the time of a potential license application.

The staff will continue to observe DOE audits and discuss quality assurance problems and corrective actions with DOE. Also, the NRC onsite representatives will continue to routinely interact with DOE and its management and operating contractor to increase confidence that DOE is satisfactorily implementing the required corrective actions to address past and present quality assurance problems.

5.1.4 Conclusions

The DOE corrective action plan elements and approach appear reasonable. Although DOE has had problems implementing previous corrective action plans, DOE has made progress in implementing appropriate corrective actions to address identified quality assurance problems. Problems that have arisen since January 2001, however, indicate DOE needs to improve the implementation of its quality assurance program, especially in the areas of software control, model validation, and accuracy of information provided in DOE reports [e.g., CRWMS M&O (2000)]. Adherence to procedures and attention to detail in the preparation, independent review, and issuance of DOE documents continue to require improvement.

DOE has not yet fully qualified all the data and software needed for a potential license application, but appears to have a reasonable approach to do so by the time of a potential license application. If the data and software supporting a potential license application are fully qualified before any such license application, as agreed, there will be sufficient basis for NRC to conduct its licensing review. Taking into consideration the progress made to date and the current DOE schedule, DOE should be able to complete the qualification of data and software by the time of a potential license application.

5.1.5 References

Bechtel SAIC Company, LLC. "FY01 Supplemental Science and Performance Analyses." Vol. 1: Scientific Bases and Analyses. TDR–MGR–MD–000007. Revision 00 ICN 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2001a.

———. "FY01 Supplemental Science and Performance Analyses." Vol. 2: Performance Analyses. TDR–MGR–PA–000001. Revision 00. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2001b.

CRWMS M&O. "Total System Performance Assessment for the Site Recommendation." TDR–WIS–PA–000001. Revision 00 ICN 01. Las Vegas, Nevada: TRW Environmental Safety Systems, Inc. 2000.

DOE. "Office of Civilian Radioactive Waste Management (OCRWM) Management Improvement Initiatives." Revision 00. Las Vegas, Nevada: U.S. Department of Energy, Office of Civilian Radioactive Waste Management. 2002.

5.2 Records, Reports, Tests, and Inspections

Text in this section will be provided at a later date.

5.3 Training and Certification of Personnel

Text in this section will be provided at a later date.

5.4 Expert Elicitation

5.4.1 Description of Issue

Nearly every aspect of site characterization, design, and performance assessment will involve significant uncertainties. The primary method to evaluate and, to the extent practical, reduce these uncertainties should be through collection of sufficient data and information during site characterization. Factors such as temporal and spatial variations in the data, the possibility for multiple interpretations of the same data, and the absence of validated theories for predicting the performance of a repository for thousands of years, however, will result in some residual uncertainty. Consequently, the staff anticipate it will be necessary to complement and supplement the data obtained during site characterization with the interpretations and subjective judgments of technical experts (i.e., expert elicitation) as well as to conduct confirmatory testing and analyses during and after construction, should NRC authorize construction.

In the review process, NRC traditionally accepted expert elicitation to evaluate and interpret the factual bases of license applications. Thus, NRC is to give appropriate consideration to the judgments of DOE experts on a possible geologic repository at Yucca Mountain. Such consideration, however, envisions DOE using expert elicitation to complement and supplement more objective sources of scientific and technical information, such as data collection, analyses, and experimentation. The NRC staff believe formal elicitation procedures, used prudently and appropriately, can help ensure the expert elicitations are well documented, and the technical reasoning used to reach those judgments is open and traceable for independent review. If conducted optimally, formal elicitation can reveal a wide range of scientific and technical interpretations, thereby exposing (and possibly quantifying) the uncertainties in estimates concerning repository siting, design, and performance attributable to limitations in the state of technical knowledge. Formal procedures may also help groups of experts resolve differences in their estimates by providing a common scale of measurement and a common vocabulary for expressing their judgments.

5.4.2 Background

Recognizing that DOE intended to use expert elicitation in its geologic repository program, the NRC completed work, in late 1996, on its Branch Technical Position on the Use of Expert Elicitation in the High-Level Waste Program. This document, designated NUREG–1563 (NRC, 1996), provides general guidelines on those circumstances that may warrant the use of a formal process for obtaining the judgments of more than one expert (i.e., expert elicitation) and describes acceptable procedures for conducting expert elicitation, when formally elicited judgments are used to support a demonstration of compliance with NRC geologic repository disposal regulation. At the time, DOE was independently developing its own internal guidance on the use of expert elicitation. As part of the public comment process, however, DOE reviewed the Branch Technical Position and noted that it is in substantial agreement with staff technical positions. Moreover, DOE committed[1,2,3] to modify its internal procedures to be consistent with the Branch Technical Position and to follow this document in any formal elicitations DOE conducts for Yucca Mountain.

There are no precise criteria for determining when an expert elicitation should be undertaken. To implement the risk-informed performance-based approach, the language in 10 CFR Part 63 is intentionally nonprescriptive; that is, it leaves to DOE the opportunity and responsibility to determine how best to design any potential geologic repository at Yucca Mountain. Typically, programmatic concerns (timing, cost, and compliance demonstration requirements) will have a major influence on when the repository developer (DOE) uses an expert elicitation or gathers additional objective information. For example, programmatic concerns dominate the choices of (i) gathering additional field or laboratory data, (ii) undertaking additional theoretical analyses, (iii) using expert elicitation, or (iv) altering the compliance demonstration strategy, to lessen or eliminate the need to resolve a particular issue. Thus, DOE is responsible for determining a data-information-gathering approach, as long as an effective demonstration of compliance with the regulations can be made.

Consequently, DOE has the flexibility to determine if the costs and benefits of performing an expert elicitation are advantageous when compared with the costs and benefits of performing theoretical analyses, gathering additional field and experimental data, or both. As noted in NUREG–1563 (NRC, 1996), "... the use of expert elicitation should not be considered as an acceptable substitute for traditional analyses based on adequate field or experimental data, when such data are reasonably available or obtainable, or the analyses are practicable to perform .... ." Moreover, the guidance also states that adherence to the Branch Technical Position does not guarantee the specific technical conclusions will be accepted and adopted by the staff, an independent Licensing Board, NRC itself, or any other party to a potential high-level waste licensing proceeding. Rigid adherence to a sound elicitation process, in and of itself, does not guarantee the resulting judgments will be sufficient to satisfy the applicant burden of proof regarding the substantive issues addressed by the elicitation. Conversely, expert elicitation obtained through an evidently flawed or poorly documented process will not be adequate to support demonstrations of compliance.

5.4.3 Staff Oversight of DOE Use of Expert Elicitation

For years, the use of expert elicitation supported DOE incremental (DOE, 1998) decisionmaking related to determining the suitability of the Yucca Mountain site. DOE used expert elicitation to

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[1]Austin, J.H. "Issue Resolution for Site Characterization Analysis Comment 3 and Other Open Items Related to the Use of Expert Judgment." Letter (December 26) to R.A. Milner, DOE, Office of Civilian Radioactive Waste Management. Washington, DC: NRC, Division of Waste Management. 1996.

[2]Brocoum, S.J. "Resolution of U.S. Nuclear Regulatory Commission Site Characterization Analysis Comment 3 and Other Comments Related to the Use of Expert Elicitations in the High-Level Waste Program." Letter (August 6) to M. Bell, NRC. Washington DC: NRC. 1997.

[3]Bell, M.J. "U.S. Department of Energy Proposals to Implement Appendix E to NUREG–1563—Resolution of Site Characterization Analysis Comment 3." Letter (February 12) to S. Brocoum, DOE. Washington, DC: NRC. 1998.

resolve important performance issues, such as volcanism, and to select parameter distribution for the Total System Performance Assessment–Viability Assessment (TRW Environmental Safety Systems, Inc., 1996) [i.e., understanding unsaturated/saturated flow, defining waste form degradation and radionuclide mobilization modes, explaining near-field/altered-zone coupled effects, and determining sorption coefficient (k_d)].

NRC and CNWRA staffs have observed most DOE-sponsored formal elicitations. Furthermore, the Branch Technical Position was under development at the time DOE began elicitations on volcanism and seismic hazard. DOE now only relies on the use of expert elicitation in the areas noted in the next sections.

5.4.3.1 Probabilistic Volcanic Hazards Analysis

Major silicic volcanic eruptions have not occurred in southern Nevada in the last 10 million years. There is evidence, however, of lesser-magnitude basaltic volcanic activity in the Yucca Mountain area during this period, with activity at the Lathrop Wells cone—approximately 15 km [9.3 mi] southwest of the proposed repository site—possibly occurring as recently as 80,000 years ago. Because of the potentially undesirable consequences of a low-probability disruptive event, volcanism has been intensely investigated and debated for the last two decades. The uncertainties include

- Age of the most recent volcanism
- Mode of volcanic activity
- Structural control of past and future volcanic activity
- Adequacy of probabilistic models of volcanic activity
- Sufficiency of existing data for reliable probabilistic estimates of the volcanic hazard

There are no generally accepted methodologies for calculating the probability of future igneous activity during the regulatory period of interest. In addition, more than one conceptual model can be applied to this problem, resulting in a wide range of probability values. In an attempt to address the areas of controversy as well as to establish a credible basis for probabilistic calculations that could be used to assess the potential impact of volcanism on repository performance, DOE assembled 10 experts and conducted expert elicitations between 1995 and 1997. The elicitation process consisted of four workshops and two field trips to the Yucca Mountain site. The resulting elicitation, documented in Geomatrix Consultants (1996), evaluated a range of probability models, estimated uncertainties in model results caused by reasonable variations in model parameters, and determined a probability distribution for use in performance assessment models for Yucca Mountain. NRC and CNWRA staffs observed the expert elicitation workshops and reviewed the information developed through the documentation process and found it generally sufficient to use in a potential Yucca Mountain license application. Overall, DOE adequately justified the need for the elicitation and generally conducted the elicitation in accordance with the guidance set forth in NUREG–1563 (NRC, 1996).

Nevertheless, as explained in Section 3.2.2.4.1 of this document, the staff performed a technical review of the Geomatrix Consultants (1996) and have several technical concerns regarding these results and their application in the Yucca Mountain program.

Administrative and Programmatic Requirements

As a result of the various concerns, NRC reached two agreements with DOE.[4] Hence, the probability subissue is considered closed-pending. In the first agreement, DOE will include, in any possible site recommendation and possible license application, for information purposes, the results of a single-point sensitivity analysis for extrusive and intrusive igneous processes at a probability of 10^{-7}/yr—a value supported by the NRC staff. This analysis has been previously presented in such documents as Bechtel SAIC Company LLC (2001a Figure 4.3-1). In addition, at the August 2000 Igneous Activity Technical Exchange,[5] it was indicated that a new aeromagnetic survey had been undertaken for the site area. In the second agreement, DOE agreed to examine the results of this new survey for potential unrecognized buried igneous features and to evaluate the effect of these features on the Geomatrix Consultants (1996) probability estimates.

5.4.3.2 Probabilistic Seismic Hazards Analysis

DOE developed comprehensive probabilistic seismic and faulting hazard assessments necessary to characterize the potential seismic and faulting hazards at Yucca Mountain. The approach was similar to that suggested for a Level 4 Probabilistic Seismic Hazard Assessment, as defined in Budnitz, et al. (1997). The Level 4 Probabilistic Seismic Hazard Assessment includes the use of expert elicitation. Because of the limited availability of sufficient strong motion data and uncertainties in the seismologic characteristics of the Yucca Mountain site and region, DOE convened two expert panels. One panel was to evaluate the seismic source characterization. The other panel was to develop probabilistic models for ground-motion attenuation specific to the regional conditions of the western Basin and Range in proximity to Yucca Mountain. In the context of these circumstances, the use of an expert elicitation process was reasonable and appropriate.

Development of Budnitz, et al. (1997) followed a methodology first proposed by Cornell (1986) and McGuire (1976) and used a modified version of the FRISK88 computer code (Risk Engineering Inc., 1998). Within this approach, uncertainties were propagated through the analyses, and the results were presented as mean, median, and fractile hazard curves that incorporate uncertainties in the input parameters.

5.4.3.2.1 Seismic Source and Fault Displacement Characterization

For this elicitation, DOE assembled 18 experts, divided into 6 expert teams, and held 6 elicitation workshops between 1995 and 1998 (CRWMS M&O, 1998a). In addition to developing earthquake and ground-motion hazard assessments, the seismic source zone characterization experts also were to develop fault-specific probabilistic fault displacement hazards. These fault displacement hazard assessments used an approach similar to the one used in the seismic source zone characterization. Technical details of aspects of the seismic

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[4]Schlueter, J. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (August 29–31, 2000)." Letter (October 23) to S. Brocoum, DOE. Washington DC: NRC. 2000.

[5]Ibid.

and fault displacement hazard results are provided in Section 3.3.2.4.2, Faulting, and Section 3.3.2.4.3, Seismicity, of this Integrated Resolution Status Report.

The staff reviewed the information developed by DOE through the documentation process on fault displacement and seismic source zone characterization (CRWMS M&O, 1998a) and found it sufficient to use in a potential Yucca Mountain license application. DOE adequately justified the need for the elicitation and conducted the elicitation in accordance with the guidance set forth in NUREG–1563 (NRC, 1996).

5.4.3.2.2 Ground-Motion Attenuation

DOE assembled seven experts for the ground-motion elicitation, and the elicitation process was conducted in parallel with that of the seismic source zone elicitation. The ground-motion experts were to provide input (e.g., data, scientific interpretations, and estimates of parameter uncertainties) for developing the probabilistic ground-motion attenuation model (i.e., mathematical relationships between ground-motion and earthquake magnitude, distance, site conditions, and style of faulting). Unlike seismic source characterization, experts for this elicitation team were asked to provide intermediary results that were then used to develop the final Probabilistic Seismic Hazard Assessment (Budnitz, et al., 1997) ground-motion relationships. The seven experts each developed a probabilistic ground-motion attenuation model. These models were subsequently aggregated to (probabilistically) represent the current state of knowledge with regard to ground motions possible at the Yucca Mountain site due to earthquake phenomena. Technical details of aspects of the ground-motion attenuation results are provided in Section 3.3.2.4.3, Seismicity, of this Integrated Resolution Status Report.

The staff reviewed the information developed by DOE through the documentation process on ground-motion attenuation (CRWMS M&O, 1998b) and found it insufficient to use in a potential Yucca Mountain license application (subject to the agreement described in Section 5.4.5, Status and Path Forward of this report). The staff review concluded that, although DOE adequately justified the need for elicitation in this area, DOE did not conduct the elicitation in accordance with the guidance set forth in NUREG–1563 (NRC, 1996), particularly as it relates to the documentation provision of the elicitation process itself. Specifically, DOE has not provided documentation demonstrating that the ground-motion experts clearly understood the implications of their ground-motion parameter inputs, (part of postelicitation feedback) which are necessary to the ground-motion model development process. This postelicitation feedback information is necessary to verify the technical integrity of the elicitation process as well as the traceability of the assessment itself. Consequently, the absence of post-elicitation feedback documentation diminishes the acceptability and credibility of the elicitation results themselves because the process at present does not appear to be transparent and traceable.

For example, the staff independent review of the elicited ground-motion models for Yucca Mountain raised questions about the scientific basis for several of the individual expert ground-motion assessments as well as completeness of the elicitation feedback process itself. In particular, examination of several of the ground-motion models illustrated that a large range of unexplained differences exists between the experts inputs regarding predicted ground-motions and epistemic and aleatory uncertainties. In some instances, the staff noted wide differences between experts and large variability within individual expert models. The issues of proper feedback and documentation are especially crucial to the ground-motion part of Budnitz, et al. (1997) because the nature of this elicitation is the expectation that the experts will support the Probabilistic Seismic Hazard Assessment results. In the ground-motion elicitation, the experts provided intermediate results that were subsequently used by the technical facilitator/integrator to develop seven ground-motion attenuation models. The seven ground-motion attenuation models were then used to develop the curves for use in Probabilistic Seismic Hazard Assessment.

Although comparable to the generalist typically used to conduct an expert elicitation (Meyer and Booker, 1990), the role of the technical facilitator or integrator, as defined by the Senior Seismic Hazard Analysis Committee methodology (Budnitz, et al., 1997, pp. 29–48), has greater authority with the elicitation process and results. The NRC staff have expressed concerns to DOE about the potential overreaching authority of the technical facilitator or integrator in the elicitation process.[6] NRC staff concerns remain despite DOE assurances to the contrary.[7]

The staff independently examined the basis for the elicited ground-motion attenuation models and results and identified several questions about the DOE postelicitation feedback/ documentation process (CRWMS M&O, 1998b). At the October 2000 Technical Exchange on Structural Deformation and Seismicity,[8] DOE provided a brief summary of the elicitation approach used in the ground-motion portion of the Probabilistic Seismic Hazard Assessment. As a result of the staff questions after this presentation, DOE agreed[9] to provide additional documentation describing the process used to elicit the ground-motion attenuation models. In a letter dated December 21, 2000, DOE provided information it believed was responsive to the agreement made with the staff in October 2000.[10] After a review of this new submittal, staff concluded that most of the information provided was already available and, therefore, did not materially contribute to the closure of this particular issue. Nevertheless, based on the October 2000 technical exchange and follow-on discussion, it appears DOE will provide the requested and necessary documentation before submission of a potential license application for the proposed Yucca Mountain repository. Thus, this issue between DOE and NRC in this area is considered closed-pending.

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[6]Austin, J.H. "Implementation of NUREG–1563 in Elicitations for the Yucca Mountain Site Characterization Program." Letter (December 31) to R.A. Milner, DOE, Office of Civilian Radioactive Waste Management. Washington, DC: NRC. Division of Waste Management. 1996.

[7]Brocoum, S.J. "Resolution of U.S. Nuclear Regulatory Commission Site Characterization Analysis Comment 3 and Other Comments Related to the Use of Expert Elicitations in the High-Level Waste Program." Letter (August 6) to M. Bell, NRC. Washington, DC: DOE. 1997.

[8]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Structural Deformation and Seismicity (October 11–12, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[9]Ibid.

[10]Ibid.

5.4.3.3 Groundwater-Specific Discharge

The transport time of radionuclides in the saturated zone is important to estimate potential repository performance. Uncertainty and variability of the groundwater flow system are accounted for in the DOE total system performance assessment through the probability distributions for three hydrologic input parameters: (i) groundwater-specific discharge, (ii) effective porosity, and (iii) horizontal anisotropy. In 1997, DOE conducted formal expert elicitations [hereafter referred to as the Saturated Zone Flow and Transport Expert Elicitation (CRWMS M&O, 1998a)] to better understand the state of knowledge and uncertainties regarding these key input parameters to any DOE total system performance assessment for Yucca Mountain. The panel of five experts addressed a variety of technical issues related to the saturated zone beneath Yucca Mountain and the region downgradient, including groundwater-specific discharge (flux). NRC and CNWRA staffs observed the expert elicitation workshops and reviewed the information developed through the documentation process (DOE, 1998) and found it sufficient to use in a potential license application for Yucca Mountain. Overall, DOE adequately justified the need for the elicitation and conducted the elicitation in accordance with the guidance set forth in NUREG–1563 (NRC, 1996). The broader technical details of saturated zone modeling are provided in Section 3.3.8 of this report.

In the Total System Performance Assessment–Site Recommendation, specific discharge in the site-scale saturated zone flow and transport model is represented using one of three discrete cases: (i) high, (ii) medium, or (iii) low. Only the medium-specific discharge is calculated directly in the three-dimensional saturated zone model. The value for the low-specific discharge case was one-tenth the value for the medium-specific discharge case, and the value for the high-specific discharge case was 10 times that of the medium case. To arrive at these values, four Saturated Zone Flow and Transport Expert Elicitation (CRWMS M&O, 1998a) panel members evaluated the uncertainty in hydraulic conductivity separately and subsequently propagated the results into a range of uncertainty for specific discharge.

For the Supplemental Science and Performance Analyses (Bechtel SAIC Company, LLC, 2001a) (a document DOE identified as also supporting the Yucca Mountain site recommendation), rather than relying on the original Saturated Zone Flow and Transport Expert Elicitation (CRWMS M&O, 1998a) estimates, DOE alternatively selected a factor of 3 above and below the medium-specific discharge case, such that specific discharge for the low-specific case is increased from one-tenth to one-third of the medium value, and is decreased for the high-specific case from 10 times to 3 times the medium value. Volume 1 (Part 2) of the Supplemental Science and Performance Analyses (Bechtel SAIC Company, LLC, 2001a) includes the results of an unquantified uncertainty analysis used to evaluate the treatment of uncertainty in the site-scale saturated zone flow and transport model. Uncertainty in the probability distribution for specific discharge was reevaluated because the previous range of values was based on the Saturated Zone Flow and Transport Expert Elicitation (CRWMS M&O, 1998a) available data and the literature. Specifically, the principal DOE investigators developing the saturated zone model concluded that the range for specific discharge used for the DOE Total System Performance Assessment–Site Recommendation is overly conservative. In particular, the investigators believe the maximum value of the parameter range is unreasonably large. Darcy's law states that the specific discharge is the product of hydraulic conductivity and hydraulic gradient, which are best characterized for Yucca Mountain in the vicinity of the C-Wells Complex. During the Saturated Zone Flow and Transport Expert Elicitation (CRWMS M&O, 1998a), it was recognized that, although the hydraulic gradient beneath Yucca Mountain is subject to uncertainty, its relative contribution to the uncertainty in specific discharge in the area of the C-Wells Complex is small. In general, the experts believed that the data from the multiple-hole pumping tests at the C-Wells Complex constituted the most reliable source for hydraulic conductivity estimates. In most cases, the experts provided a range of hydraulic conductivity values wider than that obtained from the C-Well Complex studies, reflecting uncertainty in the range of hydraulic conductivities that might characterize the units at other locations within the region. DOE also uses the same data from the multiple-hole pumping tests at the C-Wells Complex to decrease the range of values for specific discharge, arguing that the new reduced range better represents the data from the C-Wells Complex. Unlike the Saturated Zone Flow and Transport Expert Elicitation (CRWMS M&O, 1998a), DOE excludes the uncertainty in the hydraulic conductivity for locations not influenced by pumping tests at the C-Wells Complex. DOE also argued that the scale effects do not cause single-hole tests to underestimate the hydraulic conductivity of unfaulted regions as was previously thought. Therefore, it was concluded that the single-hole hydraulic conductivities reflect the true hydraulic conductivities of the hydrologic units in unfaulted areas and can be used to represent the hydraulic conductivities of the hydrogeologic units in numerical models, provided the effects of faults are accounted for in the same manner. DOE cites the recent work by Vesselinov, et al. (2001) at the Apache Leap test site as support for its reduced range of groundwater-specific discharge.

DOE is not required to strictly adhere to the recommendations of elicitations it sponsors. Where it departs from those recommendations, however, DOE should document any additional data, analyses, or other information, not considered by the expert panel, that factored into its departure decision. The Saturated Zone Flow and Transport Expert Elicitation (CRWMS M&O, 1998a) established the uncertainty range to include hydraulic conductivity uncertainty for locations not influenced by pumping tests at the C-Well Complex. No new data or analyses have been presented that would replace the technical basis for establishing the uncertainty range. The only new information cited [Section 12.3.1.4.1, Supplemental Science and Performance Analyses (Bechtel SAIC Company LLC, 2001a)] is a reference to an analysis by Vesselinov, et al. (2001) published in proceedings of a conference on fractured rock in Canada. It is not clear however, that the conclusions reached by Vesselinov, et al. (2001) have gained general acceptance within the broader technical/scientific community. It is also not clear that the conclusions, that reached for air-injection tests in a relatively small area at the Apache Leap site are applicable to groundwater pumping testing on a much larger scale at Yucca Mountain.

Air-permeability tests are used as an additional line of evidence, showing permeability can be enhanced near fault zones. The logic is then extended to argue that the cross-hole tests at the C-Wells Complex indicate higher permeability because faults are included in the relatively large scale of the aquifer tested. It is, therefore, reasoned that, because the DOE saturated zone flow model explicitly includes major faults, the permeability assigned to the hydrostratigraphic layer properties should reflect unfaulted (but still fractured) rock, which is reflected in the smaller-scale results of single-hole tests. Because the range in variability from the population of single-hole tests alone is less than the variability among both single- and cross-hole tests, DOE reasons that the range of uncertainty considered for total system performance assessment need only consider the range of permeability from the single-hole tests.

This logic may be sound as it applies to data uncertainty, however, it fails to consider and propagate model uncertainty into the DOE total system performance assessment.

To illustrate this point, it is helpful to look at the plot of permeability data shown in Figure 14 of the Calibration of the Site-Scale Saturated Zone Flow Model (CRWMS & MO, 2000a). Within any of the relatively permeable units, the range of permeability estimates from single-hole tests spans approximately one order of magnitude. This range can be considered data uncertainty, and a factor of three above or below the mean (as DOE proposes for the total system performance assessment uncertainty) adequately captures this data uncertainty. In several instances (i.e., for the Prow, Bullfrog, and Tram Tuffs), however, the final calibrated permeabilities for the saturated zone flow model are more than one order of magnitude outside the range of permeabilities measured in the single-hole tests. The difference between the calibrated permeability and the single-hole test permeability can be considered model uncertainty because the reason for the discrepancy is not clear. To account for the additional model uncertainty, a larger range of saturated-zone-specific discharges should be considered in the DOE total system performance assessment analyses. The factor of 10 above and below the calibrated model permeability that was previously used would account for the additional model uncertainty.

5.4.3.4 Sorption Coefficient Parameter Distributions

Sorption coefficient (k_d) parameter distributions are important to understand radionuclide transport phenomena in both the unsaturated and saturated zones (see Chapters 3.3.7 and 3.3.9 of this report). Although a significant amount of laboratory work and theoretical research concerning k_d values exists in the literature, there is little information on what the respective distributions may be for the various rock types present at Yucca Mountain. Despite the importance of k_d value, it is unlikely that any technically defensible distributions for sorption modeling, which are necessary to support the DOE total system performance assessment model abstractions of radionuclide transport, would be developed at the time of any potential license application submittal. Consequently, the staff view is that DOE was justified in its decision to estimate k_d parameter distributions for Yucca Mountain sorption modeling using the judgment of experts.

In determining k_d distributions, DOE relied on its own in-house experts (Los Alamos National Laboratory staff) which, although unusual, is permissible according to the guidance in NUREG–1563 (NRC, 1996), so long as any possible conflicts of interest are recognized and minimized to the extent practical to enhance credibility.[11] In this case, all three experts had an existing relationship with DOE and the Yucca Mountain program. After its completion, the results of the k_d distribution elicitation were initially documented in Barnard, et al. (1992). In reviewing this document, however, the technical basis for the expert-selected k_d distribution(s) is not clear because of inadequate documentation reflecting how the elicitation was conducted.

[11]Austin, J.H. "Documenting and Disclosing Potential Conflict of Interest in Expert Elicitations for the Geologic Repository Program." Letter (January 7) to R.A. Milner, DOE, Office of Civilian Radioactive Waste Management. Washington, DC: NRC, Division of Waste Management. 1997.

The analytical methods used to arrive at the k_d probability distribution functions are described in general terms in Barnard, et al. (1992), but the specific process for conducting the k_d elicitation procedure itself is not described. Specifically, there is no documentation that describes how the expect elicitation itself was conducted, as outlined in NUREG–1563 (NRC, 1996). In general, this information is needed to understand how the experts arrived at their conclusions (including what initial data were used that formed a basis for the elicitation) and, in particular, how the k_d probability distribution functions themselves were arrived at using this data. For example, Wilson, et al. (1994) noted that one of the experts believed that elemental lead (Pb) should be assigned a k_d of 0, but a consensus value of 0 to 500 [subsequently adjusted in DOE (1998) from 100 to 500] was adopted during the elicitation process. This information is necessary to verify the technical integrity of the elicitation process as well as the traceability of the assessment itself. Consequently, the absence of this documentation diminishes the acceptability and credibility of the elicitation results themselves because the process at present does not appear to be transparent and traceable. This is particularly important because, in subsequent reports [Wilson, et al. (1994); Triay, et al. (1997); and CRWMS M&O (2000b)], DOE continued to make modifications to the parameter distributions without explanation. To improve the transparency and traceability of DOE decisionmaking in this area, DOE agreed[12] to provide the requisite documentation supporting this elicitation, including documentation on differing opinions regarding how the k_d probability distributions were reconciled. Thus, this issue is considered closed-pending.

5.4.4 Summary

The staff continued to monitor DOE implementation of the guidance found in NUREG–1563 (NRC, 1996). Thus far, NRC observation of the DOE-sponsored elicitations revealed few, if any, significant deviations between DOE implementation and NRC guidance. Although some elicitations may have potential weaknesses, as previously discussed with DOE,[13,14,15,16] such weaknesses do not appear to fundamentally change the conclusion or outcome of total system performance assessment presented by DOE to date. Because there are weaknesses in the

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[12]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7, 2000)." Letter (December 12) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[13]Austin, J.H. "Implementation of NUREG–1563 in Expert Elicitations for the Yucca Mountain Site Characterization Program." Letter (December 31) to R.A. Milner, DOE, Office Civilian Radioactive Waste Management. Washington, DC: NRC, Division of Waste Management. 1996.

[14]Austin, J.H. "Documenting and Disclosing Potential Conflict of Interest in Expert Elicitations for the Geologic Repository Program." Letter (January 7) to R.A. Milner, DOE, Office of Civilian Radioactive Waste Management. Washington, DC: NRC, Division of Waste Management. 1997.

[15]Bell, M.J. "Summary of U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange on the Total-System Performance Assessment (July 21–22, 1997)." Letter (December 17) to R.A. Milner, DOE. Washington, DC: NRC. 1997.

[16]Bell, M.J. "Summary of U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange on the Total-System Performance Assessment (November 5–6, 1997)." Letter (June 24) to S. Rousso, DOE. Washington, DC: NRC. 1998.

respective elicitations, the staff obtained detailed agreements from DOE to provide new information that can resolve the specific NRC concerns, as noted in Section 5.4.5 of this Integrated Resolution Status Report.

Lastly, NRC regulation requires any potential license application be as complete as possible at the time of docketing. For potential updating of elicitation results, the staff will continue to monitor DOE decisionmaking as it relates to the reexamination of elicitation results and the potential need for updating when new site characterization, design, and performance assessment information become available. In this regard, DOE had an agreement to provide the staff with its administrative procedure describing when and how new data would be treated after completion of an elicitation.[17] Staff are currently reviewing Section 5.14, Reassessment of the procedure in question (DOE, 1999) to determine how it comports with NRC guidance found in NUREG–1563 (NRC, 1996).

5.4.5 Status and Path Forward

5.4.5.1 Probabilistic Volcanic Hazards Analysis

DOE conducted the preliminary analysis of the aeromagnetic anomalies. As shown in Bechtel SAIC Company, LLC (2001b, Table 1), the new aeromagnetic data show 20 anomalies that can be interpreted as buried basalt. This increase in potential buried basalt bodies is well outside the average hidden event factor used in the 1996 Probabilistic Volcanic Hazards Analysis (Geomatrix Consultants, 1996). As stated in Bechtel SAIC Company, LLC (2001b), a determination of the Plio-Pleistocene volcanic inventory was one of the interpretations made by the volcanism experts. The increase in anomalies from 7 to 20 and the increase in quality of data to be evaluated by the experts strongly suggest that the probability of the volcanic event needs to be reviewed and updated. At present, DOE is scheduled to furnish staff with evaluation results of the aeromagnetic maps as part of a U.S. Geological Survey Open-File Report scheduled for publication in January 2002 and meet with NRC in March 2002 to discuss how the information on the anomalies will be factored into the probability estimates.

5.4.5.2 Probabilistic Seismic Hazards Analysis

5.4.5.2.1 Seismic Source and Fault Displacement Characterization

No further action in this area is required at this time.

5.4.5.2.2 Ground-Motion Attenuation

To close this issue at the staff level, DOE needs to provide the documentation originally requested by NRC during the October 2000 Structural Deformation and Seismicity Technical

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[17]Bell, M.J. "Summary of U.S. Department of Energy/U.S. Nuclear Regulatory Commission Technical Exchange on the Total-System Performance Assessment (July 21–22, 1997)." Letter (December 17) to R.A. Milner, DOE. Washington, DC: NRC. 1998.

Exchange.[18] The staff seek DOE documentation of the extent to which each of the seven ground-motion experts understood the probabilistic modeling concepts associated with the respective inputs to the attenuation models as well as the subsequent implementation of the model in the broader Probabilistic Seismic Hazard Assessment.

5.4.5.3 Groundwater-Specific Discharge

Results of the unquantified uncertainty analysis were documented in the Supplemental Science and Performance Analyses (Bechtel SAIC Company, LLC 2001a) for the first time. Consequently, the NRC staff will wait for DOE to choose which of the two alternative methods is to be applied in the DOE total system performance assessment. If DOE decides to depart from the original Saturated Zone Expert Elicitation (CRWMS M&O, 1998a) panel recommendations, the NRC staff will then review the documentation to determine if a new technical basis needs to be provided to support a new range of uncertainty values for specific discharge.

5.4.5.4 Sorption Coefficient Parameter Distributions

The informality of this elicitation could jeopardize acceptability of the DOE k_d probability distribution functions. To close this issue at the staff level, DOE needs to provide the documentation originally requested by NRC during the December 5–7, 2000, Technical Exchange on the Radionuclide Transport Key Technical Issue.[19]

5.4.6 References

Barnard, R.W., M.L. Wilson, H.A. Dockery, J.H. Gauthier, P.G. Kaplan, R.R. Eaton, F.W. Bingham, and T.H. Robey. "TSPA 1991: An Initial Total-System Performance Assessment for Yucca Mountain." SAND91–2795. Albuquerque, New Mexico: Sandia National Laboratories. 1992.

Bechtel SAIC Company, LLC. "FY01 Supplemental Science and Performance Analyses." Vol. 2: Performance Analyses. TDR–MGR–PA–000001. Revision 00. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2001a.

Bechtel SAIC Company, LLC. "Technical Update Impact Letter Report." Appendix I: Summary of Recent Information Relevant to Disruptive Events—Volcanism and Seismicity. MIS–MGR–RL–000001. Revision 00. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2001b.

[18]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Structural Deformation and Seismicity (October 11–12, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[19]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Radionuclide Transport (December 5–7, 2000)." Letter (December 12) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

Budnitz, R.J., G. Apostolakis, D.M. Boore, L.S. Cluff, K.J. Coppersmith, C.A. Cornell, and P.A. Morris. NUREG/CR–6372, “Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts—Main Report.” Vol. 1. Washington, DC: NRC. 1997.

Cornell, C.A. “Engineering Seismic Risk Analysis.” *Bulletin of the Seismological Society of America*. Vol. 58. pp. 1,583–1,606. 1986.

CRWMS M&O. “Saturated Zone Flow and Transport Expert Elicitation Project—Final Report.” North Las Vegas, Nevada: Geomatrix Consultants and TRW Environmental Safety Systems, Inc. 1998a.

———. “Probabilistic Seismic Hazard Analyses for Fault Displacement and Vibratory Ground Motion at Yucca Mountain, Nevada: Final Report.” Wong and Stepp, coords. Report SP32IM3. 3 Vols. Oakland, California: CRWMS M&O. 1998b.

———. “Calibration of the Site-Scale Saturated Zone Flow Model.” MDL–HS–000011. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000a.

———. “Unsaturated Zone and Saturated Zone Transport Properties.” ANL–NBS–HS–000019. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000b.

DOE. “Viability Assessment of a Repository at Yucca Mountain.” DOE/RW–0508. Overview and all 5 Vols. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 1998.

———. “Expert Elicitation.” OCRWM Procedure AP-AC.1Q. Revision 00. Las Vegas, Nevada: DOE, Office of Civilian Radioactive Waste Management. 1999.

Geomatrix Consultants. “Probabilistic Volcanic Hazards Analysis for Yucca Mountain, Nevada.” BA0000000–1717–2200–00082. Revision 0. San Francisco, California: Geomatrix Consultants. 1996.

McGuire, R.K. “FORTRAN Computer Program for Seismic Risk Analysis.” U.S. Geological Survey Open-File Report 76-67. 1976.

Meyer, M.A. and J.M. Booker. NUREG/CR–5424, “Eliciting and Analyzing Expert Judgment: A Practical Guide.” Washington, DC: NRC. January 1990.

NRC. NUREG–1563, “Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program.” Washington, DC: NRC. November 1996.

Risk Engineering, Inc. “FRISK88 User’s Manual (Version 2.0).” Boulder, Colorado: Risk Engineering, Inc. 1998.

Administrative and Programmatic Requirements

Triay, I.R., A. Meijer, J.L. Conca, K.S. Kung, R.S. Rundberg, and E.A. Strietelmeier. “Summary and Synthesis Report on Radionuclide Retardation for the Yucca Mountain Site Characterization Project, Milestone 3784M.” LA–13262–MS. Los Alamos, New Mexico: Los Alamos National Laboratory, Chemical Science and Technology Division. 1997.

TRW Environmental Safety Systems, Inc. “Total System Performance Assessment–Viability Assessment (TSPA-VA) Plan.” B00000000–01717–2200–00179. Las Vegas, Nevada: TRW Environmental Safety Systems, Inc. 1996.

Vesselinov, V.V., W. Illman, Y. Hyun, S.P. Neuman, V. Di Federico, and D.M. Taratakovsky. “2001, Observation and Analysis of a Pronounced Permeability and Porosity Scale-Effect in Unsaturated Fractured Tuff.” *Fractured Rock*. Toronto, Canada: Queens University. 2001.

Wilson, M.L., J.G. Gauthier, R.W. Barnard, G.E. Barr, H.A. Dockery, E. Dunn, R.R. Eaton, D.C. Guerin, N. Lu, M.J. Martinez, R. Nilson, C.A. Rautman, T.H. Robey, B. Ross, E.E. Ryder, A.R. Schenker, S.A. Shannon, L.H. Skinner, W.G. Halsey, J.D. Gansemer, L.C. Lewis, A.D. Lamont, I.R. Triay, A. Meijer, and D.E. Morris. “Total-System Performance Assessment for Yucca Mountain—SNL Second Iteration (TSPA–1993).” SAND93–2675. Albuquerque, New Mexico: Sandia National Laboratories. 1994.

5.5 Status and Path Forward

Text in this section will be provided at a later date.

6 SUMMARY AND CONCLUSIONS

This report provides the status of resolution of technical issues at the staff level to all parties that may have an interest in the proposed geologic repository at Yucca Mountain. Prelicensing consultations between DOE and NRC are called for in the Nuclear Waste Policy Act of 1982 (1982). DOE and NRC use these consultations, including document reviews and technical exchanges, to resolve technical issues. Resolution of technical issues before DOE submits any license application increases the likelihood that the license application will contain the information required for an efficient and effective regulatory review. Technical issues are considered resolved at the staff level when the NRC staff considers the information gathered by DOE sufficient for the staff to conduct their review. Resolution, however, does not imply any conclusions regarding the end result of such a review. Moreover, any issue can be reopened if new information becomes available.

Starting in August 2000, various technical exchanges were conducted between the DOE and NRC staffs with the specific objective of issue resolution. These technical exchanges were held as open public meetings. Available information was evaluated for its sufficiency for inclusion in any license application. Where such information was judged to be insufficient, NRC reached agreements with DOE, which specify the additional information DOE will collect, a schedule for obtaining such information, and a mechanism for providing the information to the NRC staff. This report incorporates the results of the technical exchanges completed before October 31, 2001. This version also includes regulatory information, such as 10 CFR Part 63, 10 CFR Part 963, and the Yucca Mountain Review Plan (NRC, 2002), through March 2002. Technical exchanges on all the preclosure topics have not been completed to date. Therefore, some of the sections in Chapter 2 are not complete. Additional information on preclosure, as well as other key areas within NRC (2002) will be included in the next update of this report.

Overall, there are 9 postclosure key technical issues partitioned into 37 subissues. As indicated in Table 1.1-3, 5 of these subissues are classified as closed and 32 as closed-pending. The majority of the subissues are classified as closed-pending. Two hundred and ninety-three agreements were reached with DOE for these subissues to gain the closed-pending classification. The full text of these agreements is provided in Appendix A.

As a part of its risk-informed approach, NRC regulatory reviews will focus on technical items significant to repository performance preclosure and postclosure safety. Chapter 3 of this report is structured according to these integrated subissues.

NRC staff will review the information received from DOE in response to DOE and NRC agreements to determine if the information is sufficient and, if not, what additional information is needed. These reviews will be provided to DOE and other interested parties via formal letters and will be documented, as appropriate, in the next revision of this report.

6.1 References

NRC. NUREG–1804, "Yucca Mountain Review Plan—Draft Report for Comment." Revision 2. Washington, DC: NRC. March 2002.

Nuclear Waste Policy Act of 1982. Pub. L. 97–425. 96 Stat. 2201 (1982).

KTI Agreement - ISI Crosswalk

Agreement	Related ISIs or	NRC/DOE Agreement	Status
CLST.1.01	ENG1	Provide the documentation for Alloy 22 and titanium for the path forward items listed on slide 8. [establish credible range of brine water chemistry; evaluate effect of introduced materials on water chemistry; determine likely concentrations and chemical form of minor constituents in YM waters; characterize YM waters with respect to the parameters which define the type of brine which would evolve; evaluate periodic water drip evaporation] DOE will provide the documentation in a revision to AMR “Environment on the Surfaces of the Drip Shield and Waste Package Outer Barrier” by LA.	Not Received
CLST.1.02	ENG1	Provide the documentation for the path forward items listed on slide 12. [surface elemental analysis of alloy test specimens is necessary for determination of selective dissolution; surface analysis of welded specimens for evidence of dealloying; continue testing including simulated saturated repository environment to confirm enhancement factor] DOE will provide the documentation in a revision to AMR “General and Localized Corrosion of Waste Package Outer Barrier” by LA.	Not Received
CLST.1.03	ENG1	Provide documentation that confirms the linear polarization resistance measurements with corrosion rate measurements using other techniques. DOE will provide the documentation in a revision to AMR “General and Localized Corrosion of Waste Package Outer Barrier” by LA.	Not Received
CLST.1.04	ENG1	Provide the documentation for Alloy 22 and titanium for the path forward items listed on slide 14. [continue testing in the LTCTF; add new bounding water test environments to LTCTF (SSW & BSW); install thinner coupons in LTCTF with larger surface area/volume rations; install high sensitivity probes of Alloy 22 in some of the LTCTF vessels; materials testing continues during performance confirmation] DOE will provide the documentation in a revision to AMR “ANL-EBS-MD-000003 and ANL-EBS-MD-000004" by LA.	Not Received
CLST.1.05	ENG1	Provide additional details on sensitivities, resolution of measurements, limitations, and deposition of silica for the high sensitivity probes. DOE will document the results of the sensitivity probes including limitation and resolution of measurements as affected by silica deposition in the Alloy 22 AMR and Ti Corrosion AMR (ANL-EBS-MD-000003 and ANL-EBS-MD-000004) prior to LA.	Not Received
CLST.1.06	ENG1	Provide the documentation on testing showing corrosion rates in the absence of silica deposition. DOE will document the results of testing in the absence of silica deposits in the revision of Alloy 22 AMR (ANL-EBS-MD-000003) prior to LA.	Not Received
CLST.1.07	ENG1	Provide the documentation for the alternative methods to measure the corrosion rate of the waste package material (e.g., ASTM G-102 testing) or provide justification for the current approach. DOE will document the alternative methods of corrosion measurement in the revision of Alloy 22 AMR (ANL-EBS-MD-000003), prior to LA.	Not Received
CLST.1.08	ENG1	Provide the documentation for Alloy 22 and titanium for the path forward items listed on slide 16 and 17. [calculate potential-pH diagrams for multi-component Alloy 22; grow oxide films at higher temperatures in autoclaves, in air and/or electrochemically to accelerate film growth for compositional and structural studies below; resolve kinetics of film growth: parabolic or higher order, whether film growth becomes linear, and if, as film grows it becomes mechanically brittle and spalls off; determine chemical, structural, and mechanical properties of films, including thicken films; correlate changes in Ecorr measured in LTCTF with compositional changes in passive film over time; perform analyses on cold-worked materials to determine changes in film structural properties; perform examination of films formed on naturally occurring Josephinite; compare films formed on Alloy 22 with other similar passive film Alloys with longer industrial experience] DOE will provide the documentation in the revision to AMRs (ANL-EBS-MD-000003 and ANL-EBS-MD-000004) prior to LA.	Not Received
CLST.1.09	ENG1	Provide the data that characterizes the passive film stability, including the welded and thermally aged specimens. DOE will provide the documentation in a revision to AMRs (ANL-EBS-MD-000003 and ANL-EBS-MD-000004) prior to LA.	Not Received

Agreement	Related ISIs or	NRC/DOE Agreement	Status
CLST.1.10	ENG1	Provide the documentation for Alloy 22 and titanium for the path forward items listed on slide 21 and 22. [measure corrosion potentials in the LTCTF to determine any shift of potential with time toward the critical potentials for LC; determine critical potentials on welded and welded and aged coupons of Alloy 22 vs those for base metal - particularly important if precipitation or severe segregation of alloying elements occurs in the welds; separate effects of ionic mix of specimens in YM waters on critical potentials - damaging species from potentially beneficial species; determine critical potentials in environments containing heavy metal concentrations] DOE will provide the documentation in a revision to AMRs (ANL-EBS-MD-000003 and ANL-EBS-MD-000004) prior to LA.	Not Received
CLST.1.11	ENG1	Provide the technical basis for the selection of the critical potentials as bounding parameters for localized corrosion, taking into account MIC. DOE will provide the documentation in a revision to AMRs (ANL-EBS-MD-000003 and ANL-EBS-MD-000004) prior to LA.	Not Received
CLST.1.12	ENG1	Provide the documentation for Alloy 22 and titanium for the path forward items listed on slides 34 and 35. [qualify and optimize mitigation processes; generate SCC data for mitigated material over full range of metallurgical conditions; new vessels for LTCTF will house many of the SCC specimens; continue SSRT in same types of environments as above, specimens in the same range of metallurgical conditions; determine repassivation constants needed for film rupture SCC model to obtain value for the model parameter 'n'; continue reversing direct current potential drop crack propagation rate determinations in same types of environments and same metallurgical conditions as for SSRT and LTCTF tests; evaluate SCC resistance of welded and laser peened material vs non-welded unpeened material; evaluate SCC resistance in induction annealed material; evaluate SCC resistance of full thickness material obtained from the demonstration prototype cylinder of Alloy 22] DOE will provide the documentation in a revision to AMRs (ANL-EBS-MD-000005 and ANL-EBS-MD-000006) prior to LA.	Not Received
CLST.1.13	ENG1 ENG2 PRE	Provide the data that characterizes the distribution of stresses due to laser peening and induction annealing of Alloy 22. DOE will provide the documentation in a revision to AMR (ANL-EBS-MD-000005) prior to LA.	Not Received
CLST.1.14	ENG2 PRE	Provide the justification for not including the rockfall effect and deadload from drift collapse on SCC of the waste package and drip shield. DOE will provide the documentation for the rockfall and dead-weight effects in the next revision of the SCC AMR (ANL-EBS-MD-000005) prior to LA.	Not Received
CLST.1.15	ENG1	Provide the documentation for Alloy 22 and titanium for the path forward items listed on slide 39. [install specimens cut from welds of SR design mock-up in LTCTF and in other SCC test environments - determine which specimen geometry is most feasible to complement SCC evaluation; evaluate scaling and weld process factors between thin coupons and dimensions in actual welded waste package containers - including thermal/metallurgical structural effects of multi-pass weld processes; provide representative weld test specimens for MIC work, thermal aging and localized corrosion evaluations] DOE will provide documentation for Alloy 22 and Ti path forward items on slide 39 in a revision to the SCC and general and localized corrosion AMRs (ANL-EBS-MD-000003, ANL-EBS-MD-000004, ANL-EBS-MD-000005) by LA.	Not Received
CLST.1.16	ENG1 ENG2 PRE	Provide the documentation on the measured thermal profile of the waste package material due to induction annealing. DOE stated that the thermal profiles will be measured during induction annealing, and the results will be reported in the next SCC AMR (ANL-EBS-MD-000005) prior to LA.	Not Received
CLST.1.17	ENG1 ENG2 PRE	Provide additional detail on quality assurance acceptance testing. DOE stated that it would provide guidance and criteria in the next revision of the Technical Guidance Document (TGD) for LA. The development of the LA sections and associated programs and process controls for the procurement and fabrication of waste package materials and components will be included. This will include consideration of the controls for compositional variations in Alloy 22. The TGD revision will be issued by June 2001, contingent upon NRC publication of the final 10 CFR 63 and the Yucca Mountain Review Plan.	Not Received

Agreement	Related ISIs or	NRC/DOE Agreement	Status
CLST.2.01	ENG2 PRE	Either provide documentation using solid element formulation, or provide justification for not using it, for the drip shield - rockfall analysis. DOE stated that shell elements include normal stresses and transverse stresses in the calculations and provide more accurate results for thin plates and use far fewer elements. Therefore, shell elements will be used instead of solid elements. This justification will be documented in the next revision of AMR ANL-XCS-ME-000001, Design Analysis for the Ex-Container Components, prior to LA.	Not Received
CLST.2.02	ENG2 PRE	Provide the documentation for the point loading rockfall analysis. DOE stated that point loading rock fall calculations will be documented in the next revisions of AMRs ANL-XCS-ME-000001, Design Analysis for the Ex-Container Components, and ANL-UDC-MD-000001, Design Analysis for UCF Waste Packages, both to be completed prior to LA.	Not Received
CLST.2.03	ENG2 PRE	Demonstrate how the Tresca failure criterion bounds a fracture mechanics approach to calculating the mechanical failure of the drip shield. DOE stated that it believes its current approach of using ASME Code is appropriate for this application. Additional justification for this conclusion will be included in the next revision of AMR ANL-XCS-ME-000001, Design Analysis for the Ex-Container Components, to be completed prior to LA.	Not Received
CLST.2.04	ENG1 ENG2 PRE	Provide information on the effect of the entire fabrication sequence on phase instability of Alloy 22, including the effect of welding thick sections using multiple weld passes and the proposed induction annealing process. DOE stated that the aging studies will be expanded to include solution annealed and induction annealed Alloy 22 weld and base metal samples from the mock-ups as well as laser peened thick, multi-pass welds. This information will be included in revisions of the AMR "Aging and Phase Stability of the Waste Package Outer Barrier," ANL-EBS-MD-000002, before LA.	Not Received
CLST.2.05	ENG1 ENG2 PRE	Provide the "Aging and Phase Stability of Waste Package Outer Barrier," AMR, including the documentation of the path forward items listed in the "Subissue 2: Effects of Phase Instability of Materials and Initial Defects on the Mechanical Failure and Lifetime of the Containers" presentation, slides 5 & 6. [data input to current models is being further evaluated and quantified to reduce uncertainty; aging of Alloy 22 samples for microstructural characterization, tensile property test, and Charpy impact test is ongoing; theoretical modeling will be employed to enhance confidence in extrapolating aging kinetic data to repository thermal conditions and time scale - modeling will utilize thermodynamic principles of the processes; Alloy 22 samples for SCC compact tension test are being added to aging studies; test program will be expanded to include welded and cold worked materials; effects of stress mitigation techniques such as laser peening and induction annealing on phase instability will be investigated; aging test facility will be expanded to include aging at lower temperatures] DOE stated that the "Aging and Phase Stability of the Waste Package Outer Barrier" AMR, ANL-EBS-MD-000002, Rev. 00 was issued 3/20/00. This AMR will be revised to include the results of the path forward items before LA.	Not Received
CLST.2.06	ENG1 ENG2 PRE	Provide the technical basis for the mechanical integrity of the inner overpack closure weld. DOE will provide the documentation in AMR, ANL-UDC-MD-000001, Rev. 00, Design Analysis for UFC Waste Packages in the next revision, prior to LA.	Not Received
CLST.2.07	ENG1 ENG2 PRE	Provide documentation for the fabrication process, controls, and implementation of the phases which affect the TSPA model assumptions for the waste package (e.g., filler metal, composition range). DOE stated that updates of the documentation on the fabrication processes and controls (TDR-EBS-ND-000003, Waste Package Operations Fabrication Process Report and TDP-EBS-ND-000005, Waste Package Operations FY-00 Closure Weld Technical Guidelines Document) will be available to the NRC in January 2001.	Received

Agreement	Related ISIs or	NRC/DOE Agreement	Status
CLST.2.08	ENG1 ENG2 PRE	Provide documentation of the path forward items in the "Subissue 2: Effects of Phase Instability of Materials and Initial Defects on the Mechanical Failure and Lifetime of the Containers" presentation, slide 16. [future rockfall evaluations will address (1) effects of potential embrittlement of WP closure material after stress annealing due to aging, (2) effects of drip shield wall thinning due to corrosion; (3) effects of hydrogen embrittlement on titanium drip shield; and (4) effects of multiple rock blocks falling on WP and drip shield; future seismic evaluations will address the effects of static loads from fallen rock on drip shield during seismic events] DOE stated that the rockfall calculations addressing potential embrittlement of the waste package closure weld and rock falls of multiple rock blocks will be included in the next revision of the AMR ANL-UDC-MD-000001, Design Analysis for UCF Waste Packages, to be completed prior to LA. Rock fall calculations addressing drip shield wall thinning due to corrosion, hydrogen embrittlement of titanium, and rock falls of multiple rock blocks will be included in the next revision of the AMR ANL-XCS-ME-000001, Design Analysis for the Ex-Container Components, to be completed prior to LA. Seismic calculations addressing the load of fallen rock on the drip shield will be included in the next revision of the AMR ANL-XCS-ME-000001, Design Analysis for the Ex-Container Components, to be completed prior to LA.	Not Received
CLST.2.09	ENG2 PRE	Demonstrate the drip shield and waste package mechanical analysis addressing seismic excitation is consistent with the design basis earthquake covered in the SDS KTI. DOE stated that the same seismic evaluations of waste packages and drip shield (revision of AMRs ANL-UDC-MD-000001 and ANL-XCS-ME-000001) will support both the SDS KTI and the CLST KTI, therefore consistency is ensured. These revisions will be completed prior to LA.	Not Received
CLST.3.01	ENG4 TSPAI	The agreement addresses CLST Subissues 3 & 4. In the revision to the "Summary of In-Package Chemistry for Waste Forms," AMR, the NRC needs to know whether and how initial failures are included in the in-package chemistry modeling, taking into account the multiple barrier analysis. DOE stated that the Summary of In-Package Chemistry for Waste Forms ANL-EBS-MD-000050 deals with time since waste package breach, instead of time of waste package failures. The model is appropriate for the current implementation in the TSPA scenarios because breaches do not occur until after aqueous films may be sustained. Multiple barrier analyses are discussed in the TSPAI IRSR, and therefore will be discussed in the TSPA KTI Technical Exchange.	Received
CLST.3.02	ENG3 ENG4	The agreement addresses CLST Subissues 3 & 4. In the revision to the "Summary of In-Package Chemistry for Waste Forms," AMR, address specific NRC questions regarding radiolysis, incoming water, localized corrosion, corrosion products, transient effects, and a sensitivity study on differing dissolution rates of components. DOE stated that these specific questions are currently being addressed in the revision of the Summary of In-Package Chemistry for Waste Forms AMR, ANL-EBS-MD-000050 and related AMRs and calculations. To be available in January 2001.	Received
CLST.3.03	ENG4	The agreement addresses CLST Subissues 3 & 4. Provide a more detailed calculation on the in-package chemistry effects of radiolysis. DOE stated that the calculations recently performed as discussed at the 9/12/00 Technical Exchange and preceeding teleconferences are being documented. These calculations will be referenced and justified in the revision of the Summary of In-Package Chemistry for Waste Forms AMR, ANL-EBS-MD-000050 and will be available in January 2001.	Received
CLST.3.04	ENG3 ENG4 TSPAI	The agreement addresses CLST Subissues 3 & 4. Need consistency between abstractions for incoming water and sensitivity studies conducted for in-package calculations, in particular, taking into account the interaction of engineered materials on the chemistry of water used for input to in-package abstractions. DOE stated that the revision of the Summary of In-Package Chemistry for Waste Forms AMR, ANL-EBS-MD-000050 will discuss the applicability of abstractions for incoming water, taking into account the revised Environment on the Surfaces of the Drip Shield and Waste Package Outer Barrier AMR. The revision will be available in January 2001.	Received
CLST.3.05	ENG4	The agreement addresses CLST Subissues 3 & 4. Provide the plan for experiments demonstrating in-package chemistry, and take into account subsequent NRC comments, if any. DOE stated that the current planning provides for the analysis of additional in-package chemistry model support. This analysis will determine which parts of the model are amenable to additional support by testing, and which parts are more amenable to sensitivity analysis, or use of analogues. Based on these results, longer range testing will be considered. If testing is determined to be appropriate, test plans will be written in FY01 and made available to the NRC.	Not Received

Agreement	Related ISIs or	NRC/DOE Agreement	Status
CLST.3.06	ENG4	The agreement addresses CLST Subissues 3 & 4. Provide additional technical basis for the failure rate and how the rate is affected by localized corrosion. DOE stated that the technical basis for local corrosion conditions will be added to by additional discussion of local chemistry in the Summary of In-package Chemistry for Waste Forms revision ANL-EBS-MD-000050 which will be available in January 2001. Current Clad Degradation Summary Abstraction AMR Section 6.3, ANL-WIS-MD-000007 and Clad Degradation - Local Corrosion of Zirconium and its Alloys Under Repository Conditions AMR, ANL-EBS-MD-000012 contain the overall technical basis.	Received
CLST.3.07	ENG4	The agreement addresses CLST Subissues 3 & 4. Provide data to address chloride induced localized corrosion and SCC under the environment predicted by in-package chemistry modeling. DOE stated that the technical basis for the models used for localized corrosion and SCC will be expanded in future revisions of the Clad Degradation Summary Abstraction AMR, ANL-WIS-MD-000007, available by LA.	Not Received
CLST.3.08	ENG4	The agreement addresses CLST Subissues 3 & 4. Provide the documentation on the distribution for cladding temperature and stress used for hydride embrittlement. DOE stated that the stresses are documented in the Initial Cladding Conditions AMR, ANL-EBS-MD-000048. CAL-UDC-ME-000001 contains the waste package internal temperatures. Waste package surface temperatures were provided within the TSPA model (ANL-EBS-HS-000003, Rev 00, ICN 01 and ANL-EBS-MD-000049). The updated versions of these documents will be available in January 2001.	Received
CLST.3.09	ENG4	The agreement addresses CLST Subissues 3 & 4. Provide a technical basis for critical stress that is relevant for the environment in which external SCC takes place. DOE stated that critical stress from SCC experiments under more aggressive conditions will be cited in the Revision of the Cladding Degradation Summary Abstraction AMR, ANL-WIS-MD-000007, which will be available in January 2001.	Received
CLST.3.10	ENG4	The agreement addresses CLST Subissues 3 & 4. Provide analysis of the rockfall and vibratory loading effects on the mechanical failure of cladding, as appropriate. DOE stated that the vibratory effects are documented in Sanders et. al. 1992 SAND90-2406, A Method For Determining The Spent-Fuel Contribution To Transport Cask Containment Requirements. This will be discussed in the SDS KTI meeting. The analysis of the rockfall effects on the mechanical failure of cladding will be addressed if the agreed to updated rockfall analysis in Subissue #2, Item 8 and Subissue #1, Item 14 demonstrate that the rock will penetrate the drip shield and damage the waste package.	Partly Received
CLST.4.01	ENG4 TSPAI	The agreement addresses CLST Subissues 3 & 4. In the revision to the “Summary of In-Package Chemistry for Waste Forms,” AMR, the NRC needs to know whether and how initial failures are included in the in-package chemistry modeling, taking into account the multiple barrier analysis. DOE stated that the Summary of In-Package Chemistry for Waste Forms ANL-EBS-MD-000050 deals with time since waste package breach, instead of time of waste package failures. The model is appropriate for the current implementation in the TSPA scenarios because breaches do not occur until after aqueous films may be sustained. Multiple barrier analyses are discussed in the TSPAI IRSR, and therefore will be discussed in the TSPA KTI Technical Exchange.	Received
CLST.4.02	ENG3 ENG4	The agreement addresses CLST Subissues 3 & 4. In the revision to the “Summary of In-Package Chemistry for Waste Forms,” AMR, address specific NRC questions regarding radiolysis, incoming water, localized corrosion, corrosion products, transient effects, and a sensitivity study on differing dissolution rates of components. DOE stated that these specific questions are currently being addressed in the revision of the Summary of In-Package Chemistry for Waste Forms AMR, ANL-EBS-MD-000050 and related AMRs and calculations. To be available in January 2001.	Received
CLST.4.03	ENG4	The agreement addresses CLST Subissues 3 & 4. Provide a more detailed calculation on the in-package chemistry effects of radiolysis. DOE stated that the calculations recently performed as discussed at the 9/12/00 Technical Exchange and preceeding teleconferences are being documented. These calculations will be referenced and justified in the revision of the Summary of In-Package Chemistry for Waste Forms AMR, ANL-EBS-MD-000050 and will be available in January 2001.	Received
CLST.4.04	ENG3 ENG4 TSPAI	The agreement addresses CLST Subissues 3 & 4. Need consistency between abstractions for incoming water and sensitivity studies conducted for in-package calculations, in particular, taking into account the interaction of engineered materials on the chemistry of water used for input to in-package abstractions. DOE stated that the revision of the Summary of In-Package Chemistry for Waste Forms AMR, ANL-EBS-MD-000050 will discuss the applicability of abstractions for incoming water, taking into account the revised Environment on the Surfaces of the Drip Shield and Waste Package Outer Barrier AMR. The revision will be available in January 2001.	Received

Agreement	Related ISIs or	NRC/DOE Agreement	Status
CLST.4.05	ENG4	The agreement addresses CLST Subissues 3 & 4. Provide the plan for experiments demonstrating in-package chemistry, and take into account subsequent NRC comments, if any. DOE stated that the current planning provides for the analysis of additional in-package chemistry model support. This analysis will determine which parts of the model are amenable to additional support by testing, and which parts are more amenable to sensitivity analysis, or use of analogues. Based on these results, longer range testing will be considered. If testing is determined to be appropriate, test plans will be written in FY01 and made available to the NRC.	Not Received
CLST.4.06	ENG4	The agreement addresses CLST Subissues 3 & 4. Provide additional technical basis for the failure rate and how the rate is affected by localized corrosion. DOE stated that the technical basis for local corrosion conditions will be added to by additional discussion of local chemistry in the Summary of In-package Chemistry for Waste Forms revision ANL-EBS-MD-000050 which will be available in January 2001. Current Clad Degradation Summary Abstraction AMR Section 6.3, ANL-WIS-MD-000007 and Clad Degradation - Local Corrosion of Zirconium and its Alloys Under Repository Conditions AMR, ANL-EBS-MD-000012 contain the overall technical basis.	Received
CLST.4.07	ENG4	The agreement addresses CLST Subissues 3 & 4. Provide data to address chloride induced localized corrosion and SCC under the environment predicted by in-package chemistry modeling. DOE stated that the technical basis for the models used for localized corrosion and SCC will be expanded in future revisions of the Clad Degradation Summary Abstraction AMR, ANL-WIS-MD-000007, available by LA.	Not Received
CLST.4.08	ENG4	The agreement addresses CLST Subissues 3 & 4. Provide the documentation on the distribution for cladding temperature and stress used for hydride embrittlement. DOE stated that the stresses are documented in the Initial Cladding Conditions AMR, ANL-EBS-MD-000048. CAL-UDC-ME-000001 contains the waste package internal temperatures. Waste package surface temperatures were provided within the TSPA model (ANL-EBS-HS-000003, Rev 00, ICN 01 and ANL-EBS-MD-000049). The updated versions of these documents will be available in January 2001.	Received
CLST.4.09	ENG4	The agreement addresses CLST Subissues 3 & 4. Provide a technical basis for critical stress that is relevant for the environment in which external SCC takes place. DOE stated that critical stress from SCC experiments under more aggressive conditions will be cited in the Revision of the Cladding Degradation Summary Abstraction AMR, ANL-WIS-MD-000007, which will be available in January 2001.	Received
CLST.4.10	ENG4	The agreement addresses CLST Subissues 3 & 4. Provide analysis of the rockfall and vibratory loading effects on the mechanical failure of cladding, as appropriate. DOE stated that the vibratory effects are documented in Sanders et. al. 1992 SAND90-2406, A Method For Determining The Spent-Fuel Contribution To Transport Cask Containment Requirements. This will be discussed in the SDS KTI meeting. The analysis of the rockfall effects on the mechanical failure of cladding will be addressed if the agreed to updated rockfall analysis in Subissue #2, Item 8 and Subissue #1, Item 14 demonstrate that the rock will penetrate the drip shield and damage the waste package.	Partly Received
CLST.4.11	ENG4	See also CLST Subissue 3 agreements. In addition, in the revision to the "Defense High Level Waste Glass Degradation," AMR, address specific NRC questions regarding (a) the inconsistency of the rates in acid leg for glasses, (b) the technical basis for use of boron versus silica in the radionuclide release from glass, and (c) clarification of the definition of long term rates of glass dissolution. DOE stated that these questions will be addressed in the Defense High Level Waste AMR revision and will be available in January 2001.	Received
CLST.5.01	ENG1 ENG2 ENG3 ENG4 TSPAI	Provide Revision 1 to the Topical Report. DOE stated that it will provide the Disposal Criticality Analysis Methodology Topical Report, Revision 01, to NRC during January 2001.	Received
CLST.5.02	TSPAI	Provide the Disruptive Events FEPs AMR, the FEPs database, and the Analyses to Support Screening of System-Level Features, Events, and Processes for the Yucca Mountain Total System Performance Assessment-Site Recommendation. DOE stated that it will provide the FEPs AMRs, the Analyses to Support Screening of System-Level Features, Events, and Processes for the Yucca Mountain Total System Performance Assessment-Site Recommendation AMR, and the FEPs database to NRC during January 2001.	Received

Agreement	Related ISIs or	NRC/DOE Agreement	Status
CLST.5.03	ENG1 ENG2 TSPAI	DOE will provide an updated technical basis for screening criticality from the post-closure performance assessment. The technical basis will include (1) a determination of whether the formation of condensed water could allow liquid water to enter the waste package without the failure of the drip shield, and (2) an assessment of improper heat treatment, if it is shown to result in early failure of waste packages, considering potential failure modes. The documentation of the technical basis is comprised of (1) Analysis of Mechanisms for Early Waste Package Failure AMR, (2) Probability of Criticality Before 10,000 years calculation, and (3) Features, Event, and Process System Level and Criticality AMR. The first document will be provided to NRC in FY02, the second and third documents will be provided in FY03.	Not Received
CLST.5.04	ENG1 ENG4 SZ2	Provide the list of validation reports and their schedules. DOE stated that the geochemical model validation reports for "Geochemistry Model Validation Report: Degradation and Release" and "Geochemistry Model Validation Report: Material Accumulation" are expected to be available during 2001. The remainder of the reports are expected to be available during FY2002 subject to the results of detailed planning and scheduling. DOE understands that these reports are required to be provided prior to LA. A list of model validation reports was provided during the technical exchange and is included as an attachment to the meeting summary.	Partly Received
CLST.5.05	ENG1 ENG3 ENG4	Provide information on how the increase in the radiation fields due to the criticality event affects the consequence evaluation because of increased radiolysis inside the waste package and at the surfaces of nearby waste packages or demonstrate that the current corrosion and dissolution models encompass the range of chemical conditions and corrosion potentials that would result from this increase in radiolysis. DOE stated that the preliminary assessment (calculation) of radiolysis effects from a criticality event will be available to NRC during February 2001. The final assessment of these conditions will be available to NRC prior to LA.	Partly Received
CLST.5.06	ENG1 ENG2 TSPAI	Provide a "what-if" analysis to evaluate the impact of an early criticality assuming a waste package failure. DOE stated that it would provide the requested analyses prior to LA. Actual schedule to be provided pending DOE planning process.	Not Received
CLST.5.07	ENG1 ENG2 ENG4 TSPAI	Provide sensitivity analyses that will include the most significant probability/consequence criticality scenarios. DOE stated that it would provide the requested analyses prior to LA. Actual schedule to be provided pending DOE planning process.	Not Received
CLST.6.01	ENG1	Provide documentation for the path forward items in the "Subissue 6: Alternate EBS Design Features - Effect on Container Lifetime" presentation, slides 7 & 8. [perform more sensitivity measurements of general corrosion rates - same approach as taken for Alloy 22; confirm no deleterious effects of fluoride ion and trace heavy metal ions in water on corrosion behavior of titanium - similar approach to that taken in electrochemically based studies on Alloy 22; establish damaging hydrogen levels in titanium alloys - Grade 2 vs Grades 7 and 16 vs Grade 5 and 24 - evaluate hydrogen charged notched tensile speciems and hydrogen pickup of galvanically coupled LTCTF specimens; conduct SCC testing of titanium, similar to approach taken for Alloy 22; confirm intergranular or internal oxidation of titanium is not applicable under YM thermal and environmental conditions] DOE stated that the documentation of the path forward items will be completed and as results become available, they will documented in the revisions of AMRs (ANL-EBS-MD-000005, Stress Corrosion Cracking of the Drip Shield, the Waste Package Outer Barrier and the Stainless Structural Material, and ANL-EBS-MD-000004, General Corrosion and Localized Corrosion of the Drip Shield), to be completed by LA.	Not Received
CLST.6.02	ENG1	Provide additional justification for the use of a 400 ppm hydrogen criterion or perform a sensitivity analysis using a lower value. DOE stated that additional justification will be found in the report "Review of Expected Behaviour of Alpha Titanium Alloys under Yucca Mountain Condition" TDR-EBS-MD-000015, which is in preparation and will be available in January 2001.	Received
CLST.6.03	ENG1	Provide the technical basis for the assumed fraction of hydrogen absorbed into titanium as a result of corrosion. DOE stated that additional justification will be found in the report "Review of Expected Behaviour of Alpha Titanium Alloys under Yucca Mountain Condition" TDR-EBS-MD-000015, which is in preparation and will be available in January 2001.	Received

Agreement	Related ISIs or	NRC/DOE Agreement	Status
CLST.6.04	ENG1 PRE	Provide temperature distribution (CCDF) of the drip shield as a function of time under the current EBS design. DOE stated that the temperature distribution will be provided in the next revision of the AMR, ANL-EBS-MD-000049, Rev 00, ICN 01, which will be available in January 2001.	Received
ENFE.1.01	ENG3 TSPAI	Provide updated FEPs AMRs with additional technical bases for those FEPs previously identified by the NRC in Rev. 03 of the ENFE IRSR as inadequately screened. In Rev 03 of the ENFE IRSR, the NRC identified 17 FEPs associated with Subissue 1 for which no screening arguments were identified in the FEPs data base, screening arguments were inconsistent with other project documents, or inadequate exclusion arguments were provided. The lack of screening arguments has been addressed in Rev 00 of the FEPs data base and Rev 00 of the supporting AMRs. Current revisions (or ICNs) of the FEPs AMRs, scheduled for completion in January 2001, will partially address the remaining NRC comments. Consideration of the remaining NRC comments will be provided in subsequent FEPs AMR revisions, expected to be available as periodic revisions, the entirety of which will be available prior to license application.	Received
ENFE.1.02	TSPAI	Provide the FEPs database. The DOE will provide the FEPs data base to the NRC during March 2001.	Received
ENFE.1.03	ENG3 UZ2	Provide the Drift-Scale Coupled Processes (DST and THC Seepage) Models AMR, Rev. 01 and 02, including (1) information on the quantity of unreacted solute mass that is trapped in dry-out zone in TOUGHREACT simulations, as well as how this would affect precipitation and the resulting change in hydrologic properties and (2) documentation of model validation consistent with the DOE QA requirements. The DOE will provide documentation of model validation, consistent with the DOE QA requirements, in the Drift-Scale Coupled Processes (DST and THC Seepage) Models AMR (MDL-NBS-HS-000001) Rev 01, expected to be available to the NRC in March 2001. The DOE will provide information on the quantity of unreacted solute mass that is trapped in the dryout zone in TOUGHREACT simulations in the Drift-Scale Coupled Processes (DST and THC Seepage) Models AMR Rev 02, expected to be available to the NRC in FY 02.	Partly Received
ENFE.1.04	ENG3 UZ2	Provide additional technical bases for the DOE's treatment of the effects of cementitious materials on hydrologic properties. The DOE will provide additional information on the effects of cementitious materials in an update to the Unsaturated Zone Flow and Transport PMR (TDR-NBS-HS-000002), available in FY 02. Information provided will include results of evaluation of the magnitude of potential effects on hydrologic properties and radionuclide transport characteristics of the unsaturated zone.	Not Received
ENFE.1.05	ENG3 UZ2	Address the various sources of uncertainty (e.g., model implementation, conceptual model, and data uncertainty (hydrologic, thermal, and geochemical)) in the THC model. The DOE will evaluate the various sources of uncertainty in the THC process model, including details as to how the propagation of various sources of uncertainty are calculated in a systematic uncertainty analysis. The DOE will document that uncertainty evaluation in the Drift-Scale Coupled Processes (DST and THC Seepage) Models AMR (MDL-NBS-HS-000001) Rev 02 (or in another future document), expected to be available in FY 02.	Not Received
ENFE.1.06	ENG3 TSPAI	Provide the technical basis for excluding entrained colloids in the analysis of FEP 2.2.10.06.00 (Thermo-Chemical Alteration) or an alternative FEP. The DOE will provide the technical basis for screening entrained colloids in the analysis of FEP 2.2.10.06.00 in a future revision of the Features, Events, and Processes in UZ Flow and Transport AMR (ANL-NBS-MD-000001), expected to be available in FY 02.	Not Received
ENFE.1.07	ENG3	Provide physical evidence that supports the model of matrix fracture interaction precipitation effects (e.g., coring). The DOE will provide the following evidence that supports the model of matrix/fracture interaction precipitation effects: (1) Existing data from the Single Heater Test (SHT) of post-test overcoring Mineralogy-Petrology (Min-Pet) analysis (SHT final report [MOL.20000103.0634] and DTN LASL831151.AQ98.001) is expected to be provided to the NRC in March 2001. (2) Results of ongoing side-wall sampling Min-Pet analyses of DST samples are expected to be provided to the NRC in FY 02. (3) The DOE expects to provide the Drift-Scale Coupled Processes (DST and THC Seepage) Models AMR (MDL-NBS-HS-000001) Rev 01 to the NRC as evidence of matrix-fracture interaction in March 2001.	Partly Received

Agreement	Related ISIs or	NRC/DOE Agreement	Status
ENFE.2.01	ENG3 TSPAI	Provide updated FEPs AMRs with additional technical bases for those FEPs previously identified by the NRC in Rev. 03 of the ENFE IRSR as inadequately screened. In Rev 03 of the ENFE IRSR, the NRC identified 24 FEPs associated with Subissue 2 for which no screening arguments were identified in the FEPs data base, screening arguments were inconsistent with other project documents, or inadequate exclusion arguments were provided. The lack of screening arguments has been addressed in Rev 00 of the FEPs data base and Rev 00 of the supporting AMRs. Current revisions (or ICNs) of the FEPs AMRs, scheduled for completion in January 2001, will partially address the remaining NRC comments. Consideration of the remaining NRC comments will be provided in subsequent FEPs AMR revisions, expected to be available as periodic revisions, the entirety of which will be available prior to license application.	Received
ENFE.2.02	TSPAI	Provide the FEPs database. The DOE will provide the FEPs data base to the NRC during March 2001.	Received
ENFE.2.03	ENG3 TSPAI	Provide the technical basis for FEP 1.2.06.00 (Hydrothermal Activity), addressing points (a) through (e) of NRC Subissue 2 slide handed out at the January 2001 ENFE technical exchange. The DOE will provide additional technical bases for the screening of FEP 1.2.06.00 (Hydrothermal Activity), in a future revision of the Features, Events, and Processes in UZ Flow and Transport AMR (ANL-NBS-MD-000001), expected to be available in FY 02. Within these technical bases, the DOE will address NRC comments [points (a) through (e)] presented on the NRC Subissue 2 slide handed out at the January 2001 ENFE technical exchange or provide justification that it is not needed.	Not Received
ENFE.2.04	ENG1 ENG3	Provide the technical basis for bounding the trace elements and fluoride for the geochemical environment affecting the drip shield and waste package, including the impact of engineered materials. The DOE will document the concentrations of trace elements and fluoride in waters that could contact the drip shield and waste package in a revision to the Environment on the Surfaces of the Drip Shield and Waste Package Outer Barrier AMR (ANL-EBS-MD-000001), which will be available in FY02. In addition, trace elements and fluoride concentrations in introduced materials in the EBS (including cement grout, structural steels, and other materials as appropriate) will be addressed in a revision to the Engineered Barrier System: Physical and Chemical Environment Model AMR (ANL-EBS-MD-000033), expected to be available in FY 02.	Not Received
ENFE.2.05	ENG3	Evaluate data and model uncertainties for specific in-drift geochemical environment submodels used in TSPA calculations and propagate those uncertainties following the approach described in Agreement #5, Subissue 1. The DOE will evaluate data and model uncertainties for specific in-drift geochemical environment submodels used in TSPA calculations and propagate those uncertainties following the approach described in Subissue 1, Agreement #5. The DOE will document the evaluation in an update to the Engineered Barrier System: Physical and Chemical Environment Model AMR (ANL-EBS-MD-000033) (or in another future document), expected to be available in FY 02.	Not Received
ENFE.2.06	ENG3	Evaluate the impact of the range of local chemistry (e.g., dripping of equilibrated evaporated cement leachate and corrosion products) conditions at the drip shield and waste package considering the chemical divide phenomena that may propagate small uncertainties into large effects. The DOE will evaluate the range of local chemical conditions at the drip shield and waste package (e.g. local variations in water composition associated with cement leaching or the presence of corrosion products), considering potential evaporative concentration and the chemical divide effect whereby small differences in initial composition could cause large differences in brine characteristics. This evaluation will be documented in a revision to the Engineered Barrier System: Physical and Chemical Environment Model AMR (ANL-EBS-MD-000033), expected to be available in FY 02.	Not Received
ENFE.2.07	ENG3	Identify specific coupling relationships that are included and excluded from TSPA, including Onsager couples, and give technical bases for their inclusion or exclusion. The DOE will identify specific coupling relationships that are included and excluded from TSPA, including Onsager couples, and give the technical basis for inclusion and exclusion. This information will be documented in a revision to the Engineered Barrier System Degradation, Flow, and Transport PMR (TDR-EBS-MD-000006), expected to be available by September 2001.	Not Received
ENFE.2.08	ENG3	Provide stronger technical basis for the suppression of individual minerals predicted by equilibrium models. The DOE will provide additional technical basis for suppression of individual minerals predicted by equilibrium models, in a revision to the Engineered Barrier System: Physical and Chemical Environment Model AMR (ANL-EBS-MD-000033), expected to be available in FY02.	Not Received

Agreement	Related ISIs or	NRC/DOE Agreement	Status
ENFE.2.09	ENG3	Provide the In-Drift Precipitates/Salts Analysis AMR, Rev. 00, ICN 02, including (1) the major anionic (e.g., fluoride or chloride) and cationic species, and (2) additional technical basis for the low relative humidity model. The DOE will provide the In-Drift Precipitates/Salts Analysis AMR (ANL-EBS-MD-000045), Rev. 00, ICN 02, including the major anionic (e.g., fluoride or chloride) and cationic species, in January 2001. The DOE will provide to the NRC an update to the In-Drift Precipitates/Salts Analysis AMR (ANL-EBS-MD-000045) that will provide additional technical bases for the low relative humidity model, expected to be available in FY 02.	Partly Received
ENFE.2.10	ENG3	Provide additional information about the range of composition of waters that could contact the drip shield or waste package, including whether such waters are of the bicarbonate or chloride-sulfate type. The DOE will describe the range of bulk composition for waters that could affect corrosion of the drip shield or waste package outer barrier, in a revision to the Environment on the Surfaces of the Drip Shield and Waste Package Outer Barrier AMR (ANL-EBS-MD-000001), expected to be available in FY02.	Not Received
ENFE.2.11	ENG3	Provide the technical basis for the current treatment of the kinetics of chemical processes in the in-drift geochemical models. This basis should address data in the figure on page 16 of the G.Gdowski Subissue 2 presentation with appropriate treatment of time as related to abstractions used in TSPA. The DOE will provide additional technical basis for the treatment of precipitation-dissolution kinetics by the in-drift geochemical models, in a revision to the Engineered Barrier System: Physical and Chemical Environment Model AMR (ANL-EBS-MD-000033), expected to be available in FY02. The technical basis will include reaction progress simulation for laboratory evaporative concentration tests, and will include appropriate treatment of time as related to the residence times associated with the abstractions used to represent in-drift processes in TSPA.	Not Received
ENFE.2.12	ENG3	Provide the documentation and analysis of the column crush tuff experiments. The DOE will provide documentation of the results obtained from the crushed tuff hydrothermal column experiment, and of post-test analysis, in new reports specific to the column test, expected to be available by September 2001.	Not Received
ENFE.2.13	ENG3	Provide documentation regarding the deposition of dust and its impact on the salt analysis. The DOE will provide documentation of dust sampling in the Exploratory Studies Facility, and analysis of the dust and evaluation of its impact on the chemical environment on the surface of the drip shield and waste package, in a revision to the Engineered Barrier System: Physical and Chemical Environment Model AMR (ANL-EBS-MD-000033), expected to be available in FY02.	Not Received
ENFE.2.14	ENG1 ENG3	Provide the analysis of laboratory solutions that have interacted with introduced materials. The DOE will provide additional information about laboratory solutions that have interacted with introduced materials, in a revision to the Environment on the Surfaces of the Drip Shield and Waste Package Outer Barrier AMR (ANL-EBS-MD-000001), expected to be available in FY02.	Not Received
ENFE.2.15	ENG3	Provide the additional data to constrain the interpolative low relative humidity salts model. The data should provide the technical basis as to why the assumption of the presence of sodium nitrate is conservative, when modeling and experimental results indicate the presence of other mineral phases for which the deliquesence point is unknown. The DOE will provide additional information to constrain the low-relative humidity salts model. The information will include the deliquescence behavior of mineral assemblages derived from alternative starting water compositions (including bulk water compositions, and local variations associated with cement leaching or the presence of corrosion products) representing the range of potential water compositions in the emplacement drifts. This information will be documented in a revision to the In-Drift Precipitates/Salts Analysis AMR (ANL-EBS-MD-000045), expected to be available in FY02.	Not Received
ENFE.2.16	ENG3	Provide the Drift-Scale Coupled Processes (DST and THC Seepage) Models, Rev. 01, including information supporting both the limited suite mineral model and the more complete extended model. The DOE will provide the Drift-Scale Coupled Processes (DST and THC Seepage) Models AMR (MDL-NBS-HS-000001) Rev 01, including information supporting both the limited suite mineral model and the more complete extended model, in March 2001.	Received

Agreement	Related ISIs or	NRC/DOE Agreement	Status
ENFE.2.17	ENG3	Provide documentation of data used to calibrate models and data to support model predictions, and an assessment of data uncertainty (e.g., sampling and analytical), that includes critical analyses of variables that affect the data measurements and their interpretations (e.g., drift-scale thermal test and evaporation tests). The DOE will provide documentation of data used to calibrate models and data to support model predictions, and an assessment of data uncertainty (e.g., sampling and analytical) in the area of water and gas chemistry from the drift-scale thermal tests and evaporation tests. This documentation will be provided in revisions to the following AMRs: Environment on the Surfaces of the Drip Shield and Waste Package Outer Barrier (ANL-EBS-MD-000001), Engineered Barrier System: Physical and Chemical Environment Model (ANL-EBS-MD-000033), and Drift-Scale Coupled Processes (DST and THC Seepage) Models (MDL-NBS-HS-000001), or other documents as appropriate. All documents or revisions are expected to be available in FY 02.	Not Received
ENFE.2.18	ENG3	The NRC and DOE agreed the following documents would be provided with the schedule indicated: Engineered Barrier System: Physical and Chemical Environment Model (ANL-EBS-MD-000033) Rev. 01: FY 02; Multiscale Thermohydrologic Model (ANL-EBS-MD-000049) Rev. 00, ICN 01: January 2001;Abstraction of Drift-Scale Coupled Processes (ANL-NBS-HS-000029) Rev 01: September 2001; Environment on the Surfaces of the Drip Shield and the Waste Package Outer Barrier (ANL-EBS-MD-000001) Rev. 00, ICN 01: January 2001; Waste Package Degradation PMR (TDR-WIS-MD-000002) Rev. 00, ICN 01: January 2001; Engineered Barrier System Degradation, Flow, and Transport PMR (TDR-EBS-MD-000006) Rev. 01: September 2001; Near Field Environment PMR (TDR-NBS-MD-000001) Rev. 00, ICN 02: January 2001 and Rev. 01: September 2001; Hydrogen Induced Cracking of Drip Shield (ANL-EBS-MD-000006) Rev. 00, ICN 01: January 2001; Drift Degradation Analysis (ANL-EBS-MD-000027) Rev. 01: January 2001; Design Analysis for the Ex-Container Components, ANL-XCS-ME-000001 Rev. 00: January 2001; Longevity of Emplacement Drift Ground Support Materials (ANL-EBS-GE-000003) Rev. 01: January 2001; Stress Corrosion Cracking of the Drip Shield, the Waste Package Outer Barrier, and the Stainless Steel Structural Material AMR (ANL-EBS-MD-000005) Rev. 00, ICN 01: January 2001; In-Drift Microbial Communities (ANL-EBS-MD-000038) Rev. 00, ICN 01: January 2001; Physical and Chemical Environmental Abstraction Model (ANL-EBS-MD-000046) Rev. 00, ICN 01: January 2001; Unsaturated Zone Flow and Transport Model PMR (TDR-NBS-HS-000002) Rev. 01: September 2001; General Corrosion and Localized Corrosion of the Drip Shield (ANL-EBS-MD-000004) Rev. 00: January 2001; Water Distribution and Removal Model (ANL-EBS-MD-000032) Rev. 01: January 2001.	Partly Received
ENFE.3.01	ENG1 ENG3	The NRC and DOE agreed the following documents would be provided in February 2001: WAPDEG Analysis of Waste Package and Drip Shield Degradation AMR (ANL-EBS-PA-000001) Rev 00 ICN 01; Near Field Environment PMR (TDR-NBS-MD-000001) Rev 00 ICN 03; Summary of In-Package Chemistry for Waste Forms AMR (ANL-EBS-MD-000050) Rev 01; Calculation of General Corrosion Rate of Drip Shield and Waste Package Outer Barrier to Support WAPDEG Analysis (CAL-EBS-PA-000002) Rev 01; Abstraction of Models for Stainless Steel Structural Material Degradation (ANL-EBS-PA-000005) Rev 00; In-Package Chemistry Abstraction AMR (ANL-EBS-MD-000037) Rev 01; Total System Performance Assessment for the Site Recommendation (TDR-WIS-PA-000001) Rev 00; Waste Form Colloid-Associated Concentrations Limits: Abstraction and Summary AMR (ANL-WIS-MD-000012) Rev 00 ICN 01	Received
ENFE.3.02	ENG3	Provide the thermodynamic database and the report associated with the database. The DOE will provide the thermodynamic data base [Input Transmittal for Thermodynamic Data Input Files for Geochemical Calculations (MO0009THRMODYN.001)] and Data Qualification Report for the Thermodynamic Data File, DATA0.ympR0 for Geochemical Code EQ 3/6 (TDR-EBS-MD-000012) to the NRC in February 2001.	Received
ENFE.3.03	ENG3 ENG4	Provide analyses to verify that bulk-scale chemical processes dominate the in-package chemical environment. The DOE will provide analyses justifying the use of bulk chemistry as opposed to local chemistry for solubility and waste form degradation models. These analyses will be documented in an update to the Miscellaneous Waste-Form FEPs AMR (ANL-WIS-MD-000009) or in an update to the Summary of In-Package Chemistry for Waste Forms AMR (ANL-EBS-MD-000050), expected to be available in FY 02.	Not Received
ENFE.3.04	ENG4	Complete validation of in-package chemistry models. Agreement #5 for CLST subissue 3 addresses testing plans. Model validation based on this testing and further analysis will be documented in an update to the Summary of In-Package Chemistry for Waste Forms AMR (ANL-EBS-MD-000050), expected to be available in FY 02.	Not Received
ENFE.3.05	ENG3 ENG4 UZ3	Provide the technical basis for selection of radionuclides that are released via reversible and irreversible attachment to colloids for different waste forms in the TSPA. The technical bases for the selection of radionuclides released via reversible and irreversible attachments to colloids for different waste forms is provided in section 3.5.6.1 of the Total System Performance Assessment (TSPA) Model for Site Recommendation (MDL-WIS-PA-000002) Rev 00. This document will be provided to the NRC in January 2001.	Received

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ENFE.4.01	ENG3	Provide the executable version of the most recently qualified version of TOUGHREACT. The DOE will provide the executable TOUGHREACT Rev 2.2 to the NRC by February 2001, subject to the NRC obtaining any applicable agreement for usage of the software.	Received
ENFE.4.02	ENG3	Provide the Drift-Scale Coupled Processes (DST and THC Seepage) Models AMR, Rev. 01 and 02. The DOE will provide the Drift-Scale Coupled Processes (DST and THC Seepage) Models AMR (MDL-NBS-HS-000001) Rev 01 to the NRC in March 2001. The DOE will provide the Drift-Scale Coupled Processes (DST and THC Seepage) Models AMR Rev 02 to the NRC in FY 02.	Partly Received
ENFE.4.03	ENG3 TSPAI	Provide the technical bases for screening out coupled THC effects on radionuclide transport properties and colloids. The DOE will provide the technical bases for screening out coupled THC effects on radionuclide transport properties and colloids in a new AMR or in a revision to an existing AMR, expected to be available in FY 02.	Not Received
ENFE.4.04	ENG3 TSPAI	Provide the technical basis for excluding entrained colloids in the analysis of FEP 2.2.10.06.00 (Thermo-Chemical Alteration) or an alternative FEP. The DOE will provide the technical basis for screening entrained colloids in the analysis of FEP 2.2.10.06.00 in a future revision of the Features, Events, and Processes in UZ Flow and Transport AMR (ANL-NBS-MD-000001), expected to be available in FY 02.	Not Received
ENFE.4.05	TSPAI	Provide the screening criteria for the radionuclides selected for PA. Provide the technical basis for selection of radionuclides that are transported via colloids in the TSPA. The screening criteria for radionuclides selected for TSPA are contained in the AMR Inventory Abstraction (ANL-WIS-MD-000006) Rev 00, ICN 01. The DOE is documenting identification of radionuclides transported via colloids for TSPA in the AMR Colloid-Associated Concentration Limits: Abstraction and Summary (ANL-WIS-MD-000012) Rev 0, in the Total System Performance Assessment for the Site Recommendation (TDR-WIS-PA-000001) Rev 00 ICN 01, and in the Total System Performance Assessment (TSPA) Model for Site Recommendation (MDL-WIS-PA-000002) Rev 00. These documents will be available to the NRC in January 2001.	Received
ENFE.4.06	ENG4 TSPAI	Provide documentation to demonstrate suitability of the bounding values used for colloid transport through the perturbed near-field environment. For example, consider sensitivity analyses to investigate the effects of varying colloid sorption parameters (Kc) on repository performance. The DOE will evaluate the suitability of the colloid transport model under perturbed conditions as discussed in agreement #3 for this subissue. As part of this work, the DOE will consider sensitivity analyses to investigate the effects of varying colloid sorption parameters (Kc) on repository performance. The DOE will also provide the TSPA-SR (TDR-WIS-PA-000001) Rev 00 ICN 01 in January 2001. The TSPA-SR includes sensitivity studies in the form of barrier degradation and parameter sensitivity analyses that investigate the effect of sorption and colloid parameters on repository performance.	Partly Received
ENFE.4.07	TSPAI	Provide updated FEPs AMRs with additional technical bases for those FEPs previously identified by the NRC in Rev. 03 of the ENFE IRSR as inadequately screened. In Rev 03 of the ENFE IRSR, the NRC identified 17 FEPs associated with Subissue 1 for which no screening arguments were identified in the FEPs data base, screening arguments were inconsistent with other project documents, or inadequate exclusion arguments were provided. The lack of screening arguments has been addressed in Rev 00 of the FEPs data base and Rev 00 of the supporting AMRs. Current revisions (or ICNs) of the FEPs AMRs, scheduled for completion in January 2001, will partially address the remaining NRC comments. Consideration of the remaining NRC comments will be provided in subsequent FEPs AMR revisions, expected to be available as periodic revisions, the entirety of which will be available prior to license application.	Received
ENFE.4.08	TSPAI	Provide the FEPs database. The DOE will provide the FEPs data base to the NRC during March 2001.	Received
ENFE.5.01	ENG3 TSPAI	Provide Revision 1 to the Topical Report. DOE will provide the Disposal Criticality Analysis Methodology Topical Report, Revision 01, to NRC during January 2001.	Received
ENFE.5.02	TSPAI	Provide the updated FEPs database. DOE stated that it would provide the FEPs AMRs and the FEPs database to NRC during January 2001.	Received

Agreement	Related ISIs or	NRC/DOE Agreement	Status
ENFE.5.03	ENG1	Provide the applicable list of validation reports and their schedules for external criticality. DOE stated that the geochemical model validation reports for "Geochemistry Model Validation Report: Degradation and Release" and "Geochemistry Model Validation Report: Material Accumulation" are expected to be available during 2001. The remainder of the reports are expected to be available during FY2002 subject to the results of detailed planning and scheduling. DOE understands that these reports are required to be provided prior to LA. A list of model validation reports was provided during the technical exchange and is included as an attachment to the meeting summary.	Partly Received
GEN.1.01	ENG1 ENG2 ENG3 ENG4 SZ1 SZ2 UZ2 UZ3	For NRC comments 3, 5, 8, 9, 10, 12, 13, 15, 16, 18, 21, 24, 27, 36, 37, 41, 42, 45, 46, 50, 56, 64, 69, 75, 78, 81, 82, 83, 93, 95, 96, 97, 98, 102, 103, 104, 106, 109, 110, 111, 113, 116, 118, 119, 120, 122, 123, 124, and 126, DOE will address the concern in the documentation for the specific KTI agreement identified in the DOE response (Attachment 2). The schedule and document source will be the same as the specific KTI agreement.	Not Received
IA.1.01	DIRECT1 TSPAI	In addition to DOE's licensing case, include for Site Recommendation and License Application, for information purposes, the results of a single point sensitivity analysis for extrusive and intrusive igneous processes at 10E-7. DOE agreed that the analysis will be included in TSPA-SR Rev. 0 and will be available to the NRC in November 2000.	Complete
IA.1.02	DIRECT1 TSPAI	Examine new aeromagnetic data for potential buried igneous features (see U.S. Geological Survey, Open-File Report 00-188, Online Version 1.0), and evaluate the effect on the probability estimate. If the data survey specifications are not adequate for this use, this action is not required. DOE agreed and will document the results of the evaluation in an update to the AMR, Characterize Framework for Igneous Activity at Yucca Mountain, Nevada (ANL-MGR-GS-000001), expected to be available in FY 2003.	Not Received
IA.2.01	DIRECT2	Re-examine the ASHPLUME Code to confirm that particle density is appropriately changed when waste particles are incorporated into the ash. (Eruptive AC-4) DOE agreed and will correct the description in the ICN to AMR, Igneous Consequences Modeling for TSPA-SR [ANL-WIS-MD-000017] as needed to address the concern. This will be available to the NRC in January 2001.	Complete
IA.2.02	DIRECT2	Document results of sensitivity studies for particle size, consistent with (1) above. (Eruptive AC-4) DOE agreed and will document the waste particle size sensitivity study in a calculation document. This will be available to the NRC in FY2002.	Not Received
IA.2.03	DIRECT2	Document how the tephra volumes from analog volcanos represent the likely range of tephra volumes from Yucca Mountain Region (YMR) volcanos. (Eruptive AC-1) DOE agreed and will document the basis for determining the range of tephra volumes that is likely from possible future volcanoes in the YMR in the Eruptive Processes AMR (ANL-MGR-GS-000002). This will be available to the NRC in FY2002.	Not Received
IA.2.04	DIRECT2	Document that the ASHPLUME model, as used in the DOE performance assessment, has been compared with an analog igneous system. (Eruptive AC-2) DOE agreed and will complete calculation CAL-WIS-MD-000011 that will document a comparison of the ASHPLUME code results to observed data from the 1995 Cerro Negro eruption. This will be available to the NRC in January 2001. DOE will consider Cerro Negro as an analog and document that in the Eruptive Processes AMR (ANL-MGR-GS-000002). This will be available to the NRC in FY2002.	Complete
IA.2.05	DIRECT1	Document how the current approach to calculating the number of waste packages intersected by conduits addresses potential effects of conduit elongation along a drift. (Eruptive AC-3) DOE agreed and will document the way in which the change in geometry of the repository drifts affects the number of waste packages incorporated into the volcanic conduit. Possible consequences of conduit elongation parallel to drifts will be documented in TSPA-SR Rev. 1, available to the NRC in June 2001.	Complete

Agreement	Related ISIs or	NRC/DOE Agreement	Status
IA.2.06	DOSE2 DOSE3	Develop a linkage between soil removal rate used in TSPA and surface remobilization processes characteristics of the Yucca Mountain region (which includes additions and deletions to the system). (Eruptive AC-5) DOE agreed and will document its approach to include uncertainty related to surface-redistribution processes in TSPA-SR, Rev. 0. DOE will revisit the approach in TSPA-SR, Rev. 1. This documentation will be available to the NRC in June 2001.	Complete
IA.2.07	DOSE2 DOSE3	Document the basis for airborne particle concentrations used in TSPA in Rev. 1 to the Input Values for External and Inhalation Radiation Exposure AMR. (Eruptive AC-5) DOE agreed and will provide documentation for the input values in the Input Parameter Values for External and Inhalation Radiation Exposure Analysis AMR [ANL-MGR-MD-000001] Rev. 1. This will be available to NRC in January 2001.	Complete
IA.2.08	DOSE2 DOSE3	Provide additional justification on the reasonableness of the assumption that the inhalation of particles in the 10-100 micron range is treated as additional soil ingestion, or change the BDCFs to reflect ICRP-30. (Eruptive AC-5) DOE agreed and will review how 10-100 micron particles are considered in the model for the eruptive scenario. The results will be documented in Input Parameter Values for External and Inhalation Radiation Exposure Analysis AMR [ANL-MGR-MD-000001] Rev. 1. This will be available to the NRC in January 2001.	Complete
IA.2.09	DIRECT2	Use the appropriate wind speeds for the various heights of eruption columns being modeled. (Eruptive AC-5) DOE agreed and will evaluate the wind speed data appropriate for the height of the eruptive columns being modeled. This will be documented in a calculation document. This will be available to the NRC in FY2002.	Not Received
IA.2.10	DIRECT1 ENG2	Document the ICNs to the Igneous Consequences AMR and the Dike Propagation AMR regarding the calculation of the number of waste packages hit by the intrusion. Include in these or other documents (1) the intermediate results of the releases from Zone 1 and 2, separately, and (2) the evaluation of thermal and mechanical effects, as well as shock, in assessing the degree of waste package damage in Zone 1 and 2. (Intrusive AC 4) DOE agreed and will provide ICN 1 of the following AMRs: Igneous Consequences Modeling for TSPA-SR AMR [ANL-WIS-MD-000017], the Dike Propagation Near Drifts AMR [ANL-WIS-MD-000015], the Characterize Framework for Igneous Activity at Yucca Mountain, Nevada AMR [ANL-MGR-GS-000001], and the Calculation Number of Waste Packages Hit by Igneous Intrusion [CAL-WIS-PA-000001]. This will be available to the NRC in January 2001. DOE will provide the results showing the relative contributions of releases from Zones 1 and 2 in a calculation document. This will be available to the NRC in FY2002. DOE will provide the evaluation of thermal mechanical effects on waste package damage in Zones 1 and 2 in ICN 1 of the Dike Propagation Near Drifts AMR [ANL-WIS-MD-000015]. This will be available to the NRC in January 2001.	Partly Received
IA.2.11	DOSE2 DOSE3	Provide an analysis that shows the relationship between any static measurements used in the TSPA and expected types and durations of surface disturbing activities associated with the habits and lifestyles of the critical group. DOE will provide an analysis that shows the relationship between any static measurements used in the TSPA and expected types and durations of surface disturbing activities associated with the habits and lifestyles of the critical group in a subsequent revision to the AMR Input Parameter Values for External and Inhalation Radiation Exposure Analysis (ANL-MGR-MD-000001) or equivalent document. This will be available to the NRC in FY02.	Not Received
IA.2.12	DOSE2 DOSE3	Provide clarifying information on how PM10 measurements have been extrapolated to TSP concentrations. This should include consideration of the difference in behavior between PM10 and TSP particulates under both static and disturbed conditions. DOE will provide clarifying information on how PM10 measurements have been extrapolated to TSP concentrations. This will include consideration of the difference in behavior between PM10 and TSP particulates under both static and disturbed conditions in a subsequent revision to the AMR Input Parameter Values for External and Inhalation Radiation Exposure Analysis (ANL-MGR-MD-000001) or equivalent document. This will be available to the NRC in FY02.	Not Received
IA.2.13	DOSE2 DOSE3	Provide the justification that sampling of range of transition period BDCFs is necessarily conservative in evaluating long-term remobilization processes. DOE will provide the justification that sampling of range of transition period BDCFs is necessarily conservative in evaluating long-term remobilization processes in a subsequent revision to the AMR Input Parameter Values for External and Inhalation Radiation Exposure Analysis (ANL-MGR-MD-000001) or equivalent document. This will be available to the NRC in FY02.	Not Received

Agreement	Related ISIs or	NRC/DOE Agreement	Status
IA.2.14	DOSE2 DOSE3	Provide information clarifying the method used in TSPA to calculate how deposit thickness effects the average mass load over the transition period. DOE will provide information clarifying the method used in TSPA to calculate how deposit thickness effects the average mass load over the transition period in a subsequent revision to the AMR Input Parameter Values for External and Inhalation Radiation Exposure Analysis (ANL-MGR-MD-000001) or equivalent document. This will be available to the NRC in FY02.	Not Received
IA.2.15	DOSE2 DOSE3	Clarify that external exposure from HLW-contaminated ash, in addition to inhalation and ingestion, was considered in the TSPA. Include in this clarification the consideration of external exposure during indoor occupancy times, or provide basis for dwelling shielding from outdoor gamma emitters. DOE will clarify that external exposure from HLW-contaminated ash, in addition to inhalation and ingestion, was considered in the TSPA. DOE will include in this clarification the consideration of external exposure during indoor occupancy times, or provide basis for dwelling shielding from outdoor gamma emitters in a subsequent revision to the AMR Input Parameter Values for External and Inhalation Radiation Exposure Analysis (ANL-MGR-MD-000001) or equivalent document. This will be available to the NRC in FY02.	Not Received
IA.2.16	DOSE2 DOSE3	Document that neglecting the effects of climate change on disruptive event BDCFs is conservative. DOE will document that neglecting the effects of climate change on disruptive event BDCFs is conservative in a subsequent revision to the AMRs Input Parameter Values for External and Inhalation Radiation Exposure Analysis (ANL-MGR-MD-000001) and Disruptive Event Biosphere Dose Conversion Factor Analysis (ANL-MGR-MD-000003) or equivalent document. This will be available to the NRC in FY02.	Not Received
IA.2.17	DOSE2 DOSE3	DOE will evaluate conclusions that the risk effects (i.e., effective annual dose) of eolian and fluvial remobilization are bounded by conservative modeling assumptions in the TSPA-SR, Rev 00, ICN1. DOE will examine rates of eolian and fluvial mobilization off slopes, rates of transport in Fortymile Wash, and rates of deposition or removal at proposed critical group location. DOE will evaluate changes in grain size caused by these processes for effects on airborne particle concentrations. DOE will also evaluate the inherent assumption in the mass loading model that the concentration of radionuclides on soil in the air is equivalent to the concentration of radionuclides on soil on the ground does not underestimate dose (i.e., radionuclides important to dose do not preferentially attach to smaller particles). DOE will document the results of investigations in the AMR, Eruptive Processes and Soil Redistribution ANL-MGR-GS-000002, expected to be available in fiscal year 2003 and in the AMR, Input Parameter Values for External and Inhalation Radiation Exposure Analysis, ANL-MGR-MD-000001, available FY 2003, or another appropriate technical document.	Not Received
IA.2.18	DIRECT1 ENG2	DOE will evaluate how the presence of repository structures may affect magma ascent, conduit localization, and evolution of the conduit and flow system. The evaluation will include the potential effects of topography and stress, strain response on existing or new geologic structures resulting from thermal loading of HLW, in addition to a range of physical conditions appropriate for the duration of igneous events. DOE will also evaluate how the presence of engineered repository structures in the LA design (e.g., drifts, waste packages, backfill, etc.) could affect magma flow processes for the duration of an igneous event. The evaluation will include the mechanical strength and durability of natural or engineered barriers that could restrict magma flow within intersected drifts. The results of this investigation will be documented in an update to the AMR, Dike Propagation and Interaction with Drifts, ANL-WIS-MD-000015, expected to be available in FY 2003, or another appropriate technical document.	Not Received
IA.2.19	DIRECT1 ENG2	DOE will evaluate waste package response to stresses from thermal and mechanical effects associated with exposure to basaltic magma, considering the results of evaluations attendant to IA Agreement 2.18. As currently planned, the evaluation, if implemented, would include (1) appropriate at-condition strength properties and magma flow paths, for duration of an igneous event; and (2) aging effects on materials strength properties when exposed to basaltic magmatic conditions for the duration of an igneous event, which will include the potential effects of subsequent seismically induced stresses on substantially intact waste packages. DOE will also evaluate the response of Zone 3 waste packages, or waste packages covered by backfill or rockfall, if exposed to magmatic gases at conditions appropriate for an igneous event, considering the results of evaluations attendant to IA Agreement 2.18. If models take credit for engineered barriers providing delay in radionuclide release, DOE will evaluate barrier performance for the duration of the hypothetical igneous event. The results of this investigation would be documented in an update to the technical product Waste Package Behavior in Magma CAL-EBS-ME-000002, which would be available by the end of FY 2003, or another appropriate technical document.	Not Received

Agreement	Related ISIs or	NRC/DOE Agreement	Status
IA.2.20	DIRECT1 DIRECT2 ENG2	DOE will evaluate how ascent and flow of basaltic magma through repository structures could result in processes that might incorporate HLW, considering the results of evaluations attendant to IA Agreements 2.18 and 2.19. As currently planned, the evaluation, if implemented, would include the potential for HLW incorporation along reasonable potential flow paths that could develop during an igneous event. The evaluation would also include the physical and chemical response of HLW and cladding after heating and potential disruption of waste package and contents, for waste packages remaining in drifts. The evaluation would examine effects that may result in increased solubility potential relative to undisturbed HLW forms. The results of this investigation would be documented in a new AMR to document the waste form response to magmatic conditions, which is expected to be available by the end of FY 2003. DOE will describe the method of HLW incorporation used in DOE models, including consideration of particle aggregation and the effect on waste transport. If models take credit for engineered barriers providing delay in radionuclide release, DOE will evaluate barrier performance for the duration of the hypothetical igneous event. This will be documented in an update to the igneous consequences AMR, ANL-WIS-MD-000017, which is expected to be available in FY 2003, or another appropriate technical document.	Not Received
PRE.03.01	PRE	Provide a plan for identification and estimation of aircraft hazards for the license application. This plan should be consistent with the guidelines in NUREG-0800 and other applicable DOE standards, as appropriate, to a nuclear waste repository. Provide a map delineating the vicinity to be considered in the detailed analysis, taking into consideration available information for civilian and military aircraft, including information from federal and local agencies concerning how such activities may reasonably change. Participate in an Appendix 7 meeting to discuss the aircraft hazards plan, initial data collection and analysis, development of the vicinity map, and the appropriate level of detail for analyses to be presented in the license application assessment. DOE agrees with the request and will provide the plan and map in June 2002. DOE agrees to participate in an Appendix 7 meeting which will be scheduled after the plan and map are provided.	Not Received
PRE.03.02	PRE	Provide an analysis, including (1) selection of the design basis tornado, together with the supporting technical basis; (2) selection of credible tornado missile characteristics for the waste package and other structures, systems, and components, together with the technical bases; and (3) analysis of the effects of impact of the design basis tornado missiles or justification for excluding such tornado missiles as credible hazards. DOE agrees to provide the analysis. The analysis will be available in FY03 and be documented in an update to ANL-MGR-SE-000001 and any other appropriate documents.	Not Received
PRE.06.01	PRE	Provide the update to Quality Assurance Procedure QAP 2-3. DOE agreed to provide the procedure. The procedure will be available in February 2002.	Not Received
PRE.06.02	PRE	Provide the Integrated Safety Analysis Guide. DOE agreed to provide the guide. The guide will be available in February 2002.	Not Received
PRE.07.01	PRE	Provide an update to the Pre-Closure Criticality Analysis Process Report. DOE agreed to provide the report. The report will be available in FY03.	Not Received
PRE.07.02	PRE	Provide the waste package finite element analysis based numerical simulations that represent a significant contribution to DOE's safety case. Provide documentation demonstrating that a sufficient finite element model mesh discretization has been used and the failure criterion adequately bounds the uncertainties associated with effects not explicitly considered in the analysis. These uncertainties include but are not limited to: (1) residual and differential thermal expansion stresses, (2) strain rate effects, (3) dimensional and material variability, (4) seismic effects on ground motion, (5) initial tip-over velocities, and (6) sliding and inertial effects of the waste package contents, etc. In addition, document the loads and boundary conditions used in the models and provide the technical bases and or rationale for them. DOE agreed to provide the information. The information will be available in FY03 and documented in Waste Package Design Methodology Report.	Not Received
PRE.07.03	PRE	Demonstrate that the allowed microstructural and compositional variations of alloy 22 base metal and the allowed compositional variations in the weld filler metals used in the fabrication of the waste packages do not result in unacceptable waste package mechanical properties. DOE will provide justification that the ASME code case for alloy 22 results in acceptable waste package mechanical properties considering allowed microstructural and compositional variations of alloy 22 base metal and the allowed compositional variations in the weld filler metals used in the fabrication of the waste packages. DOE agrees to provide the information in FY03 and document the information in the Waste Package Design Methodology Report.	Not Received

Agreement	Related ISIs or	NRC/DOE Agreement	Status
PRE.07.04	PRE	Demonstrate that the non-destructive evaluation methods used to inspect the alloy 22 and 316 nuclear grade plate material and closure welds are sufficient and are capable of detecting all defects that may alter waste package mechanical properties. DOE will provide justification that the non-destructive evaluation methods used to inspect the alloy 22 and 316 nuclear grade plate material and welds are sufficient and are capable of detecting defects that may adversely affect waste package pre-closure structural performance. DOE agrees to provide the information in FY03 and document the information in the Waste Package Operations Fabrication Process Report.	Not Received
PRE.07.05	PRE	Provide justification that the mechanical properties of the disposal container fabrication and waste package closure welds are adequately represented considering the (1) range of welding methods used to construct the disposal containers, (2) post weld annealing and stress mitigation processes, and (3) post weld repairs. DOE agrees to provide the information in FY03 and document the information in the Waste Package Operations Fabrication Process Report.	Not Received
RDTME.2.01	PRE	Provide Topical Report 3, Preclosure Seismic Design Inputs for a Geologic Repository at Yucca Mountain. Consistent with SDS Subissue 2, Agreement 2, the DOE will provide Seismic Topical Report 3, Preclosure Seismic Design Inputs for a Geologic Repository at Yucca Mountain, expected to be available to the NRC in January 2002.	Not Received
RDTME.2.02	PRE	Provide the substantive technical content of Topical Report 3. The DOE will provide the preliminary seismic design input data sets used in Site Recommendation design analyses to the NRC by April 2001. The DOE will provide the draft final seismic design inputs for license application via an Appendix 7 meeting after calculations are complete prior to delivery of Seismic Topical Report 3.	Partly Received
RDTME.3.01	PRE	Provide the technical basis for the range of relative humidities, as well as the potential occurrence of localized liquid phase water, and resulting affects on ground support systems. The DOE will provide the technical basis for the range of relative humidity and temperature, and the potential effects of localized liquid phase water on ground support systems, during the forced ventilation preclosure period, in the Longevity of Emplacement Drift Ground Support Materials, ANL-EBS-GE-000003 Rev 01, and revision 1 of the Ventilation Model, ANL-EBS-MD-000030, analysis and model reports. These are expected to be available to NRC in September and March 2001, respectively.	Partly Received
RDTME.3.02	PRE	Provide the critical combinations of in-situ, thermal, and seismic stresses, together with their technical bases, and their impacts on ground support performance. The DOE will examine the critical combinations of in-situ, thermal, and seismic stresses, together with their technical bases and their impacts on preclosure ground support performance. These results will be documented in a revision to the Ground Control for Emplacement Drifts for SR, ANL-EBS-GE-000002 (or other document) supporting any potential license application. This is expected to be available to NRC in FY 2003.	Not Received
RDTME.3.03	ENG2 PRE	Provide the Seismic Design Inputs AMR and the Preclosure Seismic Design Inputs for a Geologic Repository at Yucca Mountain, Seismic Topical Report 3. Consistent with SDS Subissue 2, Agreement 2, the DOE will provide the Seismic Design Inputs analysis and model report and Preclosure Seismic Design Inputs for a Geologic Repository at Yucca Mountain, Seismic Topical Report 3. These documents are expected to be available to NRC in January 2002.	Not Received
RDTME.3.04	PRE	Provide in the Design Parameter Analysis Report (or some other document) site-specific properties of the host rock, as a minimum those included in the NRC handout, together with the spatial and temporal variations and uncertainties in such properties, as an update to the information contained in the March 1997 Yucca Mountain Site Geotechnical Report. The DOE will: (1) evaluate the adequacy of the currently available measured and derived data to support the potential repository licensing case and identify areas where available data may warrant additional field measurements or testing to reduce uncertainty. DOE will provide a design parameters analysis report (or other document) that will include the results of these evaluations, expected to be available to NRC in FY 2002; and (2) acquire data and/or perform additional analyses as necessary to respond to the needs identified in 1 above. The DOE will provide these results prior to any potential license application.	Not Received
RDTME.3.05	PRE	Provide the Rock Mass Classification Analysis (or some other document) including the technical basis for accounting for the effects of lithophysae. The DOE will provide a rock mass classification analysis (or other document), including the technical basis for accounting for the effects of lithophysae, expected to be available to NRC in FY 2002.	Not Received

Agreement	Related ISIs or	NRC/DOE Agreement	Status
RDTME.3.06	PRE	Provide the design sensitivity and uncertainty analyses of the rock support system. The DOE will prepare a scoping analysis to determine the significance of the input parameters for review by NRC staff by August 2002. Once an agreed set of significant parameters has been determined by the DOE and the NRC staff, the DOE will prepare an analysis of the sensitivity and uncertainty of the preclosure rock support system to design parameters in a revision to the Ground Control for Emplacement Drifts for SR, ANL-EBS-GE-000002 (or other document) supporting any potential license application. This is expected to be available to NRC in FY 2003.	Not Received
RDTME.3.07	PRE	The DOE should account for the effect of sustained loading on intact rock strength or provide justification for not accounting for it. The DOE will assess the effects of sustained loading on intact rock strength. The DOE will provide the results of this assessment in a design parameters analysis report (or other document), expected to be available to NRC in FY 2002.	Not Received
RDTME.3.08	PRE	Provide the design sensitivity and uncertainty analyses of the fracture pattern (with respect to Subissue 3, Component 1). The DOE will provide sensitivity and uncertainty analysis of fracture patterns (based on observed orientation, spacing, trace length, etc) on the preclosure ground control system design in a revision to the Ground Control for Emplacement Drifts for SR, ANL-EBS-GE-000002 (or other document) supporting any potential license application. This is expected to be available to NRC in FY 2003.	Not Received
RDTME.3.09	PRE	Provide appropriate analysis that shows that rock movements in the invert are either controlled or otherwise remain within the range acceptable to provide for retrieval and other necessary operations within the deposal drifts. DOE will provide appropriate analysis that shows rock movements in the floor of the emplacement drift are within the range acceptable for preclosure operations. The analysis results will be provided in a revision to the Ground Control for Emplacement Drifts for SR, ANL-EBS-GE-000002 (or other document) supporting any potential license application. This is expected to be available to NRC in FY 2003.	Not Received
RDTME.3.10	PRE	Provide technical basis for the assessment that two-dimensional modeling for emplacement drifts is considered to be adequate, considering the fact that neither the in-situ stress field nor the principle fracture orientation are parallel or perpendicular to emplacement drift orientation. The DOE will provide the technical bases for the modeling methods used in ground control analysis in a revision to the Ground Control for Emplacement Drifts for SR, ANL-EBS-GE-000002 (or other document) supporting any potential license application. This is expected to be available to NRC in FY 2003.	Not Received
RDTME.3.11	PRE	Provide continuum and discontinuum analyses of ground support system performance that take into account long-term degradation of rockmass and joint strength properties. The DOE will justify the preclosure ground support system design (including the effects of long term degradation of rock mass and joint strength properties) in a revision to the Ground Control for Emplacement Drifts for SR, ANL-EBS-GE-000002 (or other document) supporting any potential license application. This is expected to be available to NRC in FY 2003.	Not Received
RDTME.3.12	PRE	Provide dynamic analyses (discontinuum approach) of ground support system performance using site specific ground motion time history as input. The DOE will provide appropriate analyses to include dynamic analyses (discontinuum approach) of preclosure ground support systems, using site specific ground motion time histories as input, in a revision to the Ground Control for Emplacement Drifts for SR, ANL-EBS-GE-000002 (or other document) supporting any potential license application. This is expected to be available to NRC in FY 2003.	Not Received
RDTME.3.13	PRE	Provide technical justification for boundary conditions used for continuum and discontinuum modeling used for underground facility design. The DOE will provide the technical justification for boundary conditions used in modeling for preclosure ground control analyses in a revision to the Ground Control for Emplacement Drifts for SR, ANL-EBS-GE-000002 (or other document) supporting any potential license application. This is expected to be available to NRC in FY 2003.	Not Received
RDTME.3.14	PRE UZ2	Provide the results of the ventilation modeling being conducted at the University of Nevada-Reno (Multi-Flux code) and validation testing at the Atlas Facility (validation of the ventilation model based on the ANSYS code), including: 1) the technical bases for the adequacy of discretization used in these models and 2) the technical bases for the applicability of the modeling results to prediction of heat removal from the repository. The DOE will provide the results of the ventilation tests in a update to the Ventilation Model, ANL-EBS-MD-000030, analysis and model report including: 1) the technical bases for the adequacy of discretization used in these models and 2) the technical bases for the applicability of the modeling results to prediction of heat removal from the repository. This is expected to be available to NRC in FY 2002.	Not Received

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RDTME.3.15	ENG2	Provide field data and analysis of rock bridges between rock joints that are treated as cohesion in DRKBA modeling together with a technical basis for how a reduction in cohesion adequately accounts for thermal effects. The DOE will provide clarification of the approach and technical basis for how reduction in cohesion adequately accounts for thermal effects, including any additional applicable supporting data and analyses. Additionally, the adequacy of the cohesion reduction approach will be verified according to the approach described in Subissue 3, Agreement 19, of the Repository Design and Thermal-Mechanical Effects Technical Exchange. This will be documented in a revision to the Drift Degradation Analysis, ANL-EBS-MD-000027, expected to be available to NRC in FY 2003.	Not Received
RDTME.3.16	ENG2	Provide a technical basis for the DOE position that the method used to model joint planes as circular discs does not under-represent the smaller trace-length fractures. The DOE will analyze the available small trace-length fracture data from the Exploratory Studies Facility and Enhanced Characterization of the Repository Block, including their effect on block development. This will be documented in a revision to the Drift Degradation Analysis, ANL-EBS-MD-000027, expected to be available to NRC in FY 2003.	Not Received
RDTME.3.17	ENG2	Provide the technical basis for effective maximum rock size including consideration of the effect of variation of the joint dip angle. The DOE will provide the technical basis for effective maximum rock size including consideration of the effect of variation of the joint dip angle. This will be documented in revisions to the Drift Degradation Analysis, ANL-EBS-MD-000027, and the Rockfall on Drip Shield, CAL-EBS-ME-000001, expected to be available to NRC in FY 2003.	Not Received
RDTME.3.18	ENG1 ENG2	Provide a technical basis for a stress measure that can be used as the equivalent uniaxial stress for assessing the susceptibility of the various engineered barrier system materials to stress corrosion cracking (SCC). The proposed stress measure must be consistent and compatible with the methods proposed by the DOE to assess SCC of the containers in WAPDEG and in accordance with the agreements reached at the CLST Technical Exchange. The DOE will include a detailed discussion of the stress measure used to determine nucleation of stress corrosion cracks in the calculations performed to evaluate waste package barriers and the drip shield against stress corrosion cracking criterion. DOE will include these descriptions in future revisions of the following: Design Analysis for UCF Waste Packages, ANL-UDC-MD-000001, Design Analysis for the Defense High-Level Waste Disposal Container, ANL-DDC-ME-000001, Design Analysis for the Naval SNF Waste Package, ANL-UDC-ME-000001, and Design Analysis for the Ex-Container Components, ANL-XCS-ME-000001. The stresses reported in these documents will be used in WAPDEG and will be consistent with the agreements and associated schedule made at the Container Life and Source Term Technical Exchange (Subissue 1, Agreement 14, Subissue 6, Agreement 1).	Not Received

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RDTME.3.19	ENG2 TSPAI	The acceptability of the process models that determine whether rockfall can be screened out from performance assessment abstractions needs to be substantiated by the DOE by doing the following: (1) provide revised DRKBA analyses using appropriate range of strength properties for rock joints from the Design Analysis Parameters Report, accounting for their long-term degradation; (2) provide an analysis of block sizes based on the full distribution of joint trace length data from the Fracture Geometry Analysis Report for the Stratigraphic Units of the Repository Host Horizon, including small joints trace lengths; (3) verify the results of the revised DRKBA analyses using: (a) appropriate boundary conditions for thermal and seismic loading; (b) critical fracture patterns from the DRKBA Monte Carlo simulations (at least two patterns for each rock unit); (c) thermal and mechanical properties for rock blocks and joints from the Design Analysis Parameters Report; (d) long-term degradation of rock block and joint strength parameters; and (e) site-specific groundmotion time histories appropriate for post-closure period; provide a detailed documentation of the analyses results; and (4) in view of the uncertainties related to the rockfall analyses and the importance of the outcome of the analyses to the performance of the repository, evaluate the impacts of rockfall in performance assessment calculations. DOE believes that the Drift Degradation Analysis is consistent with current understanding of the Yucca Mountain site and the level of detail of the design to date. As understanding of the site and the design evolve, DOE will: (1) provide revised DRKBA analyses using appropriate range of strength properties for rock joints from a design parameters analysis report (or other document), accounting for their long-term degradation; (2) provide an analysis of block sizes based on the full distribution of joint trace length data from the Fracture Geometry Analysis for the Stratigraphic Units of the Repository Host Horizon, ANL-EBS-GE-000006, supplemented by available small joint trace length data; (3) verify the results of the revised DRKBA analyses using: (a) appropriate boundary conditions for thermal and seismic loading; (b) critical fracture patterns from the DRKBA Monte Carlo simulations (at least two patterns for each rock unit); (c) thermal and mechanical properties for rock blocks and joints from a design parameters analysis report (or other document); (d) long-term degradation of joint strength parameters; and (e) site-specific ground motion time histories appropriate for post-closure period. This will be documented in a revision to the Drift Degradation Analysis, ANL-EBS-MD-000027, expected to be available to NRC in FY 2003. Based on the results of the analyses above and subsequent drip shield calculation revisions, DOE will reconsider the screening decision for inclusion or exclusion of rockfall in performance assessment analysis. Any changes to screening decisions will be documented in analyses prior to any potential license application.	Not Received
RDTME.3.20	ENG3 UZ2	Provide the sensitivity analyses including the effects of boundary conditions, coefficient of thermal expansion, fracture distributions, rock mass and fracture properties, and drift degradation (from Subissue 3, Component 3, Slide 39). The DOE will provide sensitivity analyses of thermal-mechanical effects on fracture permeability, including the effects of boundary conditions, coefficient of thermal expansion, fracture distributions, rock mass and fracture properties, and drift degradation. This will be provided consistent with site data and integrated with appropriate models in a future revision to the Coupled Thermal Hydrologic Mechanical Effects on Permeability, ANL-NBS-HS-000037, and is expected to be available to NRC in FY 2003.	Not Received
RDTME.3.21	ENG3 UZ2	Provide the results of additional validation analysis of field tests (from Subissue 3, Component 3, Slide 39). The DOE will provide the results of additional validation analysis of field tests related to the thermal-mechanical effects on fracture permeability in a future revision to the Coupled Thermal Hydrologic Mechanical Effects on Permeability, ANL-NBS-HS-000037, and is expected to be available to NRC in FY 2003.	Not Received
RT.1.01	UZ2 UZ3	Provide the basis for the proportion of fracture flow through the Calico Hills non-welded vitric. DOE will revise the AMR UZ Flow Models and Submodels and the AMR Calibrated Properties Model to provide the technical basis for the proportion of fracture flow through the Calico Hills Nonwelded Vitric. These reports will be available to the NRC in FY 2002. In addition, the field data description will be documented in the AMR In Situ Field Testing of Processes in FY 2002.	Not Received
RT.1.02	SZ2 UZ3	Provide analog radionuclide data from the tracer tests for Calico Hills at Busted Butte and from similar analog and radionuclide data (if available) from test blocks from Busted Butte. DOE will provide data from tracers used at Busted Butte and data from (AECL) test blocks from Busted Butte in an update to the AMR In Situ Field Testing of Processes in FY 2002.	Not Received
RT.1.03	SZ2 TSPAI UZ3	Provide the screening criteria for the radionuclides selected for PA. Provide the technical basis for selection of the radionuclides that are transported via colloids in the TSPA. The screening criteria for radionuclides selected for TSPA are contained in the AMR Inventory Abstraction. DOE is documenting identification of radionuclides transported via colloids for TSPA in the AMR Waste Form Colloid-Associated Concentration Limits: Abstraction and Summary, in the TSPA-SR Technical Report, and in the TSPA-SR Model Document. These documents will be available to the NRC in January 2001.	Received

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RT.1.04	SZ2 UZ3	Provide sensitivity studies on Kd for plutonium, uranium, and protactinium to evaluate the adequacy of the data. DOE will analyze column test data to determine whether, under the flow rates pertinent to the Yucca Mountain flow system, plutonium sorption kinetics are important to performance. If they are found to be important, DOE will also perform sensitivity analyses for uranium, protactinium, and plutonium to evaluate the adequacy of KD data. The results of this work will be documented in an update to the AMR Unsaturated Zone and Saturated Zone Transport Properties available to the NRC in FY 2002.	Not Received
RT.1.05	SZ1 SZ2 UZ3	Provide additional documentation to explain how transport parameters used for performance assessment were derived in a manner consistent with NUREG-1563, as applicable. Consistent with the less structured approach for informal expert judgment acknowledged in NUREG-1563 guidance and consistent with DOE procedure AP-3.10Q, DOE will document how it derived the transport parameter distributions for performance assessment, in a report expected to be available in FY 2002.	Not Received
RT.2.01	SZ1 SZ2	Provide further justification for the range of effective porosity in alluvium, considering possible effects of contrasts in hydrologic properties of layers observed in wells along potential flow paths. DOE will use data obtained from the Nye County Drilling Program, available geophysical data, aeromagnetic data, and results from the Alluvium Testing Complex testing to justify the range of effective porosity in alluvium, considering possible effects of contrasts in hydrologic properties of layers observed in wells along potential flowpaths. The justification will be provided in the Alluvial Testing Complex AMR due in FY 2003.	Not Received
RT.2.02	SZ1 SZ2 TSPAI	The DOE should demonstrate that TSPA captures the spatial variability of parameters affecting radionuclide transport in alluvium. DOE will demonstrate that TSPA captures the variability of parameters affecting radionuclide transport in alluvium. This information will be provided in the TSPA-LA document due in FY 2003.	Not Received
RT.2.03	SZ1 SZ2	Provide a detailed testing plan for alluvial testing (the ATC and Nye County Drilling Program) to reduce uncertainty (for example, the plan should give details about hydraulic and tracer tests at the well 19 complex and it should also identify locations for alluvium complex testing wells and tests and logging to be performed). NRC will review the plan and provide comments, if any, for DOE's consideration. In support and preparation for the October/November 2000 Saturated Zone meeting, DOE provided work plans for the Alluvium Testing Complex and the Nye County Drilling Program (FWP-SBD-99-002, Alluvial Tracer Testing Field Work Package, and FWP-SBD-99-001, Nye County Early Warning Drilling Program, Phase II and Alluvial Testing Complex Drilling). DOE will provide test plans of the style of the Alcove 8 plan as they become available. The plan will be amended to include laboratory testing. In addition, the NRC On Site Representative attends DOE/Nye County planning meetings and is made aware of all plans and updates to plans as they are made.	Not Received
RT.2.04	SZ1 SZ2	The NRC needs DOE to document the pre-test predictions for the ATC. DOE will document pretest predictions for the Alluvial Testing Complex in the SZ In Situ Testing AMR available in October 2001.	Not Received
RT.2.05	SZ2	Provide the laboratory testing plan for laboratory radionuclide transport studies. NRC will review the plan and provide comments, if any, for DOE's consideration. In support and preparation for the October/November 2000 Saturated Zone meeting, DOE provided work plans for the Alluvium Testing Complex and the Nye County Drilling Program (FWP-SBD-99-002, Alluvial Tracer Testing Field Work Package, and FWP-SBD-99-001, Nye County Early Warning Drilling Program, Phase II and Alluvial Testing Complex Drilling). DOE will provide test plans of the style of the Alcove 8 plan as they become available. The plan will be amended to include laboratory testing. In addition, the NRC On Site Representative attends DOE/Nye County planning meetings and is made aware of all plans and updates to plans as they are made.	Not Received

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RT.2.06	SZ2	If credit is taken for retardation in alluvium, the DOE should conduct Kd testing for radionuclides important to performance using alluvium samples and water compositions that are representative of the full range of lithologies and water chemistries present within the expected flow paths (or consider alternatives such as testing with less disturbed samples, use of samples from more accessible analog sites (e.g., 40-mile Wash), detailed process level modeling, or other means). DOE will conduct Kd experiments on alluvium using samples from the suite of samples obtained from the existing drilling program; or, DOE will consider supplementing the samples available for testing from the alternatives presented by the NRC. This information will be documented in an update to the SZ In Situ Testing AMR, available in FY 2003. Kd parameter distributions for TSPA will consider the uncertainties that arise from the experimental methods and measurements.	Not Received
RT.2.07	SZ2	Provide the testing results for the alluvial and laboratory testing. DOE will provide testing results for the alluvial field and laboratory testing in an update to the SZ In Situ Testing AMR available in FY 2003.	Not Received
RT.2.08	SZ1	Provide additional information to further justify the uncertainty distribution of flow path lengths in the alluvium. This information currently resides in the Uncertainty Distribution for Stochastic Parameters AMR. DOE will provide additional information, to include Nye County data as available, to further justify the uncertainty distribution of flowpath lengths in alluvium in updates to the Uncertainty Distribution for Stochastic Parameters AMR and to the Saturated Zone Flow and Transport PMR, both expected to be available in FY 2002.	Not Received
RT.2.09	SZ1	Provide the hydro-stratigraphic cross-sections that include the Nye County data. DOE will provide the hydrostratigraphic cross sections in an update to the Hydrogeologic Framework Model for The Saturated Zone Site-Scale Flow and Transport Model AMR expected to be available during FY 2002, subject to availability of Nye County data.	Not Received
RT.2.10	SZ2 TSPAI UZ3	Provide additional documentation to explain how transport parameters used for PA were derived in a manner consistent with NUREG-1563, as applicable. Consistent with the less structured approach for informal expert judgment acknowledged in NUREG-1563 guidance and consistent with AP-3.10Q, DOE will document how it derived the transport distributions for performance assessment, in a report expected to be available in FY 2002.	Not Received
RT.2.11	SZ1 TSPAI	Provide the updated UZ Flow and Transport and the SZ Flow and Transport FEPs AMRs. DOE will provide updates to the AMRs Features, Events, and Processes in UZ Flow and Transport and Features, Events, and Processes in SZ Flow and Transport, both available in January 2001.	Received
RT.3.01	SZ1 UZ3	For transport through fault zones below the repository, provide the technical basis for parameters/distributions (consider obtaining additional information, for example, the sampling of wells WT-1 and WT-2), or show the parameters are not important to performance. DOE will provide a technical basis for the importance to performance of transport through fault zones below the repository. This information will be provided in an update to the AMR Radionuclide Transport Models Under Ambient Conditions available to the NRC in FY 2002. If such transport is found to be important to performance, DOE will provide the technical basis for the parameters/distributions used in FY 2002. DOE will consider obtaining additional information.	Not Received
RT.3.02	UZ2 UZ3	Provide the analysis of geochemical data used for support of the flow field below the repository. DOE will provide the analysis of geochemical data used for support of the fluid flow patterns in the AMR UZ Flow Models and Submodels, available to the NRC in FY 2002.	Not Received
RT.3.03	SZ1	Provide additional information to further justify the uncertainty distribution of flow path lengths in the tuff. This information currently resides in the Uncertainty Distribution for Stochastic Parameters AMR. DOE will provide additional information, to include Nye County data as available, to further justify the uncertainty distribution of flowpath lengths from the tuff at the water table through the alluvium at the compliance boundary in updates to the Uncertainty Distribution for Stochastic Parameters AMR and to the Saturated Zone Flow and Transport Process Model Report, both expected to be available in FY 2002.	Not Received

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RT.3.04	UZ3	Provide sensitivity studies for the relative importance of the hydrogeological units beneath the repository for transport of radionuclides important to performance. DOE will provide a sensitivity study to fully evaluate the relative importance of the different units below the repository that could be used to prioritize data collection, testing, and analysis. This study will be documented in an update to the AMR Radionuclide Transport Models Under Ambient Conditions available to the NRC in FY 2002.	Not Received
RT.3.05	UZ2 UZ3	Provide the documentation for the Alcove 8/Niche 3 testing and predictive modeling for the unsaturated zone. DOE will provide documentation for the Alcove 8 / Niche 3 testing and predictive modeling for the unsaturated zone in updates to the AMRs In Situ Field Testing of Processes and Radionuclide Transport Models Under Ambient Conditions, both available to the NRC in FY 2002.	Not Received
RT.3.06	UZ2 UZ3	The NRC needs DOE to document the pre-test predictions for the Alcove 8/Niche 3 work. DOE responded that pre-test predictions for Alcove 8 Niche 3 work will be provided to NRC via letter report (Brocoum to Greeves) by mid-January 2001.	Received
RT.3.07	SZ2 UZ3	Provide sensitivity studies to test the importance of colloid transport parameters and models to performance for UZ and SZ. Consider techniques to test colloid transport in the Alcove 8/Niche 3 test (for example, microspheres). DOE will perform sensitivity studies as the basis for consideration of the importance of colloid transport parameters and models to performance for the unsaturated and saturated zones and will document the results in updates to appropriate AMRs, and in the TSPA-LA document, all to be available in FY 2003. DOE will evaluate techniques to test colloidal transport in Alcove 8 / Niche 3 and provide a response to the NRC in February 2001.	Partly Received
RT.3.08	SZ2 UZ3	Provide justification that microspheres can be used as analogs for colloids (for example, equivalent ranges in size, charge, etc.). DOE will provide documentation in the C-Wells AMR to provide additional justification that microspheres can be used as analogs for colloids. The C-Wells AMR will be available to the NRC in October 2001.	Not Received
RT.3.09	SZ2	Provide the documentation for the C-wells testing. Use the field test data or provide justification that the data from the laboratory tests is consistent with the data from the field tests. DOE will provide the C-Wells test documentation and will either use the test data or provide a justified reconciliation of the lab and field test data in the C-Wells AMR available in October 2001.	Not Received
RT.3.10	UZ3	Provide analog radionuclide data from the tracer tests for Calico Hills at Busted Butte and from similar analog and radionuclide data (if available) from test blocks from Busted Butte. DOE will provide data from analog tracers used at Busted Butte and data from (AECL) test blocks from Busted Butte in an update to the AMR In Situ Field Testing of Processes in FY 2002.	Not Received
RT.4.01	SZ2 TSPAI UZ3	Provide Revision 1 to the Topical Report. DOE will provide the Disposal Criticality Analysis Methodology Topical Report, Revision 01, to NRC during January 2001.	Received
RT.4.02	TSPAI	Provide the updated FEPs database. DOE stated that it would provide the FEPs AMRs and the FEPs database to NRC during January 2001.	Received
RT.4.03	ENG3 SZ2 UZ3	Provide the applicable list of validation reports and their schedules for external criticality. DOE stated that the geochemical model validation reports for “Geochemistry Model Validation Report: Degradation and Release” and “Geochemistry Model Validation Report: Material Accumulation” are expected to be available during 2001. The remainder of the reports are expected to be available during FY2002 subject to the results of detailed planning and scheduling. DOE understands that these reports are required to be provided prior to LA. A list of model validation reports was provided during the technical exchange and is included as an attachment to the meeting summary.	Partly Received
SDS.1.01	TSPAI	Provide the updated FEPs: Disruptive Events AMR. DOE will provide the updated FEPs AMR to the NRC. Expected availability is January 2001.	Received

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SDS.1.02	ENG2	Consistent with proposed 10 CFR Part 63, the NRC believes the use of the mean is appropriate, however, DOE may use any statistic as long as it is consistent with site data and technically defensible. DOE will either provide technical justification for use of median values or another statistical measure, such as the mean, or will evaluate and implement an alternative approach. The DOE-proposed approach and its basis will be provided to NRC prior to September 2001. The approach will be implemented prior to any potential LA.	Received
SDS.2.01	ENG2	Regarding ground motion, provide documentation, or point the NRC to the documentation on the expert elicitation process, regarding the feedback to the subject matter experts following the elicitation of their respective judgements. DOE will provide documentation demonstrating the adequacy of the elicitation feedback process by December 2000.	Need Additional Information
SDS.2.02	PRE TSPAI	Provide the updated FEPs: Disruptive Events AMR, the Seismic Design Input Report, and the update to the Seismic Topical Report. DOE will provide the updated FEPs AMR to NRC. Expected availability is January 2001. DOE will provide STR 3 to the NRC for their review. Expected availability is January 2002. The Seismic Design Inputs Report is expected to be available to the NRC by September 2001.	Partly Received
SDS.2.03	ENG2	Consistent with proposed 10 CFR Part 63, the NRC believes the use of the mean is appropriate, however, DOE may use any statistic as long as it is consistent with site data and technically defensible. DOE will either provide technical justification for use of median values or another statistical measure, such as the mean, or will evaluate and implement an alternative approach. The DOE-proposed approach and its basis will be provided to NRC prior to September 2001. The approach will be implemented prior to any potential LA.	Received
SDS.2.04	ENG2	The approach to evaluate seismic risk, including the assessment of seismic fragility and evaluation of event sequences is not clear to the NRC, provide additional information. DOE believes the approach contained in the FEPs AMR will be sufficient to support the Site Recommendation. The updated FEPs AMR is expected to be available in January 2001.	Received
SDS.3.01	UZ2 UZ3	The ECRB long-term test and the Alcove 8 Niche 3 test need to be "fractured-informed" (i.e., observation of seepage needs to be related to observed fracture patterns). Provide documentation which discusses this aspect. DOE responded that for the passive test, any observed seepage will be related to full periphery maps and other fracture data in testing documentation. The documentation will be available by any potential LA. For Niche 3, fracture characterization is complete and a 3-D representation will be included in testing documentation. The documentation will be available August 2001.	Partly Received
SDS.3.02	UZ2 UZ3	The NRC needs DOE to document the pre-test predictions for the Alcove 8 Niche 3 work. DOE responded that pre-test predictions for Alcove 8 Niche 3 work will be provided to NRC via letter report (Brocoum to Greeves) by mid-January 2001.	Received
SDS.3.03	ENG3	The NRC needs to review the Fracture Geometry Analysis for the Stratigraphic Units of the Repository Host Horizon AMR. The NRC will provide feedback and proposed agreements to DOE, if needed, by December 2000.	Need Additional Information
SDS.3.04	ENG2 ENG3 PRE UZ2	The NRC needs DOE to document the discussion of excavation-induced fractures. DOE responded that observations of excavation-induced fractures will be documented in a report or AMR revision by June 2001.	Received
TEF.1.01	ENG3 TSPAI	Provide the FEPs AMRs relating to TEF. The DOE will provide the following updated FEPs AMRs related to thermal effects on flow to the NRC: Disruptive Events FEPs (ANL-NBS-MD-000005) Rev 00 ICN 01; Features, Events, and Processes: System Level (ANL-WIS-MD-000019) Rev 00; Features, Events, and Processes in UZ Flow and Transport (ANL-NBS-MD-000001) Rev 01; Features, Events, and Processes in SZ Flow and Transport (ANL-NBS-MD-000002) Rev 01; Features, Events, and Processes in Thermal Hydrology and Coupled Processes (ANL-NBS-MD-000004) Rev 00 ICN 01; Miscellaneous Waste Form FEPs (ANL-WIS-MD-000009) Rev 00 ICN 01; and Engineered Barrier System Features, Events, and Processes (ANL-WIS-PA-000002) Rev 01. Expected availability: January 2001.	Complete

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TEF.1.02	TSPAI	Provide the FEPs database. The DOE will provide the FEPs data base to the NRC during March 2001.	Complete
TEF.2.01	ENG3 UZ2	Consider measuring losses of mass and energy through the bulkhead of the drift-scale test (DST) and provide the technical basis for any decision or method decided upon (include the intended use of the results of the DST such as verifying assumptions in FEP exclusion arguments or providing support for TSPA models. The DOE should analyze uncertainty in the fate of thermally mobilized water in the DST and evaluate the effect this uncertainty has on conclusions drawn from the DST results. The DOE's position is that measuring mass and energy losses through the bulkhead of the DST is not necessary for the intended use of the DST results. The DST results are intended for validation of models of thermally-driven coupled processes in the rock, and measurements are not directly incorporated into TSPA models. Results of the last two years of data support the validation of DST coupled-process models and the current treatment of mass and energy loss through the bulkhead. The DOE will provide the NRC a white paper on the technical basis for the DOE's understanding of heat and mass losses through the bulkhead and their effects on the results by April 2001. This white paper will include the DOE's technical basis for its decision regarding measurements of heat and mass losses through the DST bulkhead. This white paper will address uncertainty in the fate of thermally mobilized water in the DST and also the effect this uncertainty has on conclusions drawn from the DST results. The NRC will provide comments on this white paper. The DOE will provide analyses of the effects of this uncertainty on the uses of the DST in response to NRC comments.	Partly Received
TEF.2.02	ENG3	Provide the location and access to the Multi-Scale Thermohydrologic Model input and output files. The output files are in the Technical Data Management System. The DTNs are LL000509112312.003, LL000509012312.002, and LL000509212312.004. The input files are located in the Project records system. The document identification number is MOL.20000706.0396. The DOE will provide the requested information to the NRC in January 2001.	Complete
TEF.2.03	ENG1	Provide the following references: Multi-Scale Thermohydrologic Model AMR, ICN 01; Abstraction of Near Field Environment Drift Thermodynamic and Percolation Flux AMR, ICN 01; Engineered Barrier System Degradation Flow and Transport PMR, Rev. 01; and Near Field Environment PMR, ICN 03. DOE will provide to the NRC the following documents: Multi-Scale Thermohydrologic Model AMR (ANL-EBS-MD-00049) Rev 00 ICN 01 (January 2001); Abstraction of Near-Field Environment Drift Thermodynamic and Percolation Flux AMR (ANL-EBS-HS-000003) Rev 00 ICN 01 (January 2001); Engineered Barrier System Degradation, Flow and Transport PMR (TDR-EBS-MD-000006) Rev 01 (September 2001); Near-Field Environment PMR (TDR-NBS-MD-000001) Rev 00 ICN 03 (January 2001)	Partly Received
TEF.2.04	ENG1	Provide the Multi-Scale Thermohydrologic Model AMR, Rev. 01. The DOE will provide the Multi-Scale Thermohydrologic Model AMR (ANL-EBS-MD-00049) Rev 01 to the NRC. Expected availability is FY 02.	Not Received
TEF.2.05	ENG3	Represent the cold-trap effect in the appropriate models or provide the technical basis for exclusion of it in the various scale models (mountain, drift, etc.) considering effects on TEF and other abstraction/models (chemistry). See page 11 of the Open Item (OI) 2 presentation. The DOE will represent the "cold-trap" effect in the Multi-Scale Thermohydrologic Model AMR (ANL-EBS-MD-00049) Rev 01, expected to be available in FY 02. This report will provide technical support for inclusion or exclusion of the cold-trap effect in the various scale models. The analysis will consider thermal effects on flow and the in-drift geochemical environment abstraction.	Not Received
TEF.2.06	ENG3 PRE UZ2	Provide the detailed test plan for Phase III of the ventilation test, and consider NRC comments, if any. The DOE will provide a detailed test plan for the Phase III ventilation test in March 2001. The NRC comments will be provided no later than two weeks after receipt of the test plan, and will be considered by the DOE prior to test initiation.	Complete
TEF.2.07	ENG3 PRE UZ2	Provide the Ventilation Model AMR, Rev. 01 and the Pre-Test Predictions for Ventilation Test Calculation, Rev. 00. The DOE will provide the Ventilation Model AMR (ANL-EBS-MD-000030) Rev 01 to the NRC in March 2001. Note that ventilation test data will not be incorporated in the AMR until FY02. The DOE will provide the Pre-test Predictions for Ventilation Tests (CAL-EBS-MD-000013) Rev 00 to the NRC in February 2001. Test results will be provided in an update to the Ventilation Model AMR (ANL-EBS-MD-000030) in FY 02.	Partly Received

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TEF.2.08	ENG3 UZ2	Provide the Mountain Scale Coupled Processes AMR, or an other appropriate AMR, documenting the results of the outlined items on page 20 of the OI 7 presentation (considering the NRC suggestion to compare model results to the O.M. Phillips analytical solution documented in Water Resources Research, 1996). The DOE will provide the updated Mountain-Scale Coupled Processes Model AMR (MDL-NBS-HS-000007) Rev 01 to the NRC in FY 02, documenting the results of the outlined items on page 20 of DOE's Open Item 7 presentation at this meeting. The DOE will consider the NRC suggestion of comparing the numerical model results to the O.M. Phillips analytical solution documented in WRR (1996).	Not Received
TEF.2.09	ENG1	Provide the Multi-Scale Thermohydrologic Model AMR, ICN 03. The DOE will provide the Multi-Scale Thermohydrologic Model AMR (ANL-EBS-MD-00049) Rev 00 ICN 03 to the NRC. Expected availability July 2001.	Not Received
TEF.2.10	ENG3 UZ2	Represent the full variability/uncertainty in the results of the TEF simulations in the abstraction of thermodynamic variables to other models, or provide technical basis that a reduced representation is appropriate (considering risk significance). The DOE will discuss this issue during the TSPAI technical exchange tentatively scheduled for April 2001.	Not Received
TEF.2.11	ENG3 UZ2	Provide the Calibrated Properties AMR, incorporating uncertainty from all significant sources. The DOE will provide an updated Calibrated Properties Model AMR (MDL-NBS-HS-000003) Rev 01 that incorporates uncertainty from significant sources to the NRC in FY 02.	Not Received
TEF.2.12	UZ2 UZ3	Provide the Unsaturated Zone Flow and Transport PMR, Rev. 00, ICN 02, documenting the resolution of issues on page 5 of the OI 8 presentation. The DOE will provide the Unsaturated Zone Flow and Transport PMR (TDR-NBS-HS-000002) Rev 00 ICN 02 to the NRC in February 2001. It should be noted, however, that not all of the items listed on page 5 of the DOE's Open Item 8 presentation at this meeting are included in that revision. The DOE will include all the items listed on page 5 of the DOE's Open Item 8 presentation in Revision 02 of the Unsaturated Zone Flow and Transport PMR, scheduled to be available in FY 02.	Partly Received
TEF.2.13	UZ2 UZ3	Provide the Conceptual and Numerical Models for Unsaturated Zone Flow and Transport AMR, Rev. 01 and the Analysis of Hydrologic Properties Data AMR, Rev. 01. The DOE will provide updates to the Conceptual and Numerical Models for UZ Flow and Transport (MDL-NBS-HS-000005) Rev 01 and the Analysis of Hydrologic Properties Data (ANL-NBS-HS-000002) Rev 01 AMRs to the NRC. Scheduled availability is FY 02.	Not Received
TSPAI.1.01	TSPAI	Provide enhanced descriptive treatment for presenting barrier capabilities in their final approach for demonstrating multiple barriers. Provide discussion of the capabilities of individual barriers, in light of existing parameter uncertainty (e.g., in barrier and system characteristics) and model uncertainty. DOE will provide enhanced descriptive treatment for presenting barrier capabilities in the final approach for demonstrating multiple barriers. DOE will also provide discussion of the capabilities of individual barriers, in light of existing parameter uncertainty (e.g., in barrier and system characteristics) and model uncertainty. The information will be documented in TSPA Methods and Assumptions document, expected to be available to NRC in FY 2002, for any potential license application.	Not Received
TSPAI.1.02	TSPAI	Provide a discussion of the following in documentation of barrier capabilities and the corresponding technical bases: (1) parameter uncertainty, (2) model uncertainty (i.e., the effect of viable alternative conceptual models), (3) spatial and temporal variability in the performance of the barriers, (4) independent and interdependent capabilities of the barriers (e.g., including a differentiation of the capabilities of barriers performing similar functions), and (5) barrier effectiveness with regard to individual radionuclides. Analyze and document barrier capabilities, in light of existing data and analyses of the performance of the repository system. DOE will provide a discussion of the following in documentation of barrier capabilities and the corresponding technical bases: (1) parameter uncertainty, (2) model uncertainty (i.e., the effect of viable alternative conceptual models), (3) spatial and temporal variability in the performance of the barriers, (4) independent and interdependent capabilities of the barriers (e.g., including a differentiation of the capabilities of barriers performing similar functions), and (5) barrier effectiveness with regard to individual radionuclides. DOE will also analyze and document barrier capabilities, in light of existing data and analyses of the performance of the repository system. The information will be documented in TSPA for any potential license application expected to be available in FY 2003.	Not Received

Agreement	Related ISIs or	NRC/DOE Agreement	Status
TSPAI.2.01	DOSE1 DOSE2 DOSE3 ENG1 ENG3 ENG4 SZ1 SZ2 TSPAI UZ1 UZ2 UZ3	Provide clarification of the screening arguments, as summarized in Attachment 2. See Comment # 5, 7, 8, 9, 10, 13, 18, 19 (Part 5), 21, 32, 41, 47, 50, 53, 58, 67, J-5, J-16, and J-18. DOE will clarify the screening arguments, as summarized in Attachment 2, for the highlighted FEPs. The clarifications will be provided in the referenced FEPs AMR and will be provided to the NRC in FY03.	Not Received
TSPAI.2.02	DIRECT1 DIRECT2 DOSE1 DOSE2 DOSE3 ENG1 ENG2 ENG3 ENG4 SZ1 SZ2 TSPAI UZ1 UZ2 UZ3	Provide the technical basis for the screening argument, as summarized in Attachment 2. See Comment # 3, 4, 11, 12, 19 (Parts 1, 2, and 6), 25, 26, 29, 34, 35, 36, 37, 38, 39, 42, 43, 44, 48, 49, 51, 54, 55, 56, 57, 59, 60, 61, 62, 63, 64, 65, 66, 68, 69, 70, 78, 79, J-1, J-2, J-3, J-4, J-7, J-8, J-9, J-10, J-11, J-12, J-13, J-14, J-15, J-17, J-20, J-21, J-22, J-23, J-24, J-25, J-26, and J-27. DOE will provide the technical basis for the screening argument, as summarized in Attachment 2, for the highlighted FEPs. The technical basis will be provided in the referenced FEPs AMR and will be provided to the NRC in FY03.	Not Received

Agreement	Related ISIs or	NRC/DOE Agreement	Status
TSPAI.2.03	DOSE1 DOSE2 DOSE3 SZ1 SZ2 TSPAI UZ3	Add the FEPs highlighted in Attachment 2 to the appropriate FEPs AMRs. See Comment 19 (Part 7 and 8), 20, and J-6. DOE will add the FEPs highlighted in Attachment 2 to the appropriate FEPs AMRs. The FEPs will be added to the appropriate FEPs AMRs and the AMRs will be provided to the NRC in FY03.	Not Received
TSPAI.2.04	DOSE3 ENG1 ENG2 TSPAI	Provide a clarification of the description of the primary FEP. See Comments 24, 31, and 33. DOE will clarify the description of the primary FEPs, as summarized in Attachment 2, for the highlighted FEPs. The clarifications will be provided in the referenced FEPs AMR and will be provided to the NRC in FY03.	Not Received
TSPAI.2.05	TSPAI	It is not clear to the NRC that the current list of FEPs (i.e., the list of FEPs documented in TDR-WIS-MD-000003, 00/01) is sufficiently comprehensive or exhibits the necessary attribute of being auditable (e.g., transparent and traceable). As discussed in the two TSPAI technical exchanges, there are unclear aspects of the approach that DOE plans to use to develop the necessary documentation of those features, events, and processes that they have considered. Accordingly, to provide additional confidence that the DOE will provide NRC with: (1) auditable documentation of what has been considered by the DOE, (2) the technical basis for excluding FEPs, and (3) an indication of the way in which included FEPs have been incorporated in the performance assessment; DOE will provide NRC with a detailed plan (the Enhanced FEP Plan) for comment. In the Enhanced FEP Plan, DOE will address the following items: (1) the approach used to develop a pre-screening set of FEPs (i.e., the documentation of those things that DOE considered and which the DOE would use to provide support for a potential license application), (2) the guidance on the level-of-detail that DOE will use for redefining FEPs during the enhanced FEP process, (3) the form that the pre-screening list of FEPs will take (e.g., list, database, other descriptions), (4) the approach DOE would use for the ongoing evaluation of FEPs (e.g., how to address potentially new FEPs), (5) the approach that DOE would use to evaluate and update the existing scope and description of FEPs, (6) the approach that DOE would use to improve the consistency in the level of detail among FEPs, (7) how the DOE would evaluate the results of its efforts to update the existing scope and definition of FEPs, (8) how the Enhanced FEP process would support assertions that the resulting set of FEPs will be sufficiently comprehensive (e.g., represents a wide range of both beneficial and potential adverse effects on performance) to reflect clearly what DOE has considered, (9) how DOE would indicate their disposition of included FEPs in the performance assessment, (10) the role and definition of the different hierarchical levels used to document the information (e.g., “components of FEPs” and “modeling issues”), (11) how the hierarchical levels used to document the information would be used within DOE’s enhanced FEP process, (12) how the Enhanced FEP Plan would result in documentation that facilitates auditing (i.e., lead to a process that is transparent and traceable), (13) DOE’s plans for using configuration management controls to identify FEP dependencies on ongoing work and design changes. DOE will provide the Enhanced Plan to NRC by March 2002.	Not Received
TSPAI.2.06	TSPAI	Provide justification for the approach to: (1) the level of detail used to define FEPs; (2) the degree of consistency among FEPs; and (3) comprehensiveness of the set of FEPs initially considered (i.e., before screening). DOE proposes to meet with NRC periodically to provide assessments of the DOE’s progress, once it has initiated the Enhanced FEP process, and on changes to the approach documented in the Enhanced FEP Plan. During these progress meetings DOE agrees to provide a justification for their approach to: (1) the level of detail used to define FEPs; (2) the degree of consistency among FEPs; and (3) comprehensiveness of the pre-screening set of FEPs.	Not Received

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TSPAI.2.07	TSPAI	Provide results of the implementation of the Enhanced FEP Plan (e.g., the revised FEP descriptions, screening arguments, the mapping of FEPs to TSPA keywords, and a searchable index of FEP components), in updates to the FEP AMR documents and the FEP Database. DOE agrees to provide the results of their implementation of the Enhanced FEP Plan (e.g., the revised FEP descriptions, screening arguments, improved database navigation through, for example, the mapping of FEPs to TSPA keywords, a searchable index of FEP components, etc.), information requested in updates to the FEP documents and the FEP Database (or other suitable documents) in FY03.	Not Received
TSPAI.3.01	ENG1 TSPAI	Propagate significant sources of uncertainty into projections of waste package and drip shield performance included in future performance assessments. Specific sources of uncertainty that should be propagated (or strong technical basis provided as to why it is insignificant) include: (1) the uncertainty from measured crevice and weight-loss samples general corrosion rates and the statistical differences between the populations, (2) the uncertainty from alternative explanations for the decrease in corrosion rates with time (such as silica coatings that alter the reactive surface area), (3) the uncertainty from utilizing a limited number of samples to define the correction for silica precipitation, (4) the confidence in the upper limit of corrosion rates resulting from the limited sample size, and (5) the uncertainty from alternative statistical representations of the population of empirical general corrosion rates. The technical basis for sources of uncertainty will be established upon completion of existing agreement items CLST 1.4, 1.5, 1.6, and 1.7. DOE will then propagate significant sources of uncertainty into projections of waste package and drip shield performance included in future performance assessments. This technical basis will be documented in a future revision of the General and Localized Corrosion of Waste Package Outer Barrier AMR (ANL-EBS-MD-000003) expected to be available consistent with the scope and schedules for the specified CLST agreements. The results of the AMR analyses will be propagated into future TSPA analyses for any potential license application.	Not Received
TSPAI.3.02	ENG1 TSPAI	Provide the technical basis for resampling the general corrosion rates and the quantification of the impact of resampling of general corrosion rates in revised documentation (ENG1.1.1). DOE will provide the technical basis for resampling the general corrosion rates and the quantification of the impact of resampling of general corrosion rates in an update to the WAPDEG Analysis of Waste Package and Drip Shield Degradation AMR (ANL-EBS-PA-000001). This AMR is expected to be available to NRC in FY 2003.	Not Received
TSPAI.3.03	ENG1 TSPAI	Provide the technical basis for crack arrest and plugging of crack openings (including the impact of oxide wedging and stress redistribution) in assessing the impact of SCC of the drip shield and waste package in revised documentation (ENG1.1.2 and ENG1.4.1). DOE will provide the technical basis for crack arrest and plugging of crack openings (including the impact of oxide wedging and stress redistribution) in assessing the stress corrosion cracking of the drip shield and waste package in an update to the Stress Corrosion Cracking of the Drip Shield, Waste Package Outer Barrier, and the Stainless Steel Structural Material AMR (ANL-EBS-MD-000005) in accordance with the scope and schedule for existing agreement item CLST 1.12.	Not Received
TSPAI.3.04	ENG1 TSPAI	Provide the technical basis that the representation of the variation of general corrosion rates (if a significant portion is “lack of knowledge” uncertainty) does not result in risk dilution of projected dose responses (ENG1.3.3). DOE will provide the technical basis that the representation of the variation of general corrosion rates results in reasonably conservative projected dose rates. The technical basis will be documented in an update to the WAPDEG Analysis of Waste Package and Drip Shield Degradation AMR (ANL-EBS-PA-000001). This AMR is expected to be available to NRC in FY 2003. These results will be incorporated into future TSPA documentation for any potential license application.	Not Received
TSPAI.3.05	ENG1 TSPAI	Provide the technical basis for the representation of uncertainty/variability in the general corrosion rates in revised documentation. This technical basis should provide a detailed discussion and analyses to allow independent reviewers the ability to interpret the representations of 100% uncertainty, 100% variability, and any intermediate representations in the DOE model (ENG1.3.6). DOE will provide the technical basis for the representation of uncertainty/variability in the general corrosion rates. This technical basis will include the results of 100% uncertainty, 100% variability, and selected intermediate representations used in the DOE model. These results will be documented in an update to the WAPDEG Analysis of Waste Package and Drip Shield Degradation AMR (ANL-EBS-PA-000001) or other document. This AMR is expected to be available to NRC in FY 2003.	Not Received

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TSPAI.3.06	ENG2 TSPAI	Provide the technical basis for the methodology used to implement the effects of seismic effects on cladding in revised documentation. DOE will demonstrate that the methodology used to represent the seismic effects of cladding does not result in an underestimation of risk in the regulatory timeframe (ENG2.1.1). DOE will provide the technical basis for the methodology used to implement the effects of seismic effects on cladding in revised documentation. DOE will demonstrate that the methodology used to represent the seismic effects of cladding does not result in an underestimation of risk in the regulatory timeframe in TSPA-LA. The documentation is expected to be available to NRC in FY 2003.	Not Received
TSPAI.3.07	ENG3 TSPAI UZ2	Provide technical basis for representation of or the neglect of dripping from rockbolts in the ECRB in performance assessment, including the impacts on hydrology, chemistry, and other impacted models. Appropriate consideration will be given to the uncertainties in the source of the moisture, and how those uncertainties impact other models (ENG3.1.1). DOE will provide technical basis for determination of future sources of water in the ECRB, will evaluate the possibility of preferential dripping from engineered materials, and will give appropriate consideration to the uncertainties of the water sources, as well as their potential impact on other models. The work done to date as well as the additional work will be documented in the AMR on In-Situ Field Testing Processes (ANL-NBS-HS-000005) or other documents. This AMR will be available to NRC in FY 2003. DOE will evaluate the role of condensation as a source of water and any impacts of this on hydrologic and chemical conitions in the drift, and DOE will document this work. The effects of condensation will be included in TSPA if found to be potentially important to performance.	Not Received
TSPAI.3.08	ENG3 TSPAI	Provide the technical basis (quantification) for the abstraction of in-package chemistry and it's implementation into the TSPA which will demonstrate that the utilization of the weighted-moving-average methodology will not result in an underestimation of risk (ENG3.1.3). DOE will provide the technical basis (quantification) for the abstraction of in-package chemistry and its implementation into the TSPA, which will demonstrate that the implementation methodology will not result in an underestimation of risk. The technical basis will be documented in TSPA-LA and is expected to be available in FY 2003.	Not Received
TSPAI.3.09	ENG3 TSPAI	Provide the documentation that presents the representation of uncertainty and variability in the near-field environment abstractions in the TSPA (ENG3.1.4). DOE will present the representation of uncertainty and variability in water and gas chemistry entering the drift in the near-field environment abstractions for the TSPA. This will be documented in the Abstraction of Drift-Scale Coupled Processes (ANL-NBS-HS-000029) or other document expected to be available in FY 2003.	Not Received
TSPAI.3.10	ENG3 TSPAI	Provide the documentation of the integrated analyses and comprehensive uncertainty analyses related to the Physical and Chemical Environmental Abstraction Model (ENG3.1.5). DOE will provide the documentation of the integrated analyses and comprehensive uncertainty analyses related to the EBS physical and chemical environment in documentation associated with TSPA for any potential license application. The documentation is expected to be available to NRC in FY 2003.	Not Received
TSPAI.3.11	ENG3 TSPAI UZ2	DOE should account for appropriate integration between the 3D UZ flow model, the MSTH model, and the drift seepage model. In particular, DOE should ensure that relevant spatial distributions are propagated appropriately between the UZ flow model, the thermohydrology model, and the seepage model (ENG3.1.6). DOE will compare the infiltration flux used for the infiltration bins with the 3D Unsaturated Zone flow model and the multi-scale thermohydrologic (MSTH) model results. The technical basis for any approximations in the spatial distribution of flow rates involved in this abstraction will be provided in Abstraction of NFE Drift Thermodynamic Environment and Percolation Flow AMR (ANL-EBS-HS-000003) or other suitable document. In particular, DOE will ensure that the MSTH model output to the seepage abstraction (or any other model that may provide percolation flux to the seepage abstraction) does not lead to underestimation of seepage. This AMR is expected to be available to NRC in FY 2003.	Not Received
TSPAI.3.12	ENG3 TSPAI	DOE should complete testing of corrosion in the chemical environments predicted by the model or provide technical basis why it is not needed (ENG3.1.8). DOE will conduct testing of corrosion in the credible range of chemical environments predicted by the model in accordance with the scope and schedule for existing agreements CLST 1.4 and 1.6 or provide a technical basis why it is not needed.	Not Received
TSPAI.3.13	ENG3 TSPAI	Provide a comparison of the environments for corrosion predicted in the models, to the testing environments used to define empirical corrosion rates in revised documentation (ENG3.2.1). DOE will provide a comparison of the environments for corrosion predicted in the models, to the testing environments utilized to define empirical corrosion rates in revised documentation consistent with the scope and schedule for existing agreement item CLST 1.1.	Not Received

Agreement	Related ISIs or	NRC/DOE Agreement	Status
TSPAI.3.14	ENG4 TSPAI	DOE should account for the full range of environmental conditions for the in-package chemistry model (ENG4.1.1). DOE will update the in-package chemistry model to account for scenarios and their associated uncertainties required by TSPA. This will be documented in the In-Package Chemistry AMR (ANL-EBS-MD-000056) expected to be available to NRC in FY 2003.	Not Received
TSPAI.3.15	ENG4 TSPAI	Define a reference EQ3/6 database for the Yucca Mountain Project. DOE will provide documentation of all deviations from the reference database and justification for those deviations used by different geochemical modeling activities (ENG4.1.2). DOE will define a reference EQ3/6 database for the Yucca Mountain Project. DOE will provide documentation of all the deviations from the reference database and justification for those deviations used by different geochemical modeling activities. The database will be available in FY 2003.	Not Received
TSPAI.3.16	ENG4 TSPAI	DOE should include the possibility of localized flow pathways in the engineered barrier system in TSPA calculations, including the influence of introduced materials on water and gas chemistry on these preferential flow pathways (ENG4.1.6). DOE will evaluate the effect of localized flow pathways on water and gas chemistry in the engineered barrier system as input to TSPA calculations, including the influence of introduced materials on these preferential flow pathways consistent with existing agreements ENFE 2.4, 2.5, and 2.6. This will be documented in an update to the Physical and Chemical Environment Model AMR (ANL-EBS-MD-000033) or other suitable document. This AMR is expected to be available to NRC in FY 2003.	Not Received -
TSPAI.3.17	ENG4 TSPAI	Provide an uncertainty analysis of the diffusion coefficient governing transport of dissolved and colloidal radionuclides through the invert. The analysis should include uncertainty in the modeled invert saturation (ENG4.4.1). DOE will provide an uncertainty analysis of the diffusion coefficient governing transport of dissolved and colloidal radionuclides through the invert. The analysis will include uncertainty in the modeled invert saturation. The uncertainty analysis will be documented in the EBS Radionuclide Transport Abstraction AMR (ANL-WIS-PA-000001) expected to be available to NRC in FY 2003.	Not Received
TSPAI.3.18	TSPAI UZ1	Provide a technical basis that the water-balance plug-flow model adequately represents the non-linear flow processes represented by Richard's equation, particularly over the repository where there is thin soil (UZ1.2.1). DOE will provide a technical basis that the water-balance plug-flow model adequately represents the non-linear flow processes represented by Richard's equation, particularly over the repository where there is thin soil. The technical basis will be documented in an update to the Simulation of Net Infiltration for Modern and Potential Future Climates AMR (ANL-NBS-HS-000032). The AMR is expected to be available to NRC in FY 2003.	Not Received
TSPAI.3.19	TSPAI UZ1	DOE will provide justification for the use of its evapotranspiration model, and defend the use of the analog site temperature data (UZ1.3.1). DOE will provide justification for the use of the evapotranspiration model, and justify the use of the analog site temperature data. The justification will be documented in an update to the Simulation of Net Infiltration for Modern and Potential Future Climates AMR (ANL-NBS-HS-000032) and the Future Climate Analysis AMR (ANL-NBS-GS-000008). The AMRs are expected to be available to NRC in FY 2003.	Not Received
TSPAI.3.20	TSPAI UZ1	Provide access to data supporting the synthetic meteorologic records (4JA.s01 and Area12.s01) (UZ1.3.2). DOE will provide data supporting the synthetic meteorologic records (specifically, data files 4JA.s01 and Area12.s01). These data files will be provided to NRC September 2001.	Received
TSPAI.3.21	TSPAI UZ1	Demonstrate that effects of near surface lateral flow on the spatial variability of net infiltration are appropriately considered (UZ1.5.1). DOE will demonstrate that effects of near surface lateral flow on the spatial variability of net infiltration are appropriately considered in an update to the Simulation of Net Infiltration for Modern and Potential Future Climates AMR (ANL-NBS-HS-000032) and UZ Flow Models and Submodels AMR (MDL-NBS-HS-000006). These AMRs are expected to be available to NRC in FY 2003.	Not Received
TSPAI.3.22	TSPAI UZ2	Provide an assessment or discussion of the uncertainty involved with using a hydrologic property set obtained by calibrating a model on current climate conditions and using that model to forecast flow for future climate conditions (UZ2.3.1). DOE will provide an assessment or discussion of the uncertainty involved with using a hydrologic property set obtained by calibrating a model on current climate conditions and using that model to forecast flow for future climate conditions. This assessment will be documented in the UZ Flow Models and Submodels AMR (MDL-NBS-HS-000006) expected to be available to NRC in FY 2003.	Not Received

Agreement	Related ISIs or	NRC/DOE Agreement	Status
TSPAI.3.23	TSPAI UZ2	DOE should evaluate spatial heterogeneity of hydrologic properties within hydrostratigraphic units and the effect this heterogeneity has on model results of unsaturated flow, seepage into the drifts and transport. DOE should also provide a technical basis for the assessment that bomb-pulse Cl-36 data found below the Paint Brush tuff can be linked to a negligible amount of fast flowing water (UZ2.3.2). DOE will evaluate spatial heterogeneity of hydrologic properties within hydrostratigraphic units and the effect this heterogeneity has on model results of unsaturated flow, seepage into the drifts and transport. This evaluation will be documented in the UZ Flow Models and Submodels AMR (MDL-NBS-HS-000006), Radionuclide Transport Models under Ambient Conditions (MDL-NBS-HS-000008) and Seepage Models for PA Including Drift Collapse AMR (MDL-NBS-HS-000002) expected to be available to NRC in FY 2003. DOE will also provide a technical basis for the assessment that bomb-pulse Cl36 data found below the PTn can be linked to a negligible amount of fast flowing water. The technical basis will be documented in the UZ Flow Models and Submodels AMR (MDL-NBS-HS-000006) expected to be available to NRC in FY 2003.	Not Received
TSPAI.3.24	TSPAI UZ2	Provide the analysis of geochemical and hydrological data (water content, water potential, and temperature) used for support of the flow field below the repository, particularly in the Calico Hills, Prow Pass, and Bullfrog hydrostratigraphic layers. Demonstrate that potential bypassing of matrix flow pathways below the area of the proposed repository, as opposed to the entire site-scale model area, is adequately incorporated for performance assessment, or provide supporting analyses that the uncertainties are adequately included in the TSPA (UZ2.3.3). DOE will provide an analysis of available geochemical and hydrological data (water content, water potential, and temperature) used for support of the flow field below the repository, particularly in the Calico Hills, Prow Pass, and Bullfrog hydrostratigraphic layers. The analyses will demonstrate that potential bypassing of matrix flow pathways below the area of the proposed repository, as opposed to the entire site-scale model area, is adequately incorporated for performance assessment, or provide supporting analyses that the uncertainties are adequately included in the TSPA. These analyses will be documented in the UZ Flow Models and Submodels AMR (MDL-NBS-HS-000006), In-Situ Field Testing of Processes AMR (ANL-NBS-HS-000005), and Calibrated Properties Model AMR (MDL-NBS-HS-000003) expected to be available to NRC in FY 2003.	Not Received
TSPAI.3.25	TSPAI UZ2	DOE should use the Passive Cross Drift Hydrologic test, the Alcove 8 - Niche 3 tests, the Niche 5 test, and other test data to either provide additional confidence in or a basis for revising the TSPA seepage abstraction and associated parameter values (e.g., flow focusing factor, van Genuchten alpha for fracture continuum, etc.), or a provide technical basis for not using it (UZ2.3.4). DOE will utilize field test data (e.g., the Passive Cross Drift Hydrologic test, the Alcove 8 - Niche 3 tests, the Niche 5 test, and other test data) to either provide additional confidence in or a basis for revising the TSPA seepage abstraction and associated parameter values (e.g., flow focusing factor, van Genuchten alpha for fracture continuum, etc.), or provide technical basis for not using it. This will be documented in Seepage Calibration Model and Seepage Testing Data AMR (MDL-NBS-HS-000004) expected to be available to NRC in FY 2003.	Not Received
TSPAI.3.26	TSPAI UZ2	Calibrate the UZ flow model using the most recent data on saturations and water potentials, and clearly document the sources of calibration data and data collection methods (UZ2.3.5). DOE will calibrate the UZ flow model using the most recent data on saturations and water potentials, and document the sources of calibration data and data collection methods. The results will be documented in the Calibrated Properties Model AMR (MDL-NBS-HS-000003) expected to be available to NRC in FY 2003.	Not Received
TSPAI.3.27	TSPAI UZ2	Provide an overview of water flow rates used in the UZ model above and below the repository, in the MSTHM, in the seepage abstraction, and in the in-drift flow path models, to ensure appropriate integration between the various models (UZ2.TT.3). DOE will provide an overview of water flow rates used in the UZ model above and below the repository, in the Multi-Scale Thermohydrologic Model (MSTHM), in the seepage abstraction, and in the in drift flow path models, to ensure appropriate integration between the various models. This will be documented in the TSPA for any potential license application expected to be available to NRC in FY 2003.	Not Received
TSPAI.3.28	TSPAI UZ3	DOE needs to provide independent lines of evidence to provide additional confidence in the use of the active-fracture continuum concept in the transport model (UZ3.5.1). DOE will provide independent lines of evidence to provide additional confidence in the use of the active fracture continuum concept in the transport model. This will be documented in Radionuclide Transport Models under Ambient Conditions AMR (MDL-NBS-HS-000008) and UZ Flow Models and Submodels AMR (MDL-NBS-HS-000006) expected to be available to NRC in FY 2003.	Not Received

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TSPAI.3.29	TSPAI UZ3	Provide verification that the integration of the active fracture model with matrix diffusion in the transport model is properly implemented in the TSPA abstraction (UZ3.TT.3). DOE will provide verification that the integration of the active fracture model with matrix diffusion in the transport model is properly implemented in the TSPA abstraction. This verification will be documented in the Particle Tracking Model and Abstraction of Transport Processes (ANL-NBS-HS-000026) expected to be available to NRC in FY 2003.	Not Received
TSPAI.3.30	SZ2 TSPAI	Provide the technical basis for the contrasting concentrations of colloids available for reversible attachment in the engineered barrier system and the saturated zone. Sensitivity analyses planned in response to RT Agreement 3.07 should address the effect of colloid concentration on Kc. Update, as necessary, the Kc parameter as new data become available from the Yucca Mountain region (SZ2.3.1). DOE will provide the technical basis for the contrasting concentrations of colloids available for reversible attachment in the engineered barrier system and the saturated zone. The sensitivity analyses planned in response to RT Agreement 3.07 will address the effect of colloid concentration on the Kc parameter. The technical basis will be documented in the Waste Form Colloid Associated Concentration Limits: Abstractions and Summary (ANL-WIS-MD-000012) in FY 2003. The Kc parameter will be updated as new data become available from the Yucca Mountain region in the Uncertainty Distribution for Stochastic Parameters AMR (ANL-NBS-MD-000011) in FY2003.	Not Received
TSPAI.3.31	SZ2 TSPAI	Evaluate the effects of temporal changes in saturated zone chemistry on radionuclide concentrations (SZ2.3.2). DOE will reexamine the FEPs, currently included in the performance assessment, that may lead to temporal changes in saturated zone hydrochemistry. If the DOE determines that these FEPs can be excluded, the results will be documented in the FEP Saturated Zone Flow and Transport AMR (ANL-NBS-MD-000002) in FY 2003. If the DOE determines that these FEPs cannot be excluded from the performance assessment, the DOE will evaluate the effects of temporal changes in the saturated zone chemistry on radionuclide concentrations and will document this evaluation in above mentioned AMR.	Not Received
TSPAI.3.32	SZ2 TSPAI	Provide the technical basis that the representation of uncertainty in the saturated zone as essentially all lack-of-knowledge uncertainty (as opposed to real sample variability) does not result in an underestimation of risk when propagated to the performance assessment (SZ2.4.1). DOE will provide the technical basis that the representation of uncertainty (i.e., lack-of-knowledge uncertainty) in the saturated zone does not result in an underestimation of risk when propagated to the performance assessment. A deterministic case from Saturated Zone Flow Patterns and Analyses AMR (ANL-NBS-HS-000038) will be compared to TSPA analyses. The comparison will be documented in the TSPA for any potential license application expected to be available to NRC in FY 2003.	Not Received
TSPAI.3.33	DOSE2 TSPAI	Provide justification that the Kd values used for radionuclides in the soil in Amargosa valley based on the results of a literature review are realistic or conservative for actual conditions at the receptor location (DOSE2.2.1). DOE will provide justification that the Kd values used for radionuclides in the soil in Amargosa Valley are realistic or conservative for actual conditions at the receptor location. The justification will be provided in Evaluate Soil/Radionuclide Removal by Erosion and Leaching AMR (ANL-NBS-MD-000009) or other document expected to be available to NRC in FY 2003.	Not Received
TSPAI.3.34	DOSE3 TSPAI	For the Radionuclides that dominate the TSPA dose, provide the technical basis for selection of Radionuclide or element specific biosphere parameters that are important in the BDCF calculations (e.g. soil to plant transfer factors) (DOSE3.2.1). For the radionuclides that dominate the TSPA dose, DOE will provide the technical basis for selection of radionuclide or element specific biosphere parameters (except for Kds which are addressed in TSPAI 3.33) that are important in the BDCF calculations (e.g. soil to plant transfer factors). The technical basis will be documented in the Transfer Coefficient Analysis AMR (ANL-MGR-MD-000008) or other document and is expected to be available to NRC in FY 2003.	Not Received
TSPAI.3.35	DOSE3 TSPAI	Provide additional justification to support that the assumed crop interception fraction is appropriate for all radionuclides considered and does not result in underestimations of dose. Discussions should address the impacts of electrostatic charge and particle size on the interception fraction for all radionuclides considered in the TSPA (DOSE3.2.5). DOE will provide additional justification to support that the assumed crop interception fraction is appropriate for all radionuclides that dominate the TSPA dose and does not result in underestimations of dose. The justification will include the impacts of electrostatic charge and particle size on the interception fraction. This justification will be documented in Identification of Ingestion Exposure Parameters (ANL-MGR-MD-000006) or other document expected to be available to NRC in FY 2003.	Not Received

Agreement	Related ISIs or	NRC/DOE Agreement	Status
TSPAI.3.36	DOSE3 TSPAI	Document the methodology that will be used to incorporate the uncertainty in soil leaching factors into the TSPA analysis, if that uncertainty is found to be important to the results of the performance assessment (DOSE3.3.1). DOE will document the methodology used to incorporate the uncertainty in soil leaching factors into the TSPA analysis. This will be documented in Nominal Performance Biosphere Dose Conversion Factor Analysis AMR (ANL-MGR-MD-000009), Disruptive Event Biosphere Dose Conversion Factor Analysis (ANL-MGR-MD-000003) or other document expected to be available to NRC in FY 2003.	Not Received
TSPAI.3.37	DOSE3 TSPAI	Provide a quantitative analysis that the sampling method including the correlations to NP used by the TSPA code to abstract the GENII-S process model code adequately represent the uncerrtainty and variability and correlations for the biosphere process model (DOSE3.4.1). DOE will provide a quantitative analysis that the sampling method including the correlations between BDCFs utilized by the TSPA code to abstract the GENII-S process model data adequately represent the uncertainty and variability and correlations for the biosphere process model. This will be documented in Nominal Performance Biosphere Dose Conversion Factor Analysis AMR (ANL-MGR-MD-000009), Disruptive Event Biosphere Dose Conversion Factor Analysis (ANL-MGR-MD-000003) or other document expected to be available to NRC in FY 2003. Results of these analyses will be documented in the TSPA for any potential license application expected to be available to NRC in FY 2003.	Not Received
TSPAI.3.38	TSPAI	DOE will develop guidance in the model abstraction process that can be adhered to by all model developers so that (1) the abstraction process, (2) the selection of conservatism in components, and (3) representation of uncertainty are systematic across the TSPA model. DOE will evaluate and define approaches to deal with: (1) evaluating non-linear models as to what their most conservative settings may be if conservatism is being used to address uncertainty, and (2) trying to utilize human intuition in a complex system. In addition, DOE will consider adding these items to the internal/external reviewer’s checklists to ensure proper implementation of the improved methodology (TSPA0002). DOE will develop written guidance in the model abstraction process for model developers so that (1) the abstraction process, (2) the selection of conservatism in components, and (3) representation of uncertainty, are systematic across the TSPA model. These guidelines will address: (1) evaluation of non-linear models when conservatism is being utilized to address uncertainty, and (2) utilization of decisions based on technical judgement in a complex system. These guidelines will be developed, implemented, and be made available to the NRC in FY 2002.	Not Received
TSPAI.3.39	TSPAI	In future performance assessments, DOE should document the simplifications used for abstractions per TSPAI.3.38 activities. Justification will be provided to show that the simplifications appropriately represent the necessary processes and appropriately propagate process model uncertainties. Comparisons of output from process models to performance assessment abstractions will be provided, with the level of detail in the comparisons commensurate with any reduction in propagated uncertainty and the risk significance of the model (TSPA0003). DOE will document the simplifications utilized for abstractions per TSPAI.3.38 activities for all future performance assessments. Justification will be provided to show that the simplifications appropriately represent the necessary processes and appropriately propagate process model uncertainties. Comparisons of output from process models to performance assessment abstractions will be provided, with the level of detail in the comparisons commensurate with any reduction in propagated uncertainty and the risk significance of the model. The documentation of the information will be provided in abstraction AMRs in FY 2003.	Not Received
TSPAI.3.40	TSPAI	DOE will implement effective controls to ensure that the abstractions defined in the AMR’s are consistently propagated into the TSPA, or ensure that the TSPA documentation describes any differences. Specific examples of needed revisions (if still applicable) include: (1) the implementation of flux splitting in the TSPA model, (2) the propagation of thermohydrology uncertainty/variability into the WAPDEG corrosion model calculations, and (3) the implementation of the in-package chemistry abstraction. DOE will implement program improvements to ensure that the abstractions defined in the AMRs are consistently propagated into the TSPA, or ensure that the TSPA documentation describes any differences. Program improvements may include, for example, upgrades to work plans, procedural upgrades, preparation of desktop guides, worker training, increased review and oversight. The program improvements will be implemented and be made available to the NRC during FY 2002.	Not Received
TSPAI.3.41	TSPAI	To provide support for the mathematical representation of data uncertainty in the TSPA, the DOE will provide technical basis for the data distributions used in the TSPA. An example of how this may be accomplished is the representation on a figure or chart of the data plotted as an empirical distribution and the probability distribution assigned to fit these data. DOE will provide the technical basis for the data distributions utilized in the TSPA to provide support for the mathematical representation of data uncertainty in the TSPA. The documentation of the technical basis will be incorporated in documentation associated with TSPA for any potential license application. The documentation is expected to be available to NRC in FY 2003.	Not Received

Agreement	Related ISIs or	NRC/DOE Agreement	Status
TSPAI.3.42	ENG4 TSPAI	DOE should provide a sensitivity analysis on the potentially abrupt changes in colloid concentrations due to shifts in modeled pH and ionic strength across uncertain stability boundaries. This analysis may be combined with plans to address ENFE Agreement 4.06 and RT Agreement 3.07. DOE will complete sensitivity analyses to investigate the effects of varying colloid concentration due to shifts in model predicted pH and ionic strength across uncertain stability boundaries. These analyses will be documented in TSPA for any potential license application expected to be available to NRC in FY 2003.	Not Received
TSPAI.4.01	TSPAI	DOE will document the methodology that will be used to incorporate alternative conceptual models into the performance assessment. The methodology will ensure that the representation of alternative conceptual models in the TSPA does not result in an underestimation of risk. DOE will document the guidance given to process-level experts for the treatment of alternative models. The implementation of the methodology will be sufficient to allow a clear understanding of the potential effect of alternative conceptual models and their associated uncertainties on the performance assessment. The methodology will be documented in the TSPA-LA methods and assumptions document in FY02. The results will be documented in the appropriate AMRs or the TSPA for any potential license application in FY 2003.	Not Received
TSPAI.4.02	TSPAI	DOE will provide the documentation that supports the representation of distribution coefficients (Kd's) in the performance assessment as uncorrelated is consistent with the physical processes and does not result in an underestimation of risk. This will be documented in the TSPA for any potential license application in FY03.	Not Received
TSPAI.4.03	TSPAI	DOE will document the method that will be used to demonstrate that the overall results of the TSPA are stable. DOE will provide documentation that submodels (including submodels used to develop input parameters and transfer functions) are also numerically stable. DOE will address in the method the stability of the results with respect to the number of realizations. DOE will describe in the method the statistical measures that will be used to support the argument of stability. The method will be documented in TSPA LA Methods and Assumptions Document in FY02. The results of the analyses will be provided in the TSPA (or other appropriate documentation) for any potential license application in FY 2003.	Not Received
TSPAI.4.04	TSPAI	DOE will conduct appropriate analyses and provide documentation that demonstrates the results of the performance assessment are stable with respect to discretization (e.g. spatial and temporal) of the TSPA model. This will be documented in the TSPA for any potential license application in FY 2003.	Not Received
TSPAI.4.05	TSPAI	DOE will document the process used to develop confidence in the TSPA models (e.g., steps similar to those described in NUREG-1636). The detailed process is currently documented in the model development procedures that are being evaluated for process improvement in response to the model validation corrective action report CAR-BSC-01-C-001. The upgraded model validation procedures will be available for NRC review in FY 2002.	Not Received
TSPAI.4.06	TSPAI	DOE will document the implementation of the process for model confidence building and demonstrate compliance with model confidence criteria in accordance with the applicable procedures. This will be documented in the respective AMR revisions and made available to NRC in FY 2003.	Not Received
TSPAI.4.07	TSPAI	DOE's software qualification requirements are currently documented in procedure AP SI.1Q which is under review for process improvement as part of software CAR-BSC-01-C-002. During its review of AP SI.1Q, DOE will consider: 1) the procedure it would follow to conduct a systematic and uniform verification — all areas of a code analyzed at a consistent level, 2) the process it would follow to ensure correct implementation of algorithms, and 3) the process it would follow for the full disclosure of calculations and results. DOE will document compliance with the improved process in the verification documentation required by AP SI.1Q. Software qualification record packages for the affected programs will be available for NRC review in FY 2003.	Not Received
USFIC.3.01	UZ1	Provide the documentation sources and schedule for the Monte Carlo method for analyzing infiltration. DOE will provide the schedule and identify documents expected to contain the results of the Monte Carlo analyses in February 2002.	Not Received
USFIC.3.02	UZ1	Provide justification for the parameters in Table 4-1 of the Analysis of Infiltration Uncertainty AMR (for example, bedrock permeability in the infiltration model needs to be reconciled with the Alcove 1 results/observations. Also, provide documentation (source, locations, tests, test results) for the Alcove 1 and Pagany Wash tests. DOE will provide justification and documentation in a Monte Carlo analyses document. The information will be available in February 2002.	Not Received

Agreement	Related ISIs or	NRC/DOE Agreement	Status
USFIC.4.01	UZ2 UZ3	The ongoing and planned testing are a reasonable approach for a licensing application with the following comments: (i) consider a mass balance of water for alcove 8/Niche 3 cross over test; (ii) monitor evaporation during all testing; (iii) provide the documentation of the test plan for the Passive Cross Drift Hydrologic test; (iv) provide the NRC with any Cross Drift seepage predictions that may have been made for the Passive Cross Drift Hydrologic test; (v) provide documentation of the results obtained and the analysis for the Passive Cross Drift Hydrologic test. This documentation should include the analysis of water samples collected during entries into the Cross Drift (determination whether the water comes from seepage or condensation); (vi) provide documentation of the results obtained and the analysis for the Alcove 7 test. This documentation should include the analysis of water samples collected during entries into Alcove 7 (determination whether the water comes from seepage or condensation); (vii) provide the documentation of the test plan for the Niche 5 test; (viii) provide documentation of the results obtained and the analysis for the Niche 5 test; (ix) provide documentation of the results obtained and the analysis for the Systematic Hydrologic Characterization test; (x) provide documentation of the results obtained and the analysis for the Niche 4 test; and (xi) provide documentation of the results obtained from the calcite filling test. Include interpretation of the observed calcite deposits found mostly at the bottom of the lithophysal cavities. DOE stated that: (1) a mass balance of water for the Alcove 8/Niche 3 test has been considered, but is not feasible due to the size of the collection system that would be required. A collection system to obtain a mass balance is being developed for the Niche 5 test (i); (2) evaporation will be monitored for all tests where evaporation is a relevant process (ii); (3) test plans for Niche 5 and the Cross Drift Hydrologic tests are expected to be available to NRC FY 2002 (iii, vii); (4) the Cross Drift seepage predictions will be documented in the Seepage Calibration Model and Seepage Testing Data AMR (MDL-NBS-HS-000004) expected to be available to NRC by FY 2003 (iv); (5) DOE will document the results for the tests identified above (except calcite filling observations) in the In-Situ Field Testing of Processes AMR (ANL-NBS-HS-000005) expected to be available to NRC in FY 2003 (v), (vi), (viii),(ix),(x); (6) results of the calcite filling observations will be documented in Analysis of Geochemical Data for the Unsaturated Zone (ANL-NBS-HS-000017) and the UZ Flow Models and Submodels (MDL-NBS-HS-000006) expected to be available to NRC FY 2003 (xi).	Not Received
USFIC.4.02	UZ2	Include the effect of the low-flow regime processes (e.g., film flow) in DOE's seepage fraction and seepage flow, or justify that it is not needed. DOE will include the effect of the low-flow regime processes (e.g., film flow) in the seepage fraction and seepage flow, or justify that it is not needed. These studies will be documented in Seepage Models for PA Including Drift Collapse AMR (MDL-NBS-HS-000002) expected to be available to NRC in FY 2003.	Not Received
USFIC.4.03	UZ2	When conducting seepage studies, consider smaller scale tunnel irregularities in drift collapse or justify that it is not needed. When conducting seepage studies, DOE will consider smaller scale tunnel irregularities in drift collapse or justify that it is not needed. These studies will be documented in Seepage Models for PA Including Drift Collapse AMR (MDL-NBS-HS-000002) expected to be available to NRC in FY 2003.	Not Received
USFIC.4.04	UZ2	Provide final documentation for the effectiveness of the PTn to dampen episodic flow, including reconciling the differences in chloride-36 studies. DOE will provide final documentation for the effectiveness of the PTn to dampen episodic flow, including reconciling the differences in chlorine-36 studies These studies will be documented in UZ Flow Models and Submodels AMR (MDL-NBS-HS-000006) expected to be available to NRC in FY 2003.	Not Received
USFIC.4.05	UZ2	Provide the analysis of geochemical data used for support of the flow field below the repository.	Complete
USFIC.4.06	UZ2	Provide documentation of the results obtained from the Comparison of Continuum and Discrete Fracture Network Models modeling study. Alternatively, provide justification of the continuum approach at the scale of the seepage model grid (formerly June 20 letter, item xiii). DOE will provide documentation of the results obtained from the Comparison of Continuum and Discrete Fracture Network Models modeling study or provide justification of the continuum approach at the scale of the seepage model grid. This will be documented in Seepage Calibration Model and Seepage Testing Data AMR (MDL-NBS-HS-000004) or other suitable document expected to be available to NRC in FY 2003.	Not Received

Agreement	Related ISIs or	NRC/DOE Agreement	Status
USFIC.4.07	UZ2	Provide documentation of the results obtained from the Natural Analogs modeling study. The study was to apply conceptual models and numerical approaches developed from Yucca Mountain to natural analog sites with observations of seepage into drifts, drift stability, radionuclide transport, geothermal effects, and preservation of artifacts. DOE will provide documentation of the results obtained from the Natural Analogs modeling study. The study was to apply conceptual models and numerical approaches developed from Yucca Mountain to natural analog sites with observations of seepage into drifts, drift stability, radionuclide transport, geothermal effects, and preservation of artifacts. This will be documented in the Natural Analogs for the Unsaturated Zone AMR (ANL-NBS-HS-000007) expected to be available to NRC FY 2002.	Not Received
USFIC.5.01	SZ1	The NRC believes that the incorporation of horizontal anisotropy in the site scale model should be reevaluated to ensure that a reasonable range for uncertainty is captured. The data from the C-wells testing should provide a technical basis for an improved range. As part of the C-wells report, DOE should include an analysis of horizontal anisotropy for wells that responded to the long-term tests. Results should be included for the tuffs in the calibrated site scale model. DOE will provide the results of the requested analyses in C-wells report(s) in October 2001, and will carry the results forward to the site-scale model, as appropriate.	Not Received
USFIC.5.02	SZ1	Provide the update to the SZ PMR, considering the updated regional flow model. A revision to the Saturated Zone Flow and Transport PMR is expected to be available and will reflect the updated United States Geological Survey (USGS) Regional Groundwater Flow Model in FY 2002, subject to receipt of the model report from the USGS (reference item 9).	Not Received
USFIC.5.03	SZ1 SZ2	DOE's outline for collecting data in the alluvium appears reasonable but lacks detail. Provide a detailed testing plan for alluvial testing to reduce uncertainty (for example, the plan should give details about hydraulic and tracer tests at the well 19 complex and it should also identify locations for alluvium complex testing wells and tests and logging to be performed). NRC will review the plan and provide comments, if any, for DOE's consideration. In support and preparation for this meeting, DOE provided work plans for the Alluvium Testing Complex and the Nye County Drilling Program (FWP-SBD-99-002, Alluvial Tracer Testing Field Work Package, and FWP-SBD-99-001, Nye County Early Warning Drilling Program, Phase II and Alluvial Testing Complex Drilling). DOE will provide test plans of the style of the Alcove 8 plan as they become available. In addition, the NRC On Site Representative attends DOE/Nye County planning meetings and is made aware of all plans and updates to plans as they are made.	Not Received
USFIC.5.04	SZ1	Provide additional information to further justify the uncertainty distribution of flow path lengths in the alluvium. This information currently resides in the Uncertainty Distribution for Stochastic Parameters AMR. DOE will provide additional information, to include Nye County data as available, to further justify the uncertainty distribution of flowpath lengths in alluvium in updates to the Uncertainty Distribution for Stochastic Parameters AMR and to the Saturated Zone Flow and Transport PMR, both expected to be available in FY 2002.	Not Received
USFIC.5.05	SZ1	Provide the hydro-stratigraphic cross-sections that include the Nye County data. DOE will provide the hydrostratigraphic cross sections in an update to the Hydrogeologic Framework Model for the Saturated Zone Site-Scale Flow and Transport Model AMR expected to be available during FY 2002, subject to availability of the Nye County data.	Not Received
USFIC.5.06	SZ1	Provide a technical basis for residence time (for example, using C-14 dating on organic carbon in groundwater from both the tuffs and alluvium). DOE will provide technical basis for residence time in an update to the Geochemical and Isotopic Constraints on Groundwater Flow Directions, Mixing, and Recharge at Yucca Mountain, Nevada AMR during FY 2002.	Not Received
USFIC.5.07	SZ1	Provide all the data from SD-6 and WT-24. Some of this data currently resides in the Technical Data Management System, which is available to the NRC and CNWRA staff. DOE will include any additional data from SD-6 and WT-24 in the Technical Data Management System in February 2001.	Complete
USFIC.5.08	SZ1	Taking into account the Nye County information, provide the updated potentiometric data and map for the regional aquifer, and an analysis of vertical hydraulic gradients within the site scale model. DOE will provide an updated potentiometric map and supporting data for the uppermost aquifer in an update to the Water-Level Data Analysis for the Saturated Zone Site-Scale Flow and Transport Model AMR expected to be available in October 2001, subject to receipt of data from the Nye County program. Analysis of vertical hydraulic gradients will be addressed in the site-scale model and will be provided in the Calibration of the Site-Scale Saturated Zone Flow Model AMR expected to be available during FY 2002.	Not Received

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USFIC.5.09	SZ1	Provide additional information in an updated AMR or other document for both the regional and site scale model (for example, grid construction, horizontal and vertical view of the model grid, boundary conditions, input data sets, model output, and the process of model calibration). The updated USGS Regional Groundwater Flow Model is a USGS Product, not a Yucca Mountain Site Characterization Project product. It is anticipated that this document will be available in September 2001. DOE believes that the requested information is now available in the current version of the Calibration of the Site-Scale Saturated Zone Flow Model AMR and will be carried forward in future AMR revisions.	Not Received
USFIC.5.10	SZ1	Provide in updated documentation of the HFM that the noted discontinuity at the interface between the GFM and the HFM does not impact the evaluation of repository performance. DOE will evaluate the impact of the discontinuity between the Geologic Framework Model and the Hydrogeologic Framework Model on the assessment of repository performance and will provide the results in an update to the Hydrogeologic Framework Model for the Saturated-Zone Site-Scale Flow and Transport Model AMR during FY 2002.	Not Received
USFIC.5.11	SZ1	In order to test an alternative conceptual flow model for Yucca Mountain, run the SZ flow and transport code assuming a north-south barrier along the Solitario Canyon fault whose effect diminishes with depth or provide justification not to. DOE will run the saturated zone flow and transport model assuming the specified barrier and will provide the results in an update to the Calibration of the Site-Scale Saturated Zone Flow Model AMR expected to be available during FY 2002.	Not Received
USFIC.5.12	SZ1	Provide additional supporting arguments for the Site-Scale Saturated Zone Flow model validation or use a calibrated model that has gone through confidence building measures. The model has been calibrated and partially validated in accordance with AP 3.10Q, which is consistent with NUREG-1636. Additional confidence-building activities will be reported in a subsequent update to the Calibration of the Site-Scale Saturated Zone Flow Model AMR, expected to be available during FY 2002.	Not Received
USFIC.5.13	SZ1	Provide the evaluation of the ongoing fluid inclusion studies (for example, UNLV, State of Nevada, and USGS). DOE's consideration of the fluid inclusion studies will be documented in an update to the Saturated Zone Flow and Transport PMR expected to be available in FY 2002, subject to availability of the studies.	Not Received
USFIC.5.14	SZ1 TSPAI	Provide the updated SZ FEPs AMR. DOE will provide the updated Features, Events, and Processes in Saturated Zone Flow and Transport AMR in February 2001.	Complete
USFIC.6.01	UZ3	The DOE will provide the final sensitivity analysis on matrix diffusion (for UZ) in the TSPA-SR, Rev. 0. Due date: December 2000. The saturated zone information will be available in TSPA-SR, Rev.1, expected to be available in June 2001.	Partly Received
USFIC.6.02	UZ3	The DOE will provide the final detailed testing plan for Alcove 8. The testing plan will be provided by August 28, 2000. The NRC staff will provide comments, if any, no later than two weeks after receiving the testing plan.	Complete
USFIC.6.03	UZ3	The DOE will complete the Alcove 8 testing, taking into consideration the NRC staff comments, if any, and document the results in a DOE-approved AMR, due date: May 2001.	Not Received
USFIC.6.04	SZ1 SZ2	Provide the documentation for the C-wells testing. Use the field test data or provide justification that the data from the laboratory tests is consistent with the data from the field tests. DOE will provide the C-wells test documentation and will either use the test data or provide a justified reconciliation of the lab and field test data in C-wells document(s) in October 2001.	Not Received

NRC COMMENTS ON FEATURES, EVENTS, AND PROCESSES AND PATH FORWARD FOR RESOLUTION, INCLUDING DOE AND NRC AGREEMENTS

This appendix summarizes the NRC comments on the DOE consideration of features, events, and processes and the paths forward for their resolutions. The evaluation is presented in the form of a table (Table B–1) with the following fields:

Comment	A detailed explanation of the concern staff identified.
Path forward	Description of the agreed on path forward reached with DOE. Comments were discussed with DOE at the DOE and NRC Technical Exchanges on May 15–17 and August 6–10, 2001. Agreements on items related to Igneous Activity were reached at the September 5, 2001, DOE and NRC Technical Exchange.

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements			
Integrated Subissue	**Technical Exchange**	**Comment**	**Path Forward**
Direct1	75	Various features, events, and processes that could potentially influence the evolution of an igneous event intersecting the repository have not been identified as being relevant for disruptive events. These include 1.1.02.00.00 (Excavation/Construction) changes to the rock around the repository from excavation and construction could affect dike/repository interactions and influence how a dike behaves near the surface. Additionally, repository features such as ventilation shafts could provide a path to the surface that would bypass the repository. 1.1.04.01.00 (Incomplete Closure) if the design of the repository includes a seal at the end of the drifts strong enough to contain magma that is relied on for performance calculations, failure to complete these seals could significantly affect repository performance. 2.1.03.12.00 [Canister Failure (Long-Term)] for intrusive volcanism, credit is taken for the waste packages remaining mostly intact other than an end cap breach following magma interactions. The only waste package failure mechanism investigated to take this credit is internal gas pressure buildup. Other waste package failure mechanisms such as differential expansion of the inner and outer waste packages and phase changes in Alloy 22 from the long-term exposure to elevated temperatures are not considered. 2.1.07.02.00 (Mechanical Degradation or Collapse of Drift) could affect magma-repository interactions and affect the dose as a result of an igneous event. 2.3.01.00.00 (Topography and Morphology) the topography may affect dike propagation near the surface; dike propagation probably should be discussed as part of this feature, event, and process.	The following agreements reached at the September 5, 2001, DOE and NRC Technical Exchange, address the NRC comments: 1.1.02.00.00 (Excavation/Construction)—Igneous Activity Subissue 2, Agreement 18 1.1.04.01.00 (Incomplete Closure)—Igneous Activity Subissue 2, Agreement 18 2.1.03.12.00 [Canister Failure (Long-Term)]—Igneous Activity Subissue 2, Agreement 19 2.1.07.02.00 (Mechanical Degradation or Collapse of Drift)—Igneous Activity Subissue 2, Agreement 18 2.3.01.00.00 (Topography and Morphology)—Igneous Activity Subissue 2, Agreement 18 Igneous Activity Subissue 2, Agreement 18: DOE will evaluate how the presence of repository structures may affect magma ascent, conduit localization, and evolution of the conduit and flow system. The evaluation will include the potential effects of topography and stress, strain response on existing or new geologic structures resulting from thermal loading of high-level waste, and a range of physical conditions appropriate for the duration of igneous events. DOE will also evaluate how the presence of engineered repository structures in the License Application design (e.g., drifts, waste packages, and backfill) could affect magma flow processes for the duration of an igneous event. The evaluation will include the mechanical strength and durability of natural or engineered barriers that could restrict magma flow within intersected drifts. The results of this investigation will be documented in an update to the Analysis Model Report titled Dike Propagation and Interaction with Drifts, ANL-WIS-MD-000015, expected to be available in fiscal year 2003, or another appropriate technical document. Igneous Activity Subissue 2, Agreement 19: DOE will evaluate waste package response to stresses from thermal and mechanical effects associated with exposure to basaltic magma, considering the results of evaluations attendant to Igneous Activity Subissue 2, Agreement 18. As currently planned, the evaluation, if implemented, would include (i) appropriate at-condition strength properties and magma flow paths, for duration of an igneous event; and (ii) aging effects on materials strength properties when exposed to basaltic magmatic conditions for the duration of an igneous event, which will include the potential effects of subsequent seismically induced stresses on substantially intact waste packages. DOE will also evaluate the response of Zone 3 waste packages, or waste packages covered by backfill or rockfall, if exposed to magmatic gases at conditions appropriate for an igneous event, considering the results of evaluations attendant to Igneous Activity Subissue 2, Agreement 18. If models take credit for engineered barriers providing delay in radionuclide release, DOE will evaluate barrier performance for the duration of the hypothetical igneous event. The results

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)			
Integrated Subissue	**Technical Exchange**	**Comment**	**Path Forward**
			of this investigation would be documented in an update to the technical product Waste Package Behavior in Magma, CA–EBS–ME–000002, which would be available by the end of fiscal year 2003, or another appropriate technical document.
Direct1 Dose2	IA-1	2.3.02.02.00 (Radionuclide Accumulation in Soil) is included for irrigation deposition only; however, this screening argument is too limited because it excludes transport of volcanic ash from other areas to the critical group location (CRWMS M&O, 2001a). DOE indicated that redistribution will be accounted for by conservatively assuming that the wind is blowing toward the critical group and maintaining a high mass load in years after the event. DOE has not provided a demonstration that these conservatisms actually bound the effects of redistribution. Similar comment applies to the following items: 2.3.02.03.00 (Soil and Sediment Transport). In the screening argument, it is claimed that 100-percent south-blowing wind direction assumption accounts for aeolian and fluvial transport processes. Additional technical basis for this statement is needed. 2.3.13.02.00 (Biosphere Transport) excludes transport in surface water. 2.3.11.02.00 (Surface Runoff and Flooding) 2.3.01.00.00 (Topography and Morphology). It is necessary to consider the effect of this item on redistribution of radionuclides after an igneous event.	Igneous Activity Subissue 2, Agreement 17 addresses the NRC comments. DOE will evaluate conclusions that the risk effects (i.e., effective annual dose) of eolian and fluvial remobilization are bounded by conservative modeling assumptions in the document Total System Performance Assessment for Site Recommendation, Revison 00, ICN1. DOE will examine rates of eolian and fluvial mobilization off slopes, rates of transport in Fortymile Wash, and rates of deposition or removal at the proposed critical group location. DOE will evaluate changes in grain size caused by these processes for effects on airborne particle concentrations. DOE will also evaluate the inherent assumption in the mass loading model that the concentration of radionuclides on soil in the air is equivalent to the concentration of radionuclides on soil on the ground does not underestimate dose (i.e., radionuclides important to dose do not preferentially attach to smaller particles). DOE will document the results of investigations in the Analysis Model Report titled Eruptive Processes and Soil Redistribution, ANL–MGR–GS–000002, expected to be available in fiscal year 2003 and in the Analysis Model Report titled Input Parameter Values for External and Inhalation Radiation Exposure Analysis, ANL–MGR–MD–000001, to be available in fiscal year 2003, or another appropriate technical document.
Dose1 Dose2 Dose3	17	DOE selected a subset of the full list of features, events, and processes as applicable for biosphere screening in CRWMS M&O (2001a). Some entries potentially applicable to biosphere dose conversion factor calculations (that should at least be considered for screening) have not been included in the scope of the document (CRWMS M&O, 2001a). These include 2.3.11.04.00 (Groundwater Discharge to Surface) 1.3.07.02.00 (Water Table Rise) 3.2.10.00.00 (Atmospheric Transport of Contaminants) 1.2.04.01.00 (Igneous Activity) 2.2.08.01.00 (Groundwater Chemistry/Composition in Unsaturated Zone and Saturated Zone) (i.e., chemical species can impact dose coefficient selection) 2.2.08.11.00 (Distribution and Release of Nuclides from the Geosphere) 3.1.01.01.00 (Radioactive Decay and Ingrowth) 1.2.04.07.00 (Ash Fall).	DOE will provide a technical basis in the Evaluation of the Applicability of Biosphere-Related Features, Events, and Processes, ANL–MGR–MD–000011, to address the NRC comments for 2.3.11.04.00 (Groundwater Discharge to Surface), 1.3.07.02.00 (Water Table Rise), and 2.2.08.11.00 (Distribution and Release of Nuclides from the Geosphere). No further action is required for 3.2.10.00.00 (Atmospheric Transport of Contaminants) and 1.2.04.01.00 (Igneous Activity). DOE agreed to provide clarification of the screening argument in the Evaluation of the Applicability of Biosphere-Related Features, Events, and Processes, ANL–MGR–MD–000011, for 2.2.08.01.00 (Groundwater Chemistry/Composition in Unsaturated Zone and Saturated Zone), to address the NRC comment. DOE will add links to the Evaluation of the Applicability of Biosphere-Related Features, Events, and Processes, ANL–MGR–MD–000011, for 3.1.01.01.00 (Radioactive Decay and Ingrowth) and 1.2.04.07.00 (Ashfall), to address the NRC comment.
ENG1 ENG4 UZ3	57	1.1.02.03.00 (Undesirable Materials Left) is screened out on the basis of low consequences (CRWMS M&O, 2001f). Although a report cited by the DOE (CRWMS M&O, 1995) provides an analysis of acceptable upper bounds on materials introduced into the repository, no analysis has been conducted to determine if the current design will meet these limits. An assumption	DOE agreed to provide the technical basis for the screening argument in the Engineered Barrier System Features, Events, and Processes, ANL–WIS–PA–000002, to address the NRC comment. The technical basis involves use of the Waste Isolation Evaluation: Tracers, Fluids, and Materials, and Excavation Methods for Use in the Package 2C

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)			
Integrated Subissue	**Technical Exchange**	**Comment**	**Path Forward**
		that the limits will be adhered to during the preclosure period is considered inadequate to exclude 1.1.02.03.00 (Undesirable Materials Left). DOE should provide adequate technical basis for the effect of introduced materials on water chemistry.	Exploratory Studies Facility Construction, BABE00000–01717–2200–00007 Revision 04. As part of Container Life and Source Term Subissue 1, Agreement 1, DOE also agreed to provide additional justification on the effect of introduced materials on water chemistry in a revision to the analysis and model report, Environment on the Surfaces of the Drip Shield and Waste Package Outer Barrier AMR, ANL–EBS–MD–000001, before license application.
UZ2	68	1.2.02.01.00 (Fractures) Is screened as included for seepage and is screened as excluded on the basis of low consequence for permanent effects (CRWMS M&O, 2001b). Generation of new fractures and reactivation of preexisting fractures may significantly change the flow and transport paths. Newly formed and reactivated fractures typically result from thermal, seismic, or tectonic events. Thermally induced changes in stress may result in permeability changes between drifts that could act to divert flow toward drifts. See also comment on 2.2.06.01.00 [Changes in Stress (Due to Thermal, Seismic, or Tectonic Effects) Change Porosity and Permeability of Rock].	The thermal-mechanical effects on rock properties are addressed by an existing DOE and NRC agreement (Repository Design and Thermal-Mechanical Effects Subissue 3, Agreements 20 and 21). The FEPs in Thermal Hydrology and Coupled Processes, ANL–NBS–MD–000004, will be revised on completion to meet this agreement.
ENG2 UZ2 SZ1	J-25	1.2.02.02.00 (Faulting). Changes of fault characteristics have been screened as excluded on the basis of low consequence (CRWMS M&O, 2000b); formation of new faults has been excluded on the basis of low probability. 1.2.02.03.00 (Fault Movement Shears Waste Container) has been excluded on the basis of low probability. 1.2.03.02.00 (Seismic Vibration Causes Container Failure) has been excluded on the basis of low consequence (CRWMS M&O, 2000a). In these items, the DOE screening argument relies, in large part, on the median values of fault displacements and ground motions for postclosure (less than 10^{-6}/year), rather than the mean values. The screening arguments do not provide sufficient technical justification for staff review. The staff consider that the mean more reliably incorporates uncertainty and is a more reasonable and prudent statistical measure than the median. DOE agreed to address this concern in a forthcoming Request for Additional Information.	This issue is addressed by existing agreements between DOE and NRC (Structural Deformation and Seismicity Subissue 1 Agreement 2) and an NRC letter dated August 3, 2001. Features, Events, and Processes: Screening for Disruptive Events, ANL–WIS–MD–000005, will be revised on completion of this work.
ENG2	J-26	The screening argument for 1.2.02.03.00 (Fault Movement Shears Waste Container) is based, in part, on specific setback distances that will be used by DOE in the repository design (CRWMS M&O, 2000a). The setback distances are a function of fault displacement magnitudes. Thus, the setback values used in the design may need to be reassessed after the displacement issue is resolved.	This issue is addressed by existing agreements between DOE and NRC (Structural Deformation and Seismicity Subissue 1 Agreement 2) and an NRC letter dated August 3, 2001. Features, Events, and Processes: Screening for Disruptive Events, ANL–WIS–MD–000005, will be revised on completion of this work.
ENG2 UZ2 SZ1	J-27	1.2.03.01.00 (Seismic Activity) has been screened as excluded on the basis of low consequence of effects on such components as the drip shield and waste package and included with regard to effects on cladding (CRWMS M&O, 2000a). The distributions for	This issue is addressed by existing agreements between DOE and NRC (Structural Deformation and Seismicity Subissue 2 Agreement 1) and an NRC letter dated August 3, 2001. Features, Events, and

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)			
Integrated Subissue	**Technical Exchange**	**Comment**	**Path Forward**
		ground-motion parameters were developed using the Probabilistic Seismic Hazard Assessment Expert Elicitation. There are apparent discrepancies among these input parameters from several experts. DOE agreed to address this concern in a forthcoming Request for Additional Information.	Processes: Screening for Disruptive Events, ANL–WIS–MD–000005, will be revised on completion of this work.
ENG2	78	1.2.03.02.00 (Seismic Vibration Causes Container Failure) features, events, and processes have been excluded from consideration in the total system performance assessment code (CRWMS M&O, 2000a, 2001c). The screening argument cites preliminary seismic analyses of the drip shield and waste package as the basis for this screening decision (CRWMS M&O, 2000b). Because these analyses were not available at the time of this review, it is not clear whether the appropriate combinations of dead loads (caused by drift collapse, fallen rock blocks, or both), rock block impacts, and seismic excitation were considered. Moreover, the ability of these loads to initiate cracks, propagate preexisting cracks, or both may not have been adequately addressed. In addition, DOE has not demonstrated that the drip shield, pallet, and waste package will respond in a purely elastic manner when subjected to the aforementioned loading conditions. The screening argument for 1.2.03.02.00 (Seismic Vibration Causes Container Failure) also states "… it does not appear credible that the drip shield would be breached, because the drip shield has been designed to withstand up to a 6-MT rockfall" based on the rockfall on drip shield analyses performed by DOE (CRWMS M&O, 2000c). DOE, however, has not adequately demonstrated that the drip shield has, in fact, been designed to withstand 6-MT rock blocks {see the comments on 2.1.07.01.00 [Rockfall (Large Block)], 2.1.07.02.00 (Mechanical Degradation or Collapse of Drift), and 2.1.07.05.00 (Creeping of Metallic Materials in the Engineered Barrier Subsystem) for additional discussion relevant to rockfall and seismic analyses}. See also comment on 1.2.02.02.00 (Faulting).	Existing agreements from the Container Life and Source Term Subissue 2, Agreements 2 and 8; Repository Design and Thermal-Mechanical Effects Subissue 3, Agreements 17 and 19; and Structural Deformation and Seismicity Subissue 1, Agreement 2, and Subissue 2, Agreement 3, address related work. DOE agreed to provide clarification of the screening argument in FEPs Screening of Processes and Issues in Drip Shield and Waste Package Degradation, ANL–EBS–PA–000002, and Features, Events, and Processes: Screening for Disruptive Events, ANL–WIS–MD–000005, to address the NRC comment.
UZ3 Direct1	J-22	1.2.04.02.00 (Igneous Activity Causes Changes to Rock Properties) is screened as excluded from the radionuclide transport in the unsaturated zone abstraction, on the basis of low consequence (CRWMS M&O, 2000d, 2001d). Although various of the arguments presented (scale and duration) may be reasonable, natural analogs (CRWMS M&O, 2000e) suggest time scales of thousands of years (Ratcliff, et al., 1994) and alteration scales of tens of meters. Furthermore, modeling studies of the effects of silica redistribution on fracture porosity and permeability (CRWMS M&O, 2000e) have yielded conflicting results (Matyskiela, 1997), suggesting additional clarification is needed. Probability may also be an aspect to use in developing screening arguments for 1.2.04.02.00 (Igneous Activity Causes Changes to Rock Properties) provided probability is consistent with the probabilities used for the igneous disruptive scenario.	This issue is addressed by existing agreements between DOE and NRC (Evolution of the Near-Field Environment Subissue 1 Agreement 4, Subissue 4 Agreements 3 and 4, and Radionuclide Transport Subissue 1 Agreement 5). Features, Events, and Processes in Unsaturated Zone Flow and Transport, ANL–NBS–MD–000001, will be revised on completion of this work.

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)			
Integrated Subissue	**Technical Exchange**	**Comment**	**Path Forward**
SZ1 Dose1 Dose2	8	1.2.04.07.00 (Ash Fall). DOE assumes that ash fall blankets the region between the repository and the compliance boundary (CRWMS M&O, 2000f). Radionuclides associated with ash fall are then assumed to be transported instantaneously into the saturated zone. DOE presented only the case for uniform distribution. Moreover, parameter values and models used in the ash fall analysis are not clear. Some parameters used in the model are not well documented and other parameters, such as the number of waste packages that fail, are not viewed as conservative. DOE should provide additional bases for the choice of models and parameters used to screen this item.	DOE agreed to provide clarification of the screening argument in the Features, Events, and Processes in SZ Flow and Transport, ANL–NBS–MD–000002, to address the NRC comment.
Dose1 Dose2	J-24	1.2.04.07.00 (Ash Fall). The screening argument in CRWMS M&O (2000f) for ash fall impacting the saturated zone [i.e., secondary 1.2.04.07.01 (Soil Leaching Following Ash Fall)] includes a three order-of-magnitude error in calculation of the concentration of radionuclides in the well water. Although conservative assumptions are used in the analysis, the error found in Table 6-1 would cause the calculated dose to be 0.161Sv[16.1 rem], instead of 1.61×10^{-2} [1.61×10^{-4}], and would not support a low-consequence screening argument.	DOE agreed to provide the technical basis for the screening argument in Features, Events, and Processes in SZ Flow and Transport, ANL–NBS–MD–000002, screening argument, to address the NRC comment.
SZ2 ENG3	4	1.2.06.00.00 (Hydrothermal Activity). [Saturated Zone]: In CRWMS M&O (2001e), this item is excluded on the basis of low consequence. For saturated zone transport, the argument is that the adopted K_d distributions account for possible lithologic changes and thermal effects, with reference to CRWMS M&O (2000g). However, the latter document does not provide a clear technical basis that the K_ds were derived in such a fashion. In addition, though the screening argument is based on low consequence, there is a reference at the conclusion of the supplemental discussion to the low probability of hydrothermal activity (CRWMS M&O, 2001e). Resolution of this issue is necessary to address the issue of changes in the geothermal gradient in 2.2.10.13.00 [Density-Driven Groundwater Flow (Thermal)]. DOE should provide a stronger technical basis for the assertion that possible hydrothermal effects on K_d values are accounted for in the total system performance assessment. [Unsaturated Zone]: This item is excluded in the unsaturated zone on the basis of low consequence and low probability (CRWMS M&O, 2000h). DOE has not yet provided sufficient technical bases for models explaining elevated temperatures in the unsaturated zone from approximately 122 million years ago or adequately addressed the timing and mode of formation of Type B faults, which record elevated temperatures.	[Saturated Zone]: This issue is addressed by existing DOE and NRC agreements (Radionuclide Transport Subissue 1, Agreement 5, and Subissue 2, Agreement 10). Features, Events, and Processes in SZ Flow and Transport, ANL–NBS–MD–000002, will be updated as necessary to reflect the results of these existing agreements. [Unsaturated Zone]: As part of the Evolution of the Near-Field Environment Subissue 2, Agreement 3, DOE agreed to provide additional technical bases for the screening of 1.2.06.00 (Hydrothermal Activity), addressing points discussed at the January 2001 Evolution of the Near-Field Environment Technical Exchange. DOE agreed to revise the screening argument in a future revision of Features, Events, and Processes in UZ Flow and Transport AMR, (ANL–NBS–MD–000001), expected to be available in fiscal year 2002.
UZ2	J-23	1.2.06.00.00 (Hydrothermal Activity). Excluded on the basis of low consequence for basaltic magmatism and low probability for silicic magmatism (CRWMS M&O, 2001d). A consistent approach for the screening arguments is needed. The screening argument is considered incomplete because (i) past hydrothermal activity in the Yucca Mountain region is not clearly	This issue is addressed by existing DOE/NRC agreements (Radionuclide Transport Subissue 1 Agreement 5 and Subissue 2 Agreement 10). Features, Events, and Processes in SZ Flow and Transport, ANL-NBS-MD-000002, will be updated as necessary to reflect the results of these existing agreements.

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)			
Integrated Subissue	**Technical Exchange**	**Comment**	**Path Forward**
		related to basaltic igneous activity and (ii) probability screening arguments in CRWMS M&O (2001d) are incomplete with respect to silicic magmatism. In addition, DOE cites unpublished studies by the U.S. Geological Survey and the University of Nevada, Las Vegas that reportedly demonstrates hydrothermal activity was a site characteristic until about 2 million years ago. Additional unpublished work by Dublyanski and others, however, does not support this conclusion. None of the unpublished work, however, has supported the conclusion that the likelihood of hydrothermal activity at Yucca Mountain during the next 10,000 years is clearly <1:10,000. Absent a clear linkage to the consequences of basaltic igneous activity, or a demonstrated technical basis for probability values below 1 in 10,000 in 10,000 years, DOE has an incomplete technical basis to screen 1.2.06.00.00 (Hydrothermal Activity) from further consideration.	
UZ1 Dose2 Dose3	J-16	1.2.07.01.00 (Erosion/Denudation) is screened as excluded on the basis of low consequence (CRWMS M&O, 2001d). The rationale for exclusion from the unsaturated zone on the basis of low consequence is incomplete. It is necessary to consider onset and extent of erosion caused by construction and characterization activities at the ground surface and the long-term effects on shallow infiltration.	DOE agreed to provide clarification of the screening argument in Features, Events, and Processes in UZ Flow and Transport, ANL–NBS–MD–000001, to address the NRC comment.
UZ1	J-17	1.2.10.02.00 (Hydrologic Response to Igneous Activity). Excluded based on low consequence (CRWMS M&O, 2001d). Argument to exclude focuses on intrusive events. It should be noted that extrusive events could increase shallow infiltration for the repository in two ways: (i) lava flow would modify or dam a wash overlying the repository and (ii) volcanic fragment and ash layer, which would be highly permeable, may act to trap infiltrating water, shield it from evaporation, and reduce transpiration—all leading to increased shallow infiltration across the repository. There are no data to support or exclude the temporal extent of increased shallow infiltration, though this could be bounded from decades to thousands of years.	DOE agreed to provide the technical basis for the screening argument in Features, Events, and Processes in UZ Flow and Transport, ANL–NBS–MD–000001, screening argument, to address the NRC comment.
UZ1	J-18	1.3.04.00.00 (Periglacial Effects). Excluded by low probability (CRWMS M&O, 2001d). Although other periglacial processes will not likely occur at Yucca Mountain, the freeze/thaw process is currently active. Freeze/thaw mechanical erosion will likely increase as the climate cools, however. The magnitude of erosion will not likely be significant even during the cooler climate condition. The screening argument should be clarified to acknowledge the current freeze/thaw process.	DOE agreed to provide clarification of the screening argument in Features, Events, and Processes in UZ Flow and Transport, ANL–NBS–MD–000001, to address the NRC comment.
SZ1 SZ2 Dose1 Dose3	11	1.3.07.01.00 (Drought/Water Table Decline). According to information in CRWMS M&O (2001e), this item is excluded because of low consequence. DOE states "... a lower water table could result in less travel through the alluvial aquifer and as a result, less sorption and retardation of the contaminant plume." However, no evidence is presented that precludes a water table decline. Current flow models assume that groundwater flow through the saturated alluvium is	This issue is addressed by existing DOE and NRC agreements (Radionuclide Transport Subissue 2, Agreement 8, and Unsaturated and Saturated Flow Under Isothermal Conditions Subissue 5, Agreement 4). Features, Events, and Processes in SZ Flow and Transport, ANL–NBS–MD–000002, will be updated as necessary to reflect the results of these existing agreements and to clarify the screening argument.

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)			
Integrated Subissue	**Technical Exchange**	**Comment**	**Path Forward**
		relatively shallow. As water tables decline, how will flow through the alluvium be affected? Is it possible that a larger component of flow will be through the deep carbonate system? Will the upward gradient observed at some locations be affected? Are there distinct pathways that are dependent on elevation of the water table? It is likely that the transport times will stay the same or increase from water table decline, however, the exclusion argument provided seems insufficient. Additional technical justification is required to fully exclude 1.3.07.01.00 (Drought/Water Table Decline).	
SZ2	7	1.4.06.01.00 (Altered Soil or Surface Water Chemistry). This item is excluded on the basis of low probability (CRWMS M&O, 2001d), but it is not addressed as part of the scope of document ANL–NBS–MD–000002 (CRWMS M&O, 2000f). The probability argument is not supported by a calculation or estimate. This item is possibly relevant for the Integrated Subissue Radionuclide Transport in the Saturated Zone because of possible changes in groundwater chemistry.	DOE agreed to provide clarification of the screening argument in Features, Events, and Processes in SZ Flow and Transport, ANL–NBS–MD–000002, to address the NRC comments. The analysis and model report will also address the aggregate effects of 1.4.06.01.00 (Altered Soil or Surface Water Chemistry) on the unsaturated and saturated zones.
Dose3 Dose1	18	The biosphere analysis and model report on features, events, and processes (CRWMS M&O, 2001a) indicates that any future changes in 1.4.07.01.00 (Water Management Activities) can be excluded based on 10 CFR Part 63. This item includes well pumping from an aquifer as a water management activity. The conclusion that changes to water management activities may be excluded is not supported by the regulation. The draft regulation indicates the behaviors and characteristics of the farming community shall be consistent with current conditions of the region surrounding the Yucca Mountain site and that climate evolution shall be consistent with the geologic record. As the climate becomes wetter and cooler, the farming community is likely to pump less water out of the aquifer, consistent with sites analogous to the predicted future climate of Yucca Mountain. This reduction in pumping would not be considered a change in the behavior or characteristics of the critical group because the community would still be raising similar crops using similar farming methods.	DOE agreed to provide clarification of the screening argument in Features, Events, and Processes in SZ Flow and Transport, ANL–NBS–MD–000002, to address the NRC comment.
ENG1	48	2.1.01.04.00 (Spatial Heterogeneity of Emplaced Waste) is screened as excluded on the basis of low consequence (CRWMS M&O, 2000i). Waste placed in Yucca Mountain will have physical, chemical, and radiological properties that will vary. The effect of spatial heterogeneity of the waste on repository-scale response is excluded based on low consequence, however, the heterogeneity within a waste package is implicitly included in the evaluation of in-package temperature used to determine perforation of the commercial spent nuclear fuel cladding. Spatial variability that may affect degradation of engineering barriers, such as conditions leading to crevice corrosion versus passive corrosion of an outer container, is not considered in this feature-event-process.	Spatial variability that may affect degradation of the waste package will be addressed as part of the resolution of an existing agreement (Container Life and Source Term Subissue 1, Agreement 1). The scope of the agreement includes evaluation of the range of chemical environments on the waste package.
ENG 4	50	2.1.02.13.00 (General Corrosion of Cladding). Excluded based on low probability of occurrence (CRWMS M&O, 2000j). Although general corrosion of	DOE agreed to provide clarification of the screening argument in Clad Degradation Features, Events, and

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)			
Integrated Subissue	**Technical Exchange**	**Comment**	**Path Forward**
		cladding could expose large areas of irradiated fuel matrix and produce hydrides, it is argued that this corrosion is a slow process. The arguments are based on extrapolation to low temperatures at test data obtained at temperatures above 250 °C [482 °F] and in measurements of oxide thickness from specific fuel rods after reactor operation and exposure to water in reactor pool storage.	Processes Analysis and Model Report, ANL–WIS–MD–000008, to address the NRC comment.
ENG4	51	2.1.02.14.00 (Microbially Induced Corrosion of Cladding). Included as part of localized corrosion model on the basis that microbial activity may induce local pH decreases and the local acidic environment may produce multiple penetrations of the cladding (CRWMS M&O, 2000j). It is stated, however, that microbially induced corrosion resulting from sulfide produced by sulfate-reducing bacteria and organic acid-producing bacteria is not expected to occur, because of resistance of zirconium to these species. The arguments are poorly worded stating that microbially induced corrosion is not expected to occur (not probable or credible) because microbial activity is screened out at the scale of the repository model as a significant bulk process. The argument of local acidic pH causing localized corrosion of cladding contradicts experimental evidence showing that zirconium alloys are resistant to corrosion in reducing and oxidizing acids. In addition, the argument contradicts other DOE arguments to screen out pitting corrosion by chloride anions {see 2.1.02.16.00 [Localized Corrosion (Pitting) of Cladding]}. Screening arguments for inclusion or exclusion should be consistent with screening decisions for related entries [see 2.1.02.15.00 (Acid Corrosion of Cladding from Radiolysis)]. A third group of bacteria iron oxidizers should also be considered in the analysis (NRC, 2001).	This issue is addressed by an existing DOE and NRC agreement (Container Life and Source Term Subissue 3, Agreement 7). DOE agreed to provide clarification of the screening argument in Clad Degradation—FEPs Screening Arguments, ANL–WIS–MD–000008, Analysis and Model Report to address the NRC comment. The new cladding local corrosion model will reference In-Drift Microbial Communities Analysis and Model Report, ANL–EBS–MD–000038, which includes discussion of iron oxidizing bacteria. Clad Degradation—FEPs Screening Arguments, ANL–WIS–MD–000008, Analysis and Model Report will be revised to be consistent with the updated Summary–Abstraction Analysis and Model Report.
ENG4	49	2.1.02.15.00 (Acid Corrosion of Cladding from Radiolysis). Included as part of the localized corrosion model on the basis that formation of HNO_3 and H_2O_2 ions [sic] by radiolysis can enhance corrosion of cladding (CRWMS M&O, 2000j). It is stated, however, that zirconium has excellent corrosion resistance to HNO_3 and concentrated H_2O_2. The arguments are poorly worded, stating that radiolysis is not expected to occur until waste package failure; then, the gamma dose will be too low to produce sufficient HNO_3 and H_2O_2 to promote general corrosion, however, localized corrosion could be possible. The argument of local acidic pH causing localized corrosion of cladding contradicts experimental evidence showing that zirconium alloys are resistant to corrosion in reducing and oxidizing acids. In addition, the argument contradicts other DOE arguments to screen out pitting corrosion by chloride anions {see 2.1.02.16.00 [Localized Corrosion (Pitting) of Cladding]}. In the Basis for Screening, undue consideration is given to alkaline conditions arising from the concrete liner, whereas the possibility of acidic conditions (pH< 2) is not discussed.	Radiolysis is addressed by an existing DOE and NRC agreement (Container Life and Source Term Subissue 3, Agreement 7). DOE agreed to provide clarification of the screening argument in Clad Degradation—FEPs Screening Arguments, ANL–WIS–MD–000008, to address the NRC comment.
ENG4	47	2.1.02.17.00 [Localized Corrosion (Crevice Corrosion) of Cladding]. Excluded based on low probability of occurrence (CRWMS M&O, 2000j). Experimental	DOE agreed to provide clarification of the screening argument in Clad Degradation—FEPs Screening Arguments, ANL–WIS–MD–000008, to address the

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)			
Integrated Subissue	**Technical Exchange**	**Comment**	**Path Forward**
		evidence is cited to indicate that crevice corrosion has not been observed in zirconium alloys exposed to chloride solutions, including NRC and CNWRA results. There is a need to develop a better understanding of localized corrosion of zirconium alloys before confirming this conclusion because the data are limited. In the report, Clad Degradation—Local Corrosion of Zirconium and Its Alloys Under Repository Conditions (CRWMS M&O, 2000k). It is noted that crevice corrosion may occur in the presence of fluoride ions.	NRC comment using data relevant to the proposed repository. In addition, Container Life and Source Term Subissue 3, Agreement 7, also addresses part of the concern.
ENG4	41	2.1.02.20.00 (Pressurization from Helium Production Causes Cladding Failure). Included as a process of internal gas pressure buildup that increases the cladding stress contributing to delayed hydride cracking and strain (creep?) failures (CRWMS M&O, 2000j). The wording could be more precise in the text where it is stated that helium production from alpha decay is the main source of pressure buildup.	DOE agreed to provide clarification of the screening argument in Clad Degradation—FEPs Screening Arguments, ANL–WIS–MD–000008, to address the NRC comment.
ENG4	53	2.1.02.22.00 (Hydride Embrittlement of Cladding). Excluded based on low probability of occurrence (CRWMS M&O, 2000j). The DOE screening argument states that the in-package environment and cladding stresses are not conducive to hydride cracking. The NRC staff believe that reorientation of preexisting hydride and embrittlement depends on temperature in addition to the required stresses. Clarification is needed on the cladding temperature and stress distributions used in the analysis. Several of the secondary features, events, and processes related to various processes leading to hydrogen entry into the cladding are listed next. 2.1.02.22.01 [Hydride Embrittlement from Zirconium Corrosion (of Cladding)]. Excluded because of low probability of occurrence because the hydrogen pickup as a result of cladding corrosion is low, because of the low corrosion rate, and because of the relatively small pickup fraction. The experimental hydrogen pickup fraction is provided, and it is argued the corrosion rate is low. The conclusion DOE reached regarding failure of cladding as a result of hydrogen pickup from general corrosion is acceptable. The screening arguments, however, can be justified better using quantitative arguments for the corrosion rate during disposal conditions. 2.1.02.22.02 [Hydride Embrittlement from Waste Package Corrosion and Hydrogen Absorption (of Cladding)]. Excluded because of the low probability of occurrence because the hydrogen generated by corrosion of waste packages and waste package internals and present as a molecule in gas or dissolved in water is not directly absorbed by the cladding. It is argued, on the basis of experimental data, that hydrogen absorption occurred through the reaction with water and not from the dissolved molecular hydrogen. The conclusion DOE reached regarding failure of cladding as a result of absorption of hydrogen gas generated by corrosion of waste package materials is acceptable. The screening arguments, however, can be better organized.	DOE agreed to provide clarification of the screening argument in Clad Degradation—FEPs Screening Arguments, ANL–WIS–MD–000008, to address the NRC comment.

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)			
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		2.1.02.22.03 (Hydride Embrittlement from Galvanic Corrosion of Waste Package Contacting Cladding). Excluded because of the low probability of occurrence because corrosion of waste package internals will not result in hydriding of cladding. It is argued, using some experimental data as bases, that galvanic coupling to carbon steel will not be conducive to hydrogen charging because corrosion products will interrupt the electrical contact. It is claimed also that the nickel content both in Zircaloy-2 and -4 is not sufficient to induce the necessary hydrogen charging. The conclusion DOE reached regarding failure of cladding as a result of hydrogen entry from galvanic coupling with internal components of the waste packages is, in general, acceptable. The screening arguments, however, could be better supported by more relevant experimental data. 2.1.02.22.04 [Delayed Hydride Cracking (of Cladding)]. Excluded because of the low probability of occurrence. The analysis is based on the use of calculated values for the distribution of the stress intensity factor, which is compared with the threshold stress intensity for irradiated Zircaloy-2. The conclusion DOE reached regarding failure of cladding as a result of delayed hydride cracking is acceptable. The DOE analysis of delayed hydride cracking is based on material properties of cladding containing mostly circumferential hydrides. DOE needs to provide cladding temperatures and stress distributions and demonstrate these are insufficient to cause hydride reorientation. 2.1.02.22.05 [Hydride Reorientation (of Cladding)]. Excluded because of the low probability of occurrence, since tested fuel rods did not exhibit hydride reorientation at stresses higher than those expected at the repository temperatures. It is argued, in addition, that with hydride reorientation, stresses will be insufficient for hydride embrittlement and clad failure. Therefore, hydride reorientation has not been included in the model abstraction for cladding degradation. DOE agreed to provide updated documentation on the distribution of cladding temperatures and hoop stresses, which are critical parameters needed to evaluate the propensity to hydride reorientation and embrittlement [see 2.1.02.22.00 (Hydride Embrittlement of Cladding)]. 2.1.02.22.06 [Hydride Axial Migration (of Cladding)]. Excluded based on low probability because it is unlikely that sufficient hydrogen can be moved to the cooler ends of the fuel rods because of a lack of large temperature gradients in the waste packages. Based on studies for storage up to 90 years, it is concluded that the temperature gradients are not sufficient to induce redistribution of hydrides. The conclusion DOE reached regarding redistribution of hydrides caused by temperature gradients is acceptable. The screening arguments, however, should include the combined effects of stress and temperature. 2.1.02.22.07 [Hydride Embrittlement from Fuel Reaction (Causes Failure of Cladding)]. Excluded based on low probability of occurrence because hydride embrittlement from fuel reaction is only observed in boiling water reactors and a high-	

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)			
Integrated Subissue	**Technical Exchange**	**Comment**	**Path Forward**
		temperature steam environment is required for failure propagation, conditions that are unlikely even after waste package failure. The conclusion is acceptable because it is not a credible failure mechanism. However, the screening arguments are, to say the least, confusing.	
ENG1 ENG2 ENG3	34	2.1.03.02.00 (Stress Corrosion Cracking of Waste Containers). Screened as included for waste package and as excluded for drip shield on the basis of low consequence (CRWMS M&O, 2001c). The screening argument states ... Source of stress for cracks is due to cold work stress and cracks caused by rockfall. However, these cracks tend to be tight (i.e., small crack opening displacement) and fill with corrosion products and carbonate minerals. These corrosion products will limit water transport through the drip shield and, thus, not contribute significantly to the overall radionuclide release rate from the underlying failed waste packages ... . The screening argument for the drip shield is weak. Simplified DOE calculations indicate cracks will take considerable time to fill with corrosion products (Stress Corrosion Cracking of the Drip Shield, the Waste Package Outer Barrier, and the Stainless Steel Structural Material, ANL–EBS–MD–000005). Cracks that develop in the drip shield may propagate, open up, or both when subjected to subsequent loads caused by rockfall/drift collapse, seismic excitation, or both allowing significant groundwater infiltration through the drip shield.	This issue is contained in existing DOE and NRC agreement (Container Life and Source Term Subissue 2, Agreement 8). DOE will update FEPs Screening of Processes and Issues in Drip Shield and Waste Package Degradation, ANL–EBS–PA–000002, screening argument on completion of the agreement.
ENG1 ENG2 ENG3	30	2.1.03.05.00 (Microbially Mediated Corrosion of Waste Container). Screened as included for waste package and as excluded for drip shield on the basis of low consequence (CRWMS M&O, 2001c). Quantitative data on microbially influenced corrosion of drip shield materials such as Titanium Grades 7 and 16 are not available from the literature. If microbially influenced corrosion of the drip shield occurs, it would not have an effect on dose. Accelerated corrosion rates of the drip shield have been evaluated and do not affect dose (CRWMS M&O, 2000l).	This issue is addressed by an existing agreement (Container Life and Source Term Subissue 2, Agreement 8). No additional DOE action is required.
ENG1 ENG2 ENG3	35	2.1.03.08.00 (Juvenile and Early Failure of Waste Containers). Screened as included for manufacturing and welding defects in waste container degradation analysis and as excluded for manufacturing defects in drip shield degradation analysis and early failure of the waste package and drip shield from improper quality control during emplacement (CRWMS M&O, 2001c). The screening argument states The major effect of pre-existing manufacturing defects is to provide sites for crack growth by stress corrosion cracking, potentially leading to an early failure. Among other exposure condition parameters, tensile stress is required to initiated stress corrosion cracking. Because during fabrication the welds of drip shields will be annealed before placement in the emplacement drift, drip shields are not subject to stress corrosion cracking. Also, other sources of stresses in the drip shield induced by backfill and earthquakes are insignificant to cause stress corrosion cracking.	Manufacturing defects associated with the drip shield will be addressed during the resolution of an existing agreement item for the waste package (Container Life and Source Term Subissue 2, Agreement 7). FEPs Screening of Processes and Issues in Drip Shield and Waste Package Degradation, ANL–EBS–PA–000002, will be updated to reflect the results of this agreement. Mechanical integrity of the drip shield will be addressed during resolution of an existing agreement item for the waste package (Container Life and Source Term Subissue 2, Agreement 6). FEPs Screening of Processes and Issues in Drip Shield and Waste Package Degradation, ANL–EBS–PA–000002, will be updated to reflect the results of this agreement. Rockfall effects on the drip shield will be addressed during the resolution of an existing agreement item for the waste package (Container Life and Source Term Subissue 2, Agreement 8). FEPs Screening of

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)			
Integrated Subissue	**Technical Exchange**	**Comment**	**Path Forward**
		The manufacturing defects in the drip shield are excluded from TSPA analysis based on low consequence to the expected annual dose rate. The basis for this assessment is that slap-down analysis of a 21-pressurized water reactor waste packages resulted in stresses in the waste package material of less than 90 percent of the ultimate tensile strength. The impact energy associated with the emplacement error is substantially less than that expected in a vertical tip-over. Emplacement errors are not expected to result in any damage. The results of the slap-down analysis are cited as the screening analyses of several features, events, and processes. The damage reported in the slap-down analyses is concerning. Although the impact energy of emplacement errors may be substantially less than that experienced in the slap-down analyses, a proper assessment of the extent of waste package damage as a result of emplacement errors should be performed.	Processes and Issues in Drip Shield and Waste Package Degradation, ANL–EBS–PA–000002, will be updated to reflect the results of this agreement. FEPs Screening of Processes and Issues in Drip Shield and Waste Package Degradation, ANL–EBS–PA–000002, will be revised to address damage from improper quality control and emplacement of the drip shield. The criteria for damage to the waste package during emplacement will be addressed by administrative procedures for emplacement operations to be developed before operation of the facility.
ENG2	J-1	2.1.03.11.00 (Container Form) has been excluded from consideration in the total system performance assessment code (CRWMS M&O, 2001c). DOE has not addressed the varying clearance between the drip shield and different waste package designs and the concomitant effects this clearance may have on the consequences of rock block impacts, seismic excitation, or both.	This issue is addressed by existing agreements between DOE and NRC (Container Life and Source Term Subissue 2 Agreement 8). FEPs Screening of Processes and Issues in Drip Shield and Waste Package Degradation, ANL–EBS–PA–000002, will be revised on completion of this work.
UZ1 UZ2	J-19	2.1.05.01.00 (Seal Physical Properties). Excluded based on low consequence (CRWMS M&O, 2001d). It is difficult to assess this item solely based on the screening argument provided. The assessment can be performed once the actual design (ventilation tunnel locations) is released, backfill is described, and the analysis of runoff and flooding is incorporated into the screening argument. 2.1.05.02.00 (Groundwater Flow and Radionuclide Transport in Seals) and 2.1.05.03.00 (Seal Degradation). Excluded based on low consequence, using screening argument for 2.1.05.01.00 (Seal Physical Properties). The adequacy of the screening argument cannot be assessed until the actual design (ventilation tunnel locations) is released, backfill is described, and the analysis of runoff and flooding is incorporated into the screening arguments.	DOE stated it would adopt more rigorous configuration controls as the design advances. These controls will identify features, events, and processes screening arguments that could potentially change when design changes occur.
ENG3 ENG4 UZ3	J-3	2.1.06.01.00 (Degradation of Cementitious Materials in Drift). The effects of degradation of cementitious materials on seepage chemistry are excluded on the basis of low consequence (CRWMS M&O, 2001f). Exclusion is based on arguments in 2.1.09.01.00 (Properties of the Potential Carrier Plume in the Waste and Engineered Barrier Subsystem) (CRWMS M&O 2001f) that chemical models show a negligible effect of grout associated with rock bolts. NRC raised questions about these models pertaining to the treatment of evaporation and the chemical divide phenomenon (Evolution of the Near-Field Environment Technical Exchange). Concerns about grout chemical effects are related to recent observations of dripping	This issue is addressed by existing agreements between DOE and NRC (Evolution of the Near-Field Environment Subissue 2 Agreements 6, 10, and 14, and Radionuclide Transport Subissue 1 Agreement 5). Engineered Barrier System Features, Events, and Processes, ANL–WIS–PA–000002, will be revised on completion of this work.

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)			
Integrated Subissue	**Technical Exchange**	**Comment**	**Path Forward**
		from rock bolt holes in the sealed cross drift test. The argument for screening chemical effects of cementitious materials in the drift is considered inadequate. Because degradation products may affect water chemistry, and, therefore, radionuclide sorption behavior, the effect of this database entry on radionuclide transport in the unsaturated zone should e evaluated also. Currently, this entry is not addressed for the unsaturated zone (CRWMS M&O, 2001d). It is necessary to the development of technical bases that degradation of cementitious materials has a negligible effect on water chemistry within and below the drift. Screening would be supported by addressing the following technical exchange agreements: Evolution of the Near-Field Environment Subissue 2, Agreements 6 and 14, deal with model and laboratory results pertinent to the effects of engineered barrier subsystem materials, including cementitious on water chemistry. Radionuclide Transport Subissue 1, Agreement 5, and Subissue 2, Agreement 10, concern the technical bases for transport parameter uncertainty distributions.	
ENG2 UZ3	J-2 J-4	2.1.06.05.00 (Degradation of Invert and Pedestal) has been screened as excluded on the basis of low consequence (CRWMS M&O, 2001f). [Comment 1]: Rock block impact orientations with the waste package will be affected by degradation of the invert. Comment 2.1.07.01.00 [Rockfall (large block)] stated angled rock block impacts near the closure lid weld may have undesirable consequences. Furthermore, stability of the waste package during seismic excitation will be affected by a degraded invert foundation. Corrosion of the steel pallet components should be considered when evaluating stability of the waste package on its supporting pallet on a degraded invert foundation. [Comment 2]: Invert degradation is excluded on the basis of low consequence (CRWMS M&O, 2001f). The argument that changes to diffusive properties of the invert will be negligible to dose is not supported by demonstration (by sensitivity analyses) of the significant effect of diffusive release through the invert during the first 20,000 years (CRWMS M&O, 2000m, Volume II, Section 3.3). The sensitivity shown in the Repository Safety Strategy also applies to the first 10,000 years. The screening argument contradicts this information. The screening argument should directly address possible effects of degradation on invert diffusive properties.	[Comment 1]: This issue is addressed by existing agreements between DOE and NRC (Container Life and Source Term Subissue 2 Agreement 8). Engineered Barrier System Features, Events, and Processes, ANL–WIS–PA–000002, will be revised on completion of this work. [Comment 2]: DOE agreed to provide the technical basis for the screening argument in Engineered Barrier System Features, Events, and Processes, ANL–WIS–PA–000002, to address the NRC comment.
ENG1	39	2.1.06.06.00 (Effects and Degradation of Drip Shield). Excluded based on low consequence (CRWMS M&O, 2001c). The drip shield is an important component of the engineered barrier subsystem, and its function and degradation are explicitly considered in the total	The ability of the additional loading combinations to initiate, propagate, or both preexisting cracks is being addressed in existing agreements (Container Life and Source Term Subissue 2, Agreements 8 and 9). DOE agreed to provide the technical basis for the screening

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)

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		system performance assessment. The degradation of the drip shield from corrosion processes is considered directly in the model abstraction for waste package degradation, whereas remaining aspects of drip shield behavior are considered as part of the engineered barrier subsystem analysis. For the secondary feature-event-process 2.1.06.06.01 (Oxygen Embrittlement of Ti Drip Shield), DOE argues that oxygen embrittlement is explicitly considered in the screening argument, but no discussion is provided. It is noted that this issue is most relevant to mechanical failure of the drip shield, which is discussed in 2.1.07.01.00 (Rockfall) and 2.1.07.02.00 (Mechanical Degradation or Drift Collapse). Although physical and chemical degradation processes have been included in the total system performance assessment, their effects on the ability of the drip shield to withstand dead loads (caused by drift collapse, fallen rock blocks, or both), rock block impacts, and seismic excitation are not accounted for in the screening arguments (CRWMS M&O, 2001c,f). CRWMS M&O (2000o) states the impact of rockfall on the degraded drip shield has been screened as excluded until more detailed structural response calculations for the drip shield under various rock loads are available. No references are provided in this document when and where these analyses will be available.	argument in FEPs Screening of Processes and Issues in Drip Shield and Waste Package Degradation, ANL–EBS–PA–000002, to address the NRC comment.
ENG1	29	2.1.06.07.00 (Effects at Material Interfaces) is screened as excluded on the basis of low consequence (CRWMS M&O, 2001c). The basic chemical processes that occur at phase boundaries (principally liquid/solid) are included in other features, events, and processes. Solid/solid contact occurs or could occur between the drip shield and the invert, backfill, or both, (if included in the Yucca Mountain project design) between the waste package and the invert, backfill, or both, (if included in the Yucca Mountain project design) between the pedestal and the waste package, drip shield, or both, and between the waste form and any other engineered barrier subsystem component materials. Because these materials are all relatively inert, no significant solid/solid interaction mechanisms have been identified relative to the basic seepage water-induced corrosion of the engineered barrier subsystem components and, hence, this feature-event-process is excluded on the basis of low consequence. However, interfaces between solid phases in contact with an aqueous phase can accelerate degradation processes such as crevice corrosion of the waste package or galvanic coupling of the drip shield to steel components [see screening arguments 2.1.03.01.00 (Corrosion of Waste Containers) and 2.1.03.04.00 (Hydride Cracking of Waste Containers and Drip Shields)].	This issue is addressed by an existing agreement (Container Life and Source Term Subissue 6, Agreement 1). DOE agreed to provide clarification of the screening argument in FEPs Screening of Processes and Issues in Drip Shield and Waste Package Degradation, ANL–EBS–PA–000002, as necessary, on completion of the agreement item.
ENG2 ENG4	79	2.1.07.01.00 [Rockfall (Large Block)]. [Disruptive Event & Waste Package]: The effects of 2.1.07.01.00 [Rockfall (Large Block)] on the drip shield and waste package have been screened as excluded (CRWMS M&O, 2000a, 2001c,f). The Drift Degradation Analysis Analysis and	Existing agreements from Repository Design and Thermal Mechanical Effects agreements (Subissue 3, Agreements 17 and 19) and Container Life and Source Term (Subissue 2, Agreements 2, 3, and 8) address related work. DOE agreed to provide clarification of the screening argument in FEPs Screening of Processes and Issues in Drip Shield and Waste Package

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)			
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		Model Report (CRWMS M&O, 2000n) indicates that thermal loading, seismicity, and time-dependent mechanical degradation of the host rock would have minor effects on the integrity of the drifts through the entire period of regulatory concern. The NRC staff at the DOE and NRC Repository Design and Thermal-Mechanical Effects Technical Exchange identified several deficiencies [see the comments on 2.1.07.02.00 (Mechanical Degradation or Collapse of Drift) for additional discussion pertaining to the DOE rockfall analyses]. As noted at the Container Life and Source Term and Repository Design and Thermal-Mechanical Effects Technical Exchanges, the rockfall on drip shield analyses (CRWMS M&O, 2000c) did not consider (i) temperature effects on mechanical material behavior, (ii) seismic motion of the supporting invert, (iii) point load impacts, (iv) appropriate material failure criteria, (v) material degradation processes, (vi) multiple rock block impacts, or (vii) boundary conditions that account for the potential interactions between the drip shield and gantry rails. Consequently, DOE has not adequately demonstrated that the drip shield has been designed to withstand 6-, 10-, or 13-MT rock-block impacts. Because the framework for the invert is constructed from carbon steel, the potential degradation may affect orientation of the waste packages during time. In other words, the invert floor cannot be expected to keep the waste packages in a horizontal position for the entire regulatory period. As a result, rock-block impacts on the waste package may occur at angles not perpendicular to the waste package longitudinal axis. Angled rock-block impacts near the closure lid welds may have significantly different results than nonangled impacts. This scenario is new and was not presented to DOE. [Cladding]: Mechanical failure of cladding from rockfall is excluded based on low probability because rockfall on an intact waste package will not cause rod failure (CRWMS M&O, 2000j). The main screening argument is based on an intact waste package. The discussion is confusing because arguments based on the presence of backfill are also used in quantitative estimates. Although the conclusion can be acceptable, because of the presence of an intact waste package, the screening arguments should be improved on the basis of appropriate calculations.	Degradation, ANL–EBS–PA–000002, and Features, Events, and Processes: Screening for Disruptive Events, ANL–WIS–MD–000005, to address the NRC comment.
ENG1 ENG2 ENG3	77	2.1.07.02.00 (Mechanical Degradation or Collapse of Drift) has been screened as excluded (CRWMS M&O, 2000a 2001f) based on CRWMS M&O (2000n), which indicates that the emplacement drifts would essentially maintain their integrity through the period of regulatory concern. DOE is expected to revise the Drift Degradation Analysis to satisfy Repository Design and Thermal-Mechanical Effects Agreements 3.17 and 3.19 (DOE and NRC Technical Exchange on Repository Design and Thermal-Mechanical Effects, February 6–8, 2001, Las Vegas, Nevada). At this stage, the screening argument is considered closed-pending given the existence of the Repository	No additional DOE action is required. Repository Design and Thermal-Mechanical Effects Subissue 3, Agreements 17 and 19, address concern on drift collapse.

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)			
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		Design and Thermal-Mechanical Effects Agreements 3.17 and 3.19. It should be noted, however, that the current state of knowledge on unsupported openings in fractured rock indicates most drifts are likely to collapse soon after cessation of maintenance. This opinion is consistent with the conclusion of the DOE expert panel on drift stability* and recent analyses of the behavior of unsupported drifts in fractured rock during seismic loading from an earthquake (Hsiung, et al., 2001). Drift collapse could have implications on temperature, chemistry, seepage into drifts, and drip shield performance.	
ENG1 ENG2 ENG3	37	2.1.07.05.00 (Creeping of Metallic Materials in the Engineered Barrier Subsystem) has been excluded from consideration in the total-system performance assessment code (CRWMS M&O, 2001c,f). Although DOE correctly points out in the screening argument (CRWMS M&O, 2001c) " ... the deformation of many titanium alloys loaded to yield point does not increase with time" (American Society for Metals International, 1990), it still does not specifically address the potential for creeping of titanium Grades 7 and 24. For example, some titanium alloys have been shown to creep at room temperatures (Ankem, et al., 1994). Creeping of the titanium drip shield subjected to dead loads caused by fallen rock blocks, drift collapse, or both could significantly reduce the clearance between the drip shield and waste package during time. As a result, the drip shield may cause substantial damage to the waste package during its dynamic response to subsequent seismic loads. In addition, creeping could potentially cause separation of the individual drip shield units.	Treatment of creep of the drip shield will be addressed as part of an existing agreement related to drip shield rockfall analyses (Container Life and Source Term Subissue 2, Agreement 8). DOE agreed to provide the technical basis for the screening argument in FEPs Screening of Processes and Issues in Drip Shield and Waste Package Degradation, ANL–EBS–PA–000002, to address the NRC comment.
ENG1 ENG2 ENG4	56	2.1.07.06.00 (Floor Buckling) has been screened as excluded in (CRWMS M&O, 2001f) and EBS Radionuclide Transport Abstraction Analyses and Model Report (CRWMS M&O, 2000o) based on analyses documented in Repository Ground Support Analysis for Viability Assessment (CRWMS M&O, 1998a), which indicate that floor heave from thermal-mechanical effects would not exceed approximately 10 mm [0.391 in]. However, to address concerns raised by the NRC staff about appropriateness of the thermal-mechanical properties used in DOE calculations (such as the analyses cited previously), DOE agreed to revise its assessment of floor buckling [Repository Design and Thermal-Mechanical Effects Agreement 3.9 (DOE and NRC Technical Exchange on Repository Design and Thermal-Mechanical Effects, February 6–8, 2001, Las Vegas, Nevada)]. Note the screening argument relies on analyses that DOE agreed to address outstanding NRC concerns in Repository Design and Thermal-Mechanical Effects Agreements 3.2–3.13 (Repository Design and Thermal-Mechanical Effects Technical Exchange, February 6–8, 2001, Las Vegas, Nevada).	This issue is addressed by existing DOE and NRC agreements (Repository Design and Thermal-Mechanical Effects Subissue 3, Agreements 2–13). DOE agreed to include the analysis of floor buckling for postclosure conditions, consistent with the site-specific parameters and loading conditions used to satisfy Repository Design and Thermal-Mechanical Effects Subissue 3, Agreements 2–13. Engineered Barrier Subsystem Features, Events, and Processes, ANL–WIS–PA–000002, will be revised to include this information.
UZ2	59	2.1.08.04.00 (Cold Traps) is screened as excluded on the basis of low consequence (CRWMS M&O, 2001f). Emplacement of waste in the drifts creates thermal gradients within the repository that may result in condensation forming on the roof of the drifts or	This issue is addressed by an existing DOE and NRC agreement (Thermal Effects on Flow Subissue 2, Agreement 5). Engineered Barrier System Features, Events, and Processes, ANL–WIS–PA–000002, will be revised on completion of this agreement.

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)			
Integrated Subissue	**Technical Exchange**	**Comment**	**Path Forward**
		elsewhere in the engineered barrier subsystem, leading to enhanced dripping on the drip shields, waste packages, or exposed waste material. The DOE Multiscale Thermohydrologic Model does not account for mass transport along the length of drifts. The only Multiscale Thermohydrologic Model submodel that includes thermal hydrology (i.e., mass transport) is a cross section of a drift, so it accounts for potential condensation only along the radial axis.	
ENG1 ENG3	42	2.1.08.07.00 (Pathways for Unsaturated Flow and Transport in the Waste and Engineered Barrier System) evaluates unsaturated flow and radionuclide transport that may occur along preferential pathways in the waste and engineered barrier subsystem (CRWMS M&O, 2000i). DOE indicates that preferential pathways are already included via "… a series of linked one-dimensional flowpaths and mixing cells through the engineered barrier subsystem, drip shield, waste package, and into the invert" (CRWMS M&O, 2000i). Staff are concerned that preferred pathways in the engineered barrier subsystem are not being evaluated at the appropriate scale. Water has been observed to drip preferentially along grouted rock bolts in the enhanced characterization of the repository block, (e.g., demonstrating that introduced materials can influence the location of preferred flow pathways). Interactions with engineered materials, such as cementitious and metallic components, can have a significant effect on evolved water and gas compositions. Because the description of 2.1.08.07.00 (Pathways for Unsaturated Flow and Transport in the Waste and Engineered Barrier System) states "Physical and chemical properties of the engineered barrier subsystem and waste form, in both intact and degraded states, should be considered in evaluating [preferential] pathways … ", staff expect the screening arguments to be based on an evaluation of these topics (NRC, 2000).	This issue is addressed by an existing DOE and NRC agreement (Evolution of the Near-Field Environment Subissue 2, Agreements 6, 10, and 14). Engineered Barrier System Features, Events, and Processes, ANL–WIS–PA–000002, will be updated on completion of these agreement items.
ENG3	54	2.1.09.02.00 (Interaction with Corrosion Products) is excluded in the engineered barrier subsystem (except for colloid-related effects) on the basis of low consequence (CRWMS M&O, 2001f). As noted in the DOE and NRC Technical Exchange on Evolution of the Near-Field Environment, changes in seepage water chemistry resulting from interactions with engineered materials and their corrosion products were not adequately addressed in CRWMS M&O (2000p). Water has been observed to drip preferentially along grouted rock bolts in the enhanced characterization of the repository block, (e.g., demonstrating that introduced materials can influence the location of preferred flow pathways). Seepage waters that have interacted with engineered materials and their corrosion products can have a significant effect on evolved water and gas compositions.	This issue is addressed by an existing DOE and NRC agreement (Evolution of the Near-Field Environment Subissue 2, Agreements 6, 10, and 14). Engineered Barrier System Features, Events, and Processes, ANL–WIS–PA–000002, will be updated on completion of these agreement items.
ENG1	36	2.1.09.03.00 (Volume Increase of Corrosion Products) is screened as excluded on the basis of low consequence (CRWMS M&O, 2001c). The presence of waste package corrosion products with higher molar volume than the uncorroded material that may change the stress state in the material being corroded is excluded in the case of the waste package based on	DOE agreed to provide the technical basis for the screening argument in FEPs Screening of Processes and Issues in Drip Shield and Waste Package Degradation, ANL–EBS–PA–000002, to address the NRC comment.

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)			
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		low consequence. These products, however, may have an effect on corrosion processes such as stress corrosion cracking of the outer container, after its initial breaching, that may affect radionuclide release [see 2.1.03.07.00 (Mechanical Impact on the Waste Container and Drip Shield)]. The possibility of additional sources of stress arising from the formation of corrosion products should be evaluated in regard to stress corrosion cracking. See comment for 2.1.11.05.00 (Differing Thermal Expansion of Repository Components).	
ENG1	55	2.1.09.07.00 (Reaction Kinetics in Waste and Engineered Barrier Subsystem). [Engineered Barrier Subsystem]: Item is screened as excluded on the basis of low consequence (CRWMS M&O, 2001f). Consideration of chemical reactions, such as radionuclide dissolution/precipitation reactions and reactions controlling the reduction-oxidation state is included by considering reaction kinetics in the in-package equilibrium model, however, reaction kinetics are excluded based on low consequence for the engineered barrier subsystem. But these processes may affect composition of the near-field environment, particularly trace elements. The effect on corrosion of container materials could be indirect and should be considered. [Waste Form Miscellaneous]: Item is screened as excluded on the basis of low consequence (CRWMS M&O, 2000i). Adequate technical bases have not been provided to demonstrate that the combination of transport processes and reaction kinetics in the engineered barrier subsystem will not adversely impact performance by altering the composition of water contacting the drip shield and waste package.	This issue is addressed by an existing DOE and NRC agreement (Evolution of the Near-Field Environment Subissue 2, Agreements 5, 8, 11, and 12). Engineered Barrier System Features, Events, and Processes, ANL–WIS–PA–000002, will be updated on completion of these agreement items.
UZ2	63	2.1.09.12.00 [Rind (Altered Zone) Formation in Waste, Engineered Barrier Subsystem, and Adjacent Rock]. The thermal-hydrological-chemical model is screened as included, and the thermal-hydrological model, effects on transport is screened as excluded on the basis of low consequence (CRWMS M&O, 2001b). Thermal-chemical processes alter the rock forming the drift walls mineralogically. These alterations have hydrological, thermal, and mineralogical properties different from the current country rock.	This issue is addressed by existing agreements between DOE and NRC (Evolution of the Near-Field Environment Subissue 1, Agreement 3). FEPs in Thermal Hydrology and Coupled Processes, ANL–NBS–MD–000004, will be revised on completion, to meet this agreement.
ENG4 UZ3 SZ2	5	2.1.09.21.00 (Suspension of Particles Larger Than Colloids). CRWMS M&O (2001e) states these particles will be included and treated as colloids. 2.1.09.21.00 (Suspension of Particles Larger Than Colloids) is not addressed in CRWMS M&O (2001d), however, and is noted as excluded in two other model components in the Yucca Mountain FEP Database (CRWMS M&O, 2001g). Furthermore, it is not clear how the effects of particles are included with colloids. 2.1.09.21.00 (Suspension of Particles Larger Than Colloids) should be addressed as part of the scope of CRWMS M&O (2001d). In addition, the integration of 2.1.09.21.00 (Suspension of Particles Larger Than Colloids) across the engineered barrier subsystem, unsaturated zone, and saturated zone should be clarified.	DOE agreed to provide clarification for the screening argument in Features, Events, and Processes in SZ Flow and Transport, ANL–NBS–MD–000002, to address the NRC comment.

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)			
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ENG4 UZ3	J-5	2.1.09.21.00 (Suspensions of Particles Larger than Colloids) is screened excluded from the engineered barrier subsystem transport and waste form release abstractions (CRWMS M&O, 2000q, 2001d). Exclusion is based on the assumption that although particles may be transported through fractures in the unsaturated zone, low groundwater velocities through the saturated zone would lead to particle settling (CRWMS M&O, 2000q), suggesting inconsistency in the screening analysis. Without quantitative measures of particle size, pore size, groundwater velocity, and chemical variability, however, these qualitative assertions are difficult to evaluate. Because DOE includes colloid formation features, events, and processes in its screening analysis, and because of the large amounts of iron particles that may be introduced in the engineered barrier subsystem, particle transport through the engineered barrier subsystem into the unsaturated zone is plausible. Exclusion of 2.1.09.21.00 (Suspensions of Particles Larger Than Colloids) may be acceptable, but it is necessary to have a more complete technical basis and calculations to support exclusion of this item on the basis of low consequence.	DOE agreed to provide clarification of the screening argument in Waste Form Colloid-Associated Concentration Limits: Abstraction and Summary ANL–WIS–MD–000012, to address the NRC comment.
UZ2	65	2.1.11.02.00 (Nonuniform Heat Distribution/Edge Effects in Repository). The thermal-hydrological and thermal-hydrological-chemical aspects are screened as included and the (thermal-mechanical effects) are screened as excluded on the basis of low consequence (CRWMS M&O, 2001b). Temperature inhomogeneities in the repository lead to localized accumulation of moisture. Uneven heating and cooling at repository edges lead to nonuniform thermal effects during both the thermal peak and the cool-down periods.	Repository-wide, nonuniform heating effects are the subject of existing DOE and NRC agreements (Thermal Effects on Flow Subissue 2, Agreement 5, and Repository Design and Thermal-Mechanical Effects Subissue 3, Agreements 20 and 21). FEPs in Thermal Hydrology and Coupled Processes, ANL–NBS–MD–000004, will be revised on completion of this agreement. Thermal-Hydrological-Mechanical continuum modeling will address nonuniform effects at a mountain scale. This information will be provided in Coupled Thermal-Hydrological-Mechanical Effects on Permeability Analysis and Model Report, ANL–NBS–HS–000037.
ENG1 ENG2	38	2.1.11.05.00 (Differing Thermal Expansion of Repository Components) has been excluded from consideration in the total-system performance assessment code (CRWMS M&O, 2001c,f). The technical basis for excluding differing thermal expansion effects on repository performance is not comprehensive nor adequate. For example, according to the screening arguments (CRWMS M&O, 2001c), ... the difference in temperature between the inside of the waste package inner barrier (316NG) and the outside of the waste package outer barrier (Alloy 22) never exceeds 2 °C [35.6 °F]. As an illustrative example, using the coefficients of thermal expansion for the two materials discussed above (i.e., Alloy 22 and 316NG) and a bounding 5 °C [41 °F] (or 5 K) temperature difference between them, the calculated strain is $2.15 \cdot 10^{-5}$. This strain is so small that thermal expansion of waste package barriers will result in a negligible effect on expected mean dose rate. A ~1 mm [0.0394 in] gap will prevent the resultant stress due to the differing thermal expansion coefficients of the waste package materials from reaching a critical level that could lead to stresses in	DOE agreed to provide the technical basis for the screening argument in FEPs Screening of Processes and Issues in Drip Shield and Waste Package Degradation, ANL–EBS–PA–000002, to address the NRC comment.

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)			
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		the waste package barriers. The Waste Package Operation Fabrication Process Report (CRWMS M&O, 2000r, Section 8.1.8) requires a loose fit between the outer barrier (Alloy 22) and the inner shell (316NG stainless steel) to accommodate the differing thermal expansion coefficients, and so 2.1.11.05.00 (Differing thermal expansion of repository components) can be excluded for the waste packages based on low consequence to the expected annual dose. The quoted rationale is not technically correct and does not address the limited clearance between the inner and outer barriers of the waste package in the axial direction, which may be as small as 2 mm [0.0787 in] according to design drawings (CRWMS M&O, 2000s). In addition, the differential thermal expansion between various invert components and the drift wall (to which they are attached) has not been addressed. 2.1.11.05.00 (Differing Thermal Expansion of Repository Components) is excluded on the basis of low consequence (CRWMS M&O, 2001c,f). Peak temperature of waste packages with 0.5-m [19.68-in] spacing and 50-year ventilation is 278 °C [532.4 °F] with backfill and 176 °C [348.8 °F] without backfill. The screening argument is that the temperature differential between the inner type 316NG barrier and the outer Alloy 22 barrier is 5C° [41 °F] with a corresponding strain of 2.15×10^{-5}. This calculation is performed using the difference between the thermal expansion coefficients for Type 316NG stainless steel and Alloy 22 using the maximum expected temperature difference between the waste package barriers. There will be at least a 1-mm [0.0394-in]gap between the barriers, and no thermal stresses are predicted. Calculations should use a temperature of the waste package rather than the difference between waste package barriers. The clearance between the inner type 316NG barrier and the outer Alloy 22 barrier is 0 to 4 mm [0.1575 in] as specified in the waste package design and fabrication process report (CRWMS M&O, 2000r). It is implicit that this clearance is specified at ambient temperature [i.e., 25 °C (77 °F)] because (i) no temperature is specified and (ii) the Alloy 22 waste package outer barrier will be heated to 371 °C [700 °F] for inner 316NG barrier cylinder installation. Using a temperature of 186 °C [366.8 °F], the calculated strain is 7.99×10^{-4}. For a waste package with clearance gaps of 1 mm [0.0394 in] or less at 25 °C [77 °F], thermal stresses will occur as a result of the differences in thermal expansion.	
ENG3	60	The exclusion of 2.1.12.01.00 (Gas Generation) and 2.1.12.05.00 (Gas Generation from Concrete) in CRWMS M&O (2000i, 2001f) is unacceptable, because adequate technical bases have not been provided to justify the characterization of chemical environments in the engineered barrier subsystem in terms of bulk water and gas compositions. The possibility of local heterogeneity in gas composition in the drift, altering the chemistry of the drip shield/waste package environment and adversely	This issue is partially addressed by an existing DOE and NRC agreement (Evolution of the Near-Field Environment Subissue 2, Agreement 6). DOE agreed to provide the technical basis for the screening argument in Engineered Barrier System Features, Events, and Processes, ANL–WIS–PA–000002, to address the NRC comment.

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)			
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		impacting repository performance, should be explored. Local variations in the efficiency of advection/diffusion processes, relative to reaction rates, should be evaluated.	
ENG1 ENG3 ENG4	32	2.1.13.01.00 (Radiolysis) is excluded based on low consequence (CRWMS M&O, 2000i, 2001c). [Waste Package]: Alpha, beta, gamma, and neutron irradiations of air saturated water can cause changes in chemical conditions (Eh, pH, and concentration of reactive radicals) and positive shifts in corrosion potential from the formation of hydrogen peroxide. DOE, on the bases of experimental work, concluded that radiolysis will not lead to localized corrosion of Alloy 22. Additional work, however, is necessary to complete the evaluation of the critical potentials related to localized corrosion of Alloy 22. [Waste Form Miscellaneous]: Screening argument considers only radiolysis of water to produce hydrogen and oxidants. No consideration of the formation of nitric acid resulting from radiolysis in presence of air. Spent nuclear fuel is expected to have higher dissolution rates at lower pH, thus, ignoring nitric acid may underestimate radionuclide release. Potential production of nitric acid from radiolysis of N_2 in air should be considered. It is necessary to consider potential effect of acid environments on the corrosion of Alloy 22 and titanium.	DOE agreed to provide additional information on critical potentials for localized corrosion at the DOE and NRC Container Life and Source Term Technical Exchange (September 12–13, 2000). DOE agreed to provide clarification of the screening argument in FEPs Screening of Processes and Issues in Drip Shield and Waste Package Degradation, ANL–EBS–PA–000002, to address the NRC comment.
ENG4 UZ3 SZ2	74	2.1.14.01.00 (Criticality in Waste and Engineered Barrier Subsystem) was preliminarily excluded in the document (CRWMS M&O, 2000t) based on low probability. A preliminary screening status was assigned because the criticality calculations were not complete for DOE spent nuclear fuel after igneous intrusion and near-field and far-field criticality of all waste types following igneous disruption. The excluded screening status will be regarded unacceptable until concerns on the calculation of the probability for criticality are addressed. Because the probability of criticality depends on the presence of a breach of the waste package barriers, most of the discussion of criticality probability is focused on the probability of waste package failure. DOE referenced the document, Probability of Criticality in 10,000 Years (CRWMS M&O, 2000u), for addressing the criticality probability from early failure by stress corrosion cracking, waste package damage after igneous intrusion, and seismic events. DOE referenced the screening argument for rockfall [2.1.07.01 (Rockfall)] for screening damage to the waste package and drip shield from seismically induced rockfall. In general, DOE needs to address the concerns raised on the waste package and mechanical disruption related features, events, and processes, and the issues raised at the Container Life and Source Term Technical Exchange before it can conclude there is no waste package breach before 10,000 years. The concerns on the probability calculation in the document, Probability of Criticality in 10,000 Years (CRWMS M&O, 2000u) are	The current criticality agreements include concerns and DOE does not need to take any additional action. The following entries are also considered closed-pending in light of existing criticality agreements: 2.1.14.02.00 (Criticality *In Situ*, Nominal Configuration, Top Breach) 2.1.14.03.00 (Criticality *In Situ*, Waste Package Internal Structures Degrade Faster Than Waste Form, Top Breach) 2.1.14.04.00 (Criticality *In Situ*, Waste Package Internal Structures Degrade at Same Rate as Waste Form, Top Breach) 2.1.14.05.00 (Criticality *In Situ*, Waste Package Internal Structures Degrade Slower Than Waste Form, Top Breach) 2.1.14.06.00 (Criticality *In Situ*, Waste Form Degrades in Place and Swells, Top Breach) 2.1.14.07.00 (Criticality *In Situ*, Bottom Breach Allows Flow Through Waste Package, Fissile Material Collects at Bottom of Waste Package) 2.1.14.08.00 (Criticality *In Situ*, Bottom Breach Allows Flow Through Waste Package, Waste Form Degrades in Place) 2.1.14.09.00 (Near-Field Criticality, Fissile Material Deposited in Near-Field Pond) 2.1.14.10.00 (Near-Field Criticality, Fissile Solution Flows into Drift Lowpoint) 2.1.14.11.00 (Near-Field Criticality, Fissile Solution Is Adsorbed or Reduced in Invert) 2.1.14.12.00 (Near-Field Criticality, Filtered Slurry, or Colloidal Stream Collects on Invert Surface)

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)			
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		• The conclusion of waste package failure probability of 2.7 x 10^{-11} from stress corrosion cracking, based on the equation in Section 6.1.1, is contrary to the total system performance assessment results that indicate the first waste package failure, using the upper-bound curve, from stress corrosion cracking at approximately 10,000 years. • The screening argument for 1.2.03.02.00 (Seismic Vibration Causes Container Failure) fails to consider the appropriate combinations of dead loads (caused by drift collapse, fallen rock blocks, or both), rock block impact, and seismic excitation or the ability of these loads to initiate cracks, propagate preexisting cracks, or both. • The screening argument for seismic events does not consider the indirect effects, such as causing dents, which could aid in the collection and channeling of water, or tilting the waste packages, which would result in greater height of the water within the waste package. Seismic shaking, combined with a sloped waste package, may also allow materials to accumulate at one end of a waste package to form a more reactive geometry. • The screening argument for seismically induced rockfall damaging the drip shield and waste package includes several deficiencies as documented in the staff review of the Drift Degradation Analysis Analysis and Model Report (CRWMS M&O 2000n) and 2.1.07.01.00 [Rockfall (Large Block)]. Other concerns related to the impact of rockfall on the waste package are reflected in the comments on the related features, events, and processes. • The calculation of the criticality probability does not fully consider mechanisms that could result in accelerated degradation of the fuel during an igneous event, such as burning Zircaloy or creep of the fuel at high temperatures. • The analysis of damage to DOE Zone 2 waste packages (CRWMS M&O, 2000u) fails to consider long-term exposure to high temperatures changing the microstructure of Alloy 22 and reducing the mechanical strength of the material (e.g., Rebak, et al., 1999) or the differences in thermal expansion between the inner barrier type 316NG stainless steel and the outer barrier Alloy 22 causing significant hoop-stress on waste package walls, in addition to the internal pressurization effects analyzed in CRWMS M&O (2000u). Analyses in CRWMS M&O (2000u) also do not consider potentially adverse chemical reactions, such as sulfidation reactions, in response to magmatic degassing or contact with basaltic magma. These processes could cause a more significant breach than the 10-cm^2 [1.55-in^2] hole currently assumed for waste packages located in DOE Zone 2 during basaltic igneous events. • The calculation does not consider any changes to drift by the magma, such as magma solidifying in the lower part of the drift, causing ponding above	2.1.14.13.00 (Near-Field Criticality Associated with Colloidal Deposits) 2.2.14.01.00 (Critical Assembly Forms Away from Repository) 2.2.14.02.00 (Far-Field Criticality, Precipitation in Organic Reducing Zone in or Near Water Table) 2.2.14.03.00 (Far-Field Criticality, Sorption on Clay/Zeolite in Topopah Springs Basal Vitrophyre) 2.2.14.04.00 (Far-Field Criticality, Precipitation Caused by Hydrothermal Upwell or Redox Front in the Saturated Zone) 2.2.14.05.00 (Far-Field Criticality, Precipitation in Perched Water Above Topopah Springs Basal Vitrophyre) 2.2.14.06.00 (Far-Field Criticality, Precipitation in Fractures of Topopah Springs Welded Rock) 2.2.14.07.00 (Far-Field Criticality, Dryout Produces Fissile Salt in a Perched Water Basin) 2.2.14.08.00 (Far-Field Criticality Associated with Colloidal Deposits)

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)			
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		and around the waste package, or fractures forming in the cooled magma, that may provide preferential pathways to the waste package. Finally, the unsaturated flow may be modified by the presence of 1,170 °C [2,138 °F] magma so current parameters may no longer be valid. • The criticality probability document is inconsistent when discussing the water content of the magma in Section 5.3.2. The text indicates the magma would consist of a conservative 5-wt% water content, but Table 5-1 lists the water content as only 0.05 wt%. The computer files provided with the document that contained the actual calculations used a more realistic water content of 1.6 percent. A water content of 5 wt% would clearly be conservative, but justification needs to be provided if a lower water content is used in the calculations.	
UZ2	69	2.2.01.01.00 (Excavation and Construction-Related Changes in the Adjacent Host Rock). Initial effects on seepage are screened as included, and permanent thermal-hydrological-chemical and thermal-mechanical effects are screened as excluded on the basis of low consequence (CRWMS M&O, 2001b). Stress relief leading to dilation of joints and fractures is expected in an axial zone of up to one diameter-width surrounding the tunnels.	Thermal-mechanical effects on rock properties are addressed by an existing DOE and NRC agreement (Repository Design and Thermal-Mechanical Effects Subissue 3, Agreements 20 and 21). FEPs in Thermal Hydrology and Coupled Processes, ANL–NBS–MD–000004, will be revised on completion of this agreement.
ENG2 UZ2	62	2.2.01.02.00 (Thermal and Other Waste and Engineered Barrier Subsystem-Related Changes in the Adjacent Host Rock) is screened as excluded on the basis of low consequence (thermal-mechanical effects) and low probability (thermal-hydrological-chemical and backfill effects) (CRWMS M&O, 2001b). Changes in host rock properties result from thermal effects or other factors related to emplacement of the waste and engineered barrier subsystem, such as mechanical or chemical effects of backfill. Properties that may be affected include rock strength, fracture spacing and block size, and hydrologic properties such as permeability. The screening argument did not consider mechanical degradation of the rock mass, such as fracture-wall rock alteration owing to long-term exposure to heat, moisture, and atmospheric conditions. Such degradation would increase the severity of mechanical failure (Ofoegbu, 2000). DOE, however, is expected to reevaluate its assessment of long-term mechanical degradation to satisfy outstanding DOE and NRC agreements (Repository Design and Thermal-Mechanical Effects Subissue 3, Agreements 11 and 19). In the analyses, it is necessary to account for long-term mechanical degradation of the host rock mass in the assessment of drift degradation, rockfall, and changes in hydrological properties and their effects on repository performance.	Thermal-mechanical effects on fractures will be addressed by existing agreements between DOE and NRC (Repository Design and Thermal-Mechanical Effects Subissue 3, Agreements 20 and 21). FEPs in Thermal Hydrology and Coupled Processes, ANL–NBS–MD–000004, will be revised on completion of this agreement. Long-term degradation of the host rock is addressed by existing agreements between DOE and NRC (Repository Design and Thermal-Mechanical Effects Subissue 3, Agreements 11 and 19). DOE will provide an improved technical basis for 2.2.01.02.00 (Thermal and Other Waste and Engineered Barrier Subsystem-Related Changes in the Adjacent Host Rock) by performing a postclosure drift deformation analysis that incorporates postclosure loads and rock properties using relevant information from existing agreements (Repository Design and Thermal-Mechanical Effects Subissue 3, Agreements 2–13). Engineered Barrier System Features, Events, and Processes, ANL–WIS–PA–000002, will be revised to include this information.
UZ2 ENG3	66	2.2.06.01.00 [Changes in Stress (Due to Thermal, Seismic, or Tectonic Effects), Change Porosity, and Permeability of Rock] is screened as excluded on the basis of low consequence and low probability (for one secondary entry) (CRWMS M&O, 2001b). Even small changes in the fracture openings cause large changes in permeability. The rock deforms according to the	Thermal-mechanical effects on rock properties are addressed by an existing DOE and NRC agreement (Repository Design and Thermal-Mechanical Effects Subissue 3, Agreements 20 and 21). FEPs in Thermal Hydrology and Coupled Processes, ANL–NBS–MD–000004, and the Features, Events, and Processes: Screening for Disruptive Events,

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)			
Integrated Subissue	**Technical Exchange**	**Comment**	**Path Forward**
		rock stress field. Changes in the groundwater flow and in the temperature field will change the stress acting on the rock, which will, in turn, change the groundwater flow. 2.2.06.01.00 [Change in Stress (Due to Thermal, Seismic, or Tectonic Effects), Change Porosity, and Permeability of Rock] is excluded as having low consequence to dose (CRWMS M&O, 2000a). However, the DOE analyses used to support the screening argument (CRWMS M&O, 2000v) did not consider water-flux diversion toward a drift from the adjacent pillar caused by increased aperture of subhorizontal fractures in the pillar from thermal-mechanical response. Such flux diversion would cause increased water flow to the drifts.	ANL–WIS–MD–000005, will be revised on completion of this agreement.
UZ2	J-20	2.2.07.05.00 (Flow and Transport in the Unsaturated Zone from Episodic Infiltration). Excluded based on low consequence (CRWMS M&O, 2001d). Screening argument asserts that episodic infiltration is expected to be attenuated by flow in the paintbrush nonwelded tuff layer such that unsaturated zone flow beneath this layer is effectively steady-state. Analyses to support this assertion, however, have only considered episodic infiltration with an average of 5 mm/yr [0.197 in/yr] infiltration flux. Area-average infiltration flux over the proposed repository horizon at Yucca Mountain is expected to exceed 20 mm/yr [0.787 in/yr] during future wetter climate conditions.	This issue is addressed by existing agreements between DOE and NRC (Unsaturated and Saturated Flow Under Isothermal Conditions Subissue 4 Agreement 4). Features, Events, and Processes in UZ Flow and Transport, ANL–NBS–MD–000001, will be revised on completion of this work.
UZ3 SZ1 SZ2	J-6	2.2.07.15.00 (Advection and Dispersion). As defined, this item does not apply to the unsaturated zone and is not discussed in CRWMS M&O (2001d). Given that advection and dispersion are key components of the DOE radionuclide transport in the unsaturated zone model abstraction, the definition of 2.2.07.15.00 (Advection and Dispersion) should be extended to include these aspects (advection and dispersion) in the unsaturated zone.	DOE will add this features, events, processes to Features, Events, and Processes in UZ Flow and Transport, ANL–NBS–MD–000001, and present the DOE discussion in the screening argument.
UZ2	USFIC-1	2.2.07.18.00 (Film Flow into Drifts) is screened as included on the basis of low consequence (low film flow rates). Higher film flow rates into drifts are considered included (CRWMS M&O, 2001d). Technical bases for the screening argument for 2.2.07.18.00 (Film Flow into Drifts) will derive from work needed to satisfy the Unsaturated and Saturated Flow Under Isothermal Conditions Subissue 4, Agreement 2.	At the Unsaturated and Saturated Flow Under Isothermal Conditions DOE and NRC Technical Exchange, DOE agreed to include the effect of the low-flow regime processes (e.g., film flow) in the DOE seepage fraction and seepage flow, or justify that it is not needed (Subissue 4, Agreement 2). No additional work is required to derive the technical basis for the screening argument for 2.2.07.18.00 (Film Flow into Drifts).
UZ3	J-7	2.2.08.01.00 (Groundwater Chemistry/Composition in Unsaturated Zone and Saturated Zone) is excluded. DOE included the current ambient groundwater conditions in the Total System Performance Assessment–Site Recommendation abstraction of radionuclide transport in the unsaturated zone, but has excluded future changes (CRWMS M&O, 2000w, 2001d). DOE asserts that thermal effects on chemistry are minimal, but assertion focuses mainly on the effects of dissolution and precipitation on hydrologic properties. The screening argument refers to a model of thermal-chemical effects on seepage water chemistry at the drift wall (CRWMS M&O,	This issue is addressed by existing agreements between DOE and NRC (Evolution of the Near-Field Environment Subissue 1 Agreement 4, and Subissue 4 Agreements 3 and 4, Radionuclide Transport Subissue 1 Agreement 5, and Subissue 2 Agreement 10). Features, Events, and Processes in UZ Flow and Transport, ANL–NBS–MD–000001, will be revised on completion of this work.

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)			
Integrated Subissue	**Technical Exchange**	**Comment**	**Path Forward**
		2000x). Because modeled effects fell within the range of variation included in total system performance assessment, it is asserted that effects farther from the drift would be smaller, based on an unverified assumption (CRWMS M&O, 2001d). This argument does not address chemical changes below the repository, which are likely to be more significant than changes above, because of interactions with the engineered barrier subsystem and waste materials. Even so, predicted changes in key geochemical parameters (pH and total carbon) in seepage water are large enough to have an effect on sorption coefficients. Without the details on how expert judgment was used to derive the Total System Performance Assessment–Site Recommendation sorption parameters, it is not clear how the effects of changes in the ambient chemistry system are incorporated into the transport calculations. The technical basis for this exclusion is not satisfactory.	
UZ3 SZ2	J-8	2.2.08.02.00 (Radionuclide Transport Occurs in a Carrier Plume in Geosphere) is excluded from the Total System Performance Assessment–Site Recommendation abstraction of radionuclide transport in the unsaturated zone on the basis of low consequence (CRWMS M&O, 2000d, 2001d). The key assumption (CRWMS M&O, 2001d; Assumption 11) is that results from the near-field thermal-hydrological-chemical coupled processes model (CRWMS M&O, 2000x) can be used to bound the effects of similar coupled processes on far-field flow and transport. This assumption has not yet been verified. Because the screening argument for this item is focused primarily on thermal effects on the chemistry of seepage water entering the emplacement drifts, it does not appear to include other potential effects (colloids, interactions with waste forms, and engineered barrier subsystem materials). Also, 2.1.09.01.00 (Properties of a Carrier Plume in the Engineered Barrier Subsystem) is included in the process model report (CRWMS M&O, 2001f, y), suggesting that radionuclide transport in a carrier plume should be included in transport beyond the engineered barrier subsystem. The arguments presented for exclusion of 2.2.08.02.00 (Radionuclide Transport Occurs in a Carrier Plume in Geosphere) (CRWMS M&O, 2001d) do not appear sufficient at this time.	This issue is addressed by existing agreements between DOE and NRC (Evolution of the Near-Field Environment Subissue 1 Agreement 4, and Subissue 4 Agreements 3 and 4, and Radionuclide Transport Subissue 1 Agreement 5). Features, Events, and Processes in UZ Flow and Transport, ANL–NBS–MD–000001, will be revised on completion of this work.
UZ2 UZ3	J-9	2.2.08.03.00 [Geochemical Interactions in Geosphere (Dissolution, Precipitation, Weathering) and Effects on Radionuclide Transport] is excluded (CRWMS M&O, 2000d, 2001d) from the Total System Performance Assessment–Site Recommendation abstraction of radionuclide transport in the unsaturated zone on the basis of low consequence. The key assumption (CRWMS M&O, 2001d; Assumption 11) is that results from the near-field thermal-hydrological-chemical coupled processes model (CRWMS M&O, 2000x) can be used to bound the effects of similar coupled processes on far-field flow and transport. This assumption has not yet been verified. Predicted mineralogical changes (CRWMS M&O, 2000x) in response to the thermal effects of the repository are small (calcite only). Predicted changes in porosity and	This issue is addressed by existing agreements between DOE and NRC (Evolution of the Near-Field Environment Subissue 1 Agreements 4 and 7, and Subissue 2 Agreement 6). Features, Events, and Processes in UZ Flow and Transport, ANL–NBS–MD–000001, will be revised on completion of this work.

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)			
Integrated Subissue	**Technical Exchange**	**Comment**	**Path Forward**
		permeability are also small. Transport through fractures is conservatively modeled in the Total System Performance Assessment–Site Recommendation, assuming no retardation. The screening argument, however, only addresses changes in seepage water chemistry. It does not address the possibility of reduced (or enhanced) matrix diffusion through precipitation and dissolution. Diffusion into the matrix and sorption on matrix minerals can be an important retardation mechanism. The effect of small-volume changes on fracture armoring and diffusion into the matrix may be important. The current screening arguments are not sufficient and will depend, in part, on the verification of Assumption 11 that far-field changes to radionuclide transport in the unsaturated zone will be less than the calculated near-field changes (CRWMS M&O, 2001d). Effects on flow are excluded based on low consequence. Problems with modeling of drift-scale coupled processes (CRWMS M&O, 2000x) used to support this screening argument have been raised by NRC. Current agreements from the Evolution of the Near-Field Environment Technical Exchange may provide additional technical basis for the screening argument.	
UZ3	J-10	2.2.08.06.00 (Complexation in Geosphere) is excluded. DOE included the effects of ambient condition complexation in the Total System Performance Assessment–Site Recommendation abstraction of radionuclide transport in the unsaturated zone, but has excluded future changes (CRWMS M&O, 2000d, 2001d). The effects of complexation are "implicitly included in the radionuclide sorption coefficients," but there is no clear technical basis regarding the effects of organics or other ligands provided in establishing the K_d distributions (CRWMS M&O 2001d). Experimental results reported in Triay, et al. (1997) that form much of the basis for the sorption coefficient distributions only address the effects of organics on neptunium and plutonium sorption. The analysis and model report (CRWMS M&O, 2000w) does not provide additional information on the effect of organics on other radionuclides. The current process models do not address the effects of complexation on transport parameters, and the exclusion of changes to complex formation does not have sufficient support. In addition, the screening argument refers to modeling results on repository effects on seepage chemistry, which may not be relevant to transport conditions below the repository (CRWMS M&O, 2001d).	This issue is addressed by existing agreements between DOE and NRC (Evolution of the Near-Field Environment Subissue 1 Agreement 4, and Subissue 4 Agreements 3 and 4, and Radionuclide Transport Subissue 1 Agreement 5). Features, Events, and Processes in UZ Flow and Transport, ANL–NBS–MD–000001, will be revised on completion of this work.
Dose2 Dose3	20	The Yucca Mountain Project Database (CRWMS, 2001g) (Revision 00 ICN 01) does not indicate that 2.2.08.07.00 (Radionuclide Solubility Limits in the Geosphere) is relevant to the biosphere. This item is relevant for limiting the quantity of radioactive material that can leach radionuclides out of the soil or tephra deposit in the biosphere compared with the quantity of radionuclides that would be predicted to leach out of the deposit using only leach rate limits.	DOE will add this item to Evaluation of the Applicability of Biosphere-Related Features, Events, and Processes, ANL–MGR–MD–000011, and present the DOE discussion in the screening argument.

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)			
Integrated Subissue	**Technical Exchange**	**Comment**	**Path Forward**
UZ3	J-11	2.2.08.07.00 (Radionuclide Solubility Limits in the Geosphere) is excluded from the Total System Performance Assessment–Site Recommendation abstraction of radionuclide transport in the unsaturated zone on the basis of low consequence (CRWMS M&O, 2000d, 2001d). The DOE screening argument assumes that radionuclide solubility limits in the geosphere may be different and indicates that radionuclide solubility limits in the geosphere are conservatively ignored with respect to solubility reduction in the far field (CRWMS M&O, 2000d). This argument makes valid points, but the possibility of increasing solubility limits should also be considered. Solubility limits in the geosphere will be determined by interaction between the contaminant plume and the host rock.	This issue is addressed by existing agreements between DOE and NRC (Evolution of the Near-Field Environment Subissue 4 Agreement 3). Features, Events, and Processes in UZ Flow and Transport, ANL–NBS–MD–000001, will be revised on completion of this work.
UZ3 SZ2	J-12	2.2.10.01.00 (Repository-Induced Thermal Effects in Geosphere) is excluded from the Total System Performance Assessment–Site Recommendation abstraction of radionuclide transport in the unsaturated zone on the basis of low consequence (CRWMS M&O, 2000d, 2001d). The screening argument is only partially supported by near-field thermal-chemical modeling for a limited number of hydrochemical constituents and minerals (CRWMS M&O, 2000x) and is not directly related to the effects on radionuclide transport. The technical basis for the screening is not sufficient at this time and future evaluation of the exclusion of 2.2.10.01.00 (Repository-Induced Thermal Effects in Geosphere) will depend, in part, on the verification of Assumption 11 that far-field changes to radionuclide transport in the unsaturated zone will be less than the calculated near-field changes (CRWMS M&O, 2001d).	This issue is addressed by existing agreements between DOE and NRC (Evolution of the Near-Field Environment Subissue 1 Agreement 4, and Subissue 4 Agreements 3 and 4, and Radionuclide Transport Subissue 1 Agreement 5). Features, Events, and Processes in UZ Flow and Transport, ANL–NBS–MD–000001, will be revised on completion of this work.
SZ1	13	2.2.10.02.00 (Thermal Convection Cell Develops in Saturated Zone) is screened as excluded on the basis of low consequence (CRWMS M&O, 2000f). DOE indicates that temperatures at the water table are expected to approach 80 °C [176 °F]. DOE further points out the resulting concern is that thermally driven water flow in the upper tuff aquifer could increase groundwater velocities relative to the system without heat sources. Additional justification for exclusion is necessary.	DOE agreed to provide clarification of the screening argument in Features, Events, and Processes in Saturated Zone Flow and Transport, ANL–NBS–MD–000002, to address the NRC comment.
UZ2 SZ1 SZ2	3	2.2.10.03.00 (Natural Geothermal Effects). It is stated that natural geothermal effects are included because the current geothermal gradient is addressed in the saturated zone flow and transport model (CRWMS M&O, 2001e). This discussion, however, does not address the potential for spatial and temporal variations in that gradient, which is part of the description of 2.2.10.03.00 (Natural Geothermal Effects). Resolution of this issue is necessary to address changes in the geothermal gradient in 2.2.10.13.00 [Density-Driven Groundwater Flow (Thermal)].	This issue is addressed by an existing DOE and NRC agreement (Unsaturated and Saturated Flow Under Isothermal Conditions Subissue 5, Agreement 13). Features, Events, and Processes in Saturated Zone Flow and Transport, ANL–NBS–MD–000002, will be updated, as necessary, to reflect the results of this existing agreement.
ENG2 ENG3 UZ2	70	2.2.10.04.00 (Thermal-Mechanical Alteration of Fractures Near Repository) is screened excluded on the basis of low consequence (CRWMS M&O, 2000h, 2001b). See discussion in 2.2.06.01.00 [Changes in Stress (Due to Thermal, Seismic, or Tectonic Effects), Change Porosity, and Permeability of Rock]. Heat	The thermal-mechanical effects on rock properties are addressed by an existing DOE and NRC agreement (Repository Design and Thermal-Mechanical Effects Subissue 3, Agreements 20 and 21). FEPs in Thermal Hydrology and Coupled Processes, ANL–NBS–MD–000004, will be revised on completion

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)			
Integrated Subissue	**Technical Exchange**	**Comment**	**Path Forward**
		from the waste causes thermal expansion of the surrounding rock, generating compressive stresses near the drifts and extensional stresses away from them. The zone of compression migrates with time.	of this work.
UZ2	67	2.2.10.05.00 (Thermal-Mechanical Alteration of Rocks Above and Below the Repository) is screened as excluded on the basis of low consequence (CRWMS M&O, 2001b). Thermal-mechanical compression at the repository produces tension-fracturing in the paintbrush nonwelded tuff and other units above the repository. These fractures alter unsaturated zone flow between the surface and the repository. Extreme fracturing may propagate to the surface, affecting infiltration. Thermal fracturing in rocks below the repository affects flow and radionuclide transport to the saturated zone.	DOE planned to analyze the effects of thermal-hydrological-mechanical coupled processes with regard to drainage in the pillars and flow in the vicinity of the drifts and thermal-hydrological/thermal-hydrological-chemical/thermal-hydrological-mechanical analyses to quantify uncertainties in the thermal seepage model. In addition, thermal-hydrological-mechanical continuum modeling will address thermal-mechanical effects in rocks above and below the repository at a mountain scale in an update to the Coupled Thermal-Hydrological-Mechanical Effects on Permeability Analysis and Model Report, ANL–NBS–HS–000037. DOE will clarify the screening arguments in the FEPs in Thermal Hydrology and Coupled Processes, ANL–NBS–MD–000004, on completion of this agreement.
ENG3 UZ2 UZ3 SZ1 SZ2 UZ3	 64 9	2.2.10.06.00 [Thermal-Chemical Alteration (Solubility Speciation, Phase Changes, and Precipitation/Dissolution)] [Near-Field Environment]: Screened as excluded on the basis of low consequence (CRWMS M&O, 2001b). Changes in the groundwater temperature in the farfield, if significant, may change the solubility and speciation of certain radionuclides. This change would have the effect of altering radionuclide transport processes. Relevant processes include volume effects associated with silica phase changes, precipitation and dissolution of fracture-filling minerals (including silica and calcite), and alteration of zeolites and other minerals to clays. [Saturated Zone]: Screened as excluded on the basis of low consequence (CRWMS M&O, 2001e) with reference to the screening argument for 2.2.07.10.00 (Condensation Zone Forms Around Drifts) in (CRWMS M&O, 2001d). The argument that repository thermal effects on saturated zone radionuclide transport will be minimal is based on a to-be-verified assumption (CRWMS M&O, 2001d). There is no explicit technical basis presented that rock alteration or temperature effects on geochemical properties and processes will negligibly affect saturated zone transport. In addition, it is asserted in CRWMS M&O (2001e) that any such effects would be within the bounds of uncertainty ranges established for transport properties such as K_d. However, the relevant analysis and model report (CRWMS M&O, 2000w) does not provide a clear technical basis that this is the case. The DOE current technical justification is considered inadequate. DOE should provide additional technical justification for exclusion. Same comment applies to 2.2.10.08.00 (Thermal-Chemical Alteration of the Saturated Zone). [Unsaturated Zone]: DOE has not provided the technical basis for excluding entrained colloids in the analysis of	[Near-Field Environment]: This issue is addressed by an existing agreement between DOE and NRC (Evolution of the Near-Field Environment Subissue 1, Agreement 3). FEPs in Thermal Hydrology and Coupled Processes, ANL–NBS–MD–000004, will be revised on completion of this work. [Saturated Zone]: DOE agreed to provide clarification of the screening argument in Features, Events, and Processes in SZ Flow and Transport, ANL–NBS–MD–000002, to address the NRC comment. [Unsaturated Zone]: At the Evolution of the Near-Field Environment Technical Exchange, DOE agreed to provide the technical basis for excluding entrained colloids in the analysis of 2.2.10.06.00 (Thermal-Chemical Alteration) or an alternative features, events, and processes (Evolution of the Near-Field Environment Subissue 1, Agreement 6). DOE will provide the technical basis for screening entrained colloids in the analysis of 2.2.10.06.00 (Thermal-Chemical Alteration) in a future revision of Features, Events, and Processes in SZ Flow and Transport, ANL–NBS–MD–000001, expected to be available in fiscal year 2002.

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)			
Integrated Subissue	**Technical Exchange**	**Comment**	**Path Forward**
		2.2.10.06.00 [Thermo-Chemical Alteration (Solubility Speciation, Phase Changes, and Precipitation/Dissolution)] or an alternative database entry (CRWMS M&O, 2001d). DOE has not considered possible entrainment of colloids and particulates in convecting/advecting boiling fluids or by otherwise vigorous water movement in the drift.	
UZ3	J-13	2.2.10.06.00 [Thermal-Chemical Alteration (Solubility, Speciation, Phase Changes, Precipitation/Dissolution)] is excluded from the Total System Performance Assessment–Site Recommendation abstraction of radionuclide transport in the unsaturated zone on the basis of low consequence (CRWMS M&O, 2000d, 2001d). Thermal effects on chemistry at the mountain scale are expected to be low, based on near-field coupled thermal-hydrological-chemical models that indicate the thermal effects of the repository result in only small changes in major hydrochemical constituents and limited changes in mineralogy, however, model results in the cited report (CRWMS M&O, 2000x) only consider a few components in hydrochemistry important to container life (e.g., pH, total carbon, and calcium). The model is limited to calcite precipitation/dissolution and addresses only seepage water chemistry. Thermal-chemical effects on transport beneath the repository, which could reflect the influence of the engineered barrier subsystem and waste form materials, are not considered. In addition, although the assumption that far-field changes are likely to be less than near-field changes is reasonable, it has not been verified (CRWMS M&O, 2001d). The technical basis is not sufficient at this time to demonstrate low consequence. The evaluation of this exclusion will depend in part on the verification of Assumption 11 that far-field changes to radionuclide transport in the unsaturated zone will be less than the calculated near-field changes (CRWMS M&O, 2001d).	This issue is addressed by existing agreements between DOE and NRC (Evolution of the Near-Field Environment Subissue 1 Agreement 4, and Subissue 4 Agreements 3 and 4, and Radionuclide Transport Subissue 1 Agreement 5). Features, Events, and Processes in UZ Flow and Transport, ANL–NBS–MD–000001, will be revised on completion of this work.
UZ2 UZ3	J-14	2.2.10.07.00 (Thermal-Chemical Alteration of the Calico Hills Unit) is excluded from the Total System Performance Assessment–Site Recommendation abstraction of radionuclide transport in the unsaturated zone on the basis of low consequence (CRWMS M&O, 2001d). The screening argument is based on the prediction of small changes in aqueous geochemistry and mineralogy in response to coupled thermal-hydrological-chemical processes in the near field (CRWMS M&O, 2000x). Thermal-chemical changes in the far field, including the Calico Hills unit, will be even less significant (CRWMS M&O, 2001d; Assumption 11). The screening argument indicates that temperatures in the zeolite-bearing Calico Hills unit, will not be high enough to cause significant zeolite alteration. Because the radionuclide transport abstraction assumes no retardation in fractures, this exclusion may be appropriate (however, see next paragraph). Again, final evaluation of this exclusion will depend, in part, on the verification of Assumption 11 that far-field changes to radionuclide transport in the unsaturated zone will be less than the calculated near-field changes (CRWMS M&O, 2001d). Alteration of the uppermost nonwelded layers below the repository could significantly reduce the fraction of	This issue is addressed by existing agreements between DOE and NRC (Evolution of the Near-Field Environment Subissue 1 Agreement 4, and Subissue 4 Agreements 3 and 4, and Radionuclide Transport Subissue 1 Agreement 5). Features, Events, and Processes in UZ Flow and Transport, ANL–NBS–MD–000001, will be revised on completion of this work. DOE also stated that alteration of vitric rock has not been addressed and will need to be included in the overall thermal-hydrological-chemical analyses.

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)			
Integrated Subissue	**Technical Exchange**	**Comment**	**Path Forward**
		matrix flow below the repository. Nonwelded vitric horizons, either basal Topopah Springs vitrophyre or the uppermost Calico Hills unit, cover nearly half the repository. In the southwestern portion of the repository footprint, the nonwelded, nonaltered tuffs lie as little as 45 m [147.64 ft] below the repository. The screening argument (CRWMS M&O, 2001d) includes the assertion that temperatures in the Calico Hills unit will remain below 70 °C [158 °F], which is not high enough to cause significant zeolite alteration. According to the cited reference, however, it appears temperatures can exceed 70 °C [158 °F] {up to 85 °C [185 °F]} is estimated from figures in the cited section of CRWMS M&O, 2000z} where the nonwelded, nonaltered tuff is closest to the repository.	
SZ1 SZ2	9	2.2.10.08.00 (Thermal-Chemical Alteration of the Saturated Zone). See comment on 2.2.10.06.00 [Thermal-Chemical alteration (solubility speciacion, phase changes, precipitation/dissolution)].	See comment on 2.2.10.06.00 [Thermal-Chemical Alteration (solubility speciacion, phase changes, precipitation/dissolution)].
UZ2 UZ3	J-15	2.2.10.09.00 (Thermal-Chemical Alteration of the Topopah Spring Basal Vitrophyre) is excluded from the Total System Performance Assessment–Site Recommendation abstraction of radionuclide transport in the unsaturated zone on the basis of low consequence (CRWMS M&O, 2000d, 2001d). The screening argument is based on predicting small changes in aqueous geochemistry and mineralogy in response to coupled thermal-hydrological-chemical processes in the near field (CRWMS M&O, 2000x). Thermal-chemical changes in the far field, including the Topopah Spring basal vitrophyre, are expected to be even less significant (CRWMS M&O, 2001d). Although the assumption that far-field changes are likely to be less than near-field changes (Assumption 11) is reasonable, this assumption has not been verified (CRWMS M&O, 2001d). It is important to note that the near-field analyses (CRWMS M&O, 2000x) focus on seepage chemistry and how it might affect container life, rather than considering thermal effects on radionuclide transport. The technical basis is not sufficient to demonstrate low consequence to radionuclide transport. Because the Total System Performance Assessment–Site Recommendation radionuclide transport abstraction assumes no retardation in fractures, this exclusion may be appropriate. Final evaluation of this exclusion will depend on verification of Assumption 11 that far-field changes to radionuclide transport in the unsaturated zone will be less than the calculated near-field changes (CRWMS M&O, 2001d). Alteration of the uppermost nonwelded layers below the repository could significantly reduce the fraction of matrix flow below the repository. Nonwelded vitric horizons, either basal Topopah Spring vitrophyre or the uppermost Calico Hills unit, cover nearly half the repository. In the southwestern portion of the repository footprint, the nonwelded, nonaltered tuffs lie as little as 45 m [147.64 ft] below the repository. The screening argument for 2.2.10.07.00 (CRWMS M&O, 2001d) includes the assertion that temperatures in the Calico Hills unit will remain below 70°C [158 °F] which is not high enough to cause significant zeolite alteration. According to the cited reference, however, it	This issue is addressed by existing agreements between DOE and NRC (Evolution of the Near-Field Environment Subissue 1 Agreement 4, and Subissue 4 Agreements 3 and 4, and Radionuclide Transport Subissue 1 Agreement 5). Features, Events, and Processes in UZ Flow and Transport, ANL–NBS–MD–000001, will be revised on completion of this work.

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)			
Integrated Subissue	**Technical Exchange**	**Comment**	**Path Forward**
		appears temperatures can exceed 70°C [158 °F] {up to 85°C [185 °F] is estimated from figures in the cited section of CRWMS M&O (2000z)} where the nonwelded, nonaltered tuff is closest to the repository. Temperatures would be higher in the overlying Topopah Spring basal vitrophyre than in Calico Hills.	
UZ1 UZ2	61	2.2.10.12.00 (Geosphere Dryout Due to Waste Heat). It is necessary to develop a screening argument for this item as part of the scope of the analysis and model report (CRWMS M&O, 2001d). Elevated thermal effects on shallow infiltration from changes in soil water content were not addressed for 2.2.10.12.00 (Geosphere Dryout Due to Waste Heat). The DOE study of a natural thermal gradient on Yucca Mountain addresses this item (CRWMS M&O, 1998b). 2.2.10.12.00 (Geosphere Dryout Due to Waste Heat) is screened as included in CRWMS M&O (2001b) for issues related to the near-field environment, but does not address the effects on infiltration.	DOE agreed to provide the technical basis for the screening argument in Features, Events, and Processes in UZ Flow and Transport, ANL–NBS–MD–000001, to address the NRC comment.
UZ2 SZ1 SZ2	12	2.2.10.13.00 [Density-Driven Groundwater Flow (Thermal)]. The analysis and model report (CRWMS M&O, 2001e) addresses this item in two parts: repository-induced effects (excluded, low consequence) and natural geothermal effects (included). Exclusion of repository effects on flow based on the DOE analyses is accepted. Natural effects are included only to the extent that the natural geothermal gradient is applied in the saturated zone flow and transport model. However, changes in thermal gradients are excluded on the basis of low consequence, with reference to 1.2.06.00.00 (Hydrothermal Activity) and 1.2.10.02.00 (Hydrologic Response to Igneous Activity) (CRWMS M&O, 2001e). A clear technical basis is not provided for these items that all possible changes in thermal gradients will be localized. The screening argument for 1.2.06.00.00 (Hydrothermal Activity) focuses on geochemical effects (see separate entry), whereas 1.2.10.02.00 (Hydrologic Response to Igneous Activity) is focused on highly localized igneous intrusions. How these arguments apply to 2.2.10.13.00 [Density-Driven Groundwater Flow (Thermal)] is not entirely clear.	This issue is addressed by an existing DOE and NRC agreement (Unsaturated and Saturated Flow Under Isothermal Conditions Subissue 5, Agreement 13). Features, Events, and Processes in SZ Flow and Transport, ANL–NBS–MD–000002, will be updated to clarify the screening argument and to reflect the results of this existing agreement.
UZ2	J-21	2.2.11.02.00 (Gas Pressure Effects) is excluded based on low consequence and low probability (CRWMS M&O, 2001d). Consistency is needed in the screening arguments. Buildup of water vapor pressure within rock matrix blocks from waste heat has not been considered. Gas pressure can build up within matrix blocks that have low permeability. This condition can increase the boiling point and keep water in the liquid phase at higher temperatures. Flashing to vapor as liquid water leaves the matrix block can result in mineral deposition that can later affect flow pathways.	This issue is addressed by existing agreements between DOE and NRC (Evolution of the Near-Field Environment Subissue 1 Agreements 5 and 7, and Subissue 4 Agreement 3). Features, Events, and Processes in UZ Flow and Transport, ANL–NBS–MD–000001, will be revised on completion of this work.
SZ1 SZ2 Dose1 Dose2 Dose3	10	2.3.11.04.00 (Groundwater Discharge to Surface) is excluded on the basis of low consequence (CRWMS M&O, 2001e). Modeling shows that spring discharge within the 20-km [12.4-mi] radius is not likely, yet past discharges occurred within the 20-km [12.4-mi] radius (e.g., paleospring deposits at 9S and 1S). See discussion of 1.3.07.02.00 (Water Table Rise). Any screening argument that spring discharges are outside the proposed compliance area is insufficient.	DOE agreed to provide clarification of the screening argument in Features, Events, and Processes in SZ Flow and Transport, ANL–NBS–MD–000002, to address the NRC comment.

Table B–1. NRC Comments on Features, Events, and Processes and Path Forward for Resolution Including, DOE and NRC Agreements (continued)			
Integrated Subissue	**Technical Exchange**	**Comment**	**Path Forward**
		Additional technical justification is required to fully exclude 2.3.11.04.00 (Groundwater Discharge to Surface).	
Dose3 Dose2	21	2.3.13.01.00 (Biosphere Characteristics) screening argument indicates the Yucca Mountain region lacks permanent surface water (CRWMS M&O, 2001a). It is not clear this statement is consistent with the geologic record of past climate change in the area.	DOE agreed to provide clarification of the screening argument in Evaluation of the Applicability of Biosphere-Related Features, Events, and Processes (FEP), ANL–MGR–MD–000011, to address the NRC comment.
Dose3	24	2.3.13.02.00 (Biosphere Transport) contains only two secondary entries related to surface water, gas, and biogeochemical transport processes (CRWMS M&O, 2001a). The Yucca Mountain Project feature, event, and process description and the originator description are different and question whether the focus is transport processes, alterations during transport, or both.	DOE agreed to clarify the description of the primary features, events, and processes in Evaluation of the Applicability of Biosphere-Related Features, Events, and Processes (FEP), ANL–MGR–MD–000011, to address the NRC comment.
Dose3	25	2.4.07.00.00 (Dwellings) includes a secondary entry, household cooling, which has an inappropriate screening argument (CRWMS M&O, 2001a). The screening argument indicates that because use of an evaporative cooler would only increase inhalation and direct exposure pathways, and these pathways are only minor contributors to the current dose conversion factors, the use of evaporative coolers can be screened. However, the direct exposure and inhalation dose from evaporative coolers is the result of significantly different processes than the direct exposure and inhalation dose from radionuclides deposited on soils and, hence, could have a more significant dose impact.	DOE agreed to provide the technical basis for the screening argument in Evaluation of the Applicability of Biosphere-Related Features, Events, and Processes (FEP), ANL–MGR–MD–000011, to address the NRC comment.
Dose3 Dose2 Direct2	26	The analysis and model report (CRWMS M&O, 2001a) states that 3.3.08.00.00 (Radon and Daughter Exposure) is screened as excluded on the basis the parent radionuclide (Th-230) will not reach the critical group in 10,000 years in the basecase scenario (CRWMS M&O, 2000aa, 2001a). This rationale, however, does not apply to the direct release scenario, where transport times are much shorter.	DOE agreed to provide the technical basis for the screening argument in Evaluation of the Applicability of Biosphere-Related Features, Events, and Processes (FEP), ANL–MGR–MD–000011, to address the NRC comment.
* Brekke T.L., E.J. Cording, J. Daemen, R.D. Hart, J.A. Hudson, P.K. Kaiser, and S. Pelizza. “Panel Report on the Drift Stability Workshop, Las Vegas, Nevada, 9–11 December, 1998.” Yucca Mountain Site Characterization Project. 1999.			

REFERENCES

American Society for Metals International. "Properties and Selection: Nonferrous Alloys and Special-Purpose Materials, Specific Metals and Alloys." Vol. 2, *Metals Handbook*. 10th Edition. Metals Park, Ohio: American Society for Metals. 1990.

Ankem, S., C.A. Greene, and S. Singh. "Time Dependent Twinning During Ambient Temperature Creep of α Ti-Mn Alloy." *Scripta Metallurgica et Materialia*. Vol. 30, No. 6. pp. 803–808. 1994.

CRWMS M&O. "Waste Isolation Evaluation: Tracers, Fluids, and Materials, and Excavation Methods for Use in the Package 2C Exploratory Studies Facility Construction." BABE00000–01717–2200–00007. Revision 04. Las Vegas, Nevada: CRWMS M&O. 1995.

———. "Repository Ground Support Analysis for Viability Assessment." BCAA00000–01717–0200–0004. Revision 01. Las Vegas, Nevada: CRWMS M&O. 1998a.

———. "Final Report: Plant and Soil Related Processes Along a Natural Thermal Gradient at Yucca Mountain, Nevada." B00000000–01717–5705–00109. Revision 00. Las Vegas, Nevada: CRWMS M&O. 1998b.

———. "Features, Events, and Processes: Screening for Disruptive Events." ANL–WIS–MD–000005. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000a.

———. "Input Request for Seismic Evaluations of Waste Packages and Emplacement Pallets." Input Transmittal 00230.T. Las Vegas, Nevada: CRWMS M&O. 2000b.

———. "Rock Fall on Drip Shield." CAL–EDS–ME–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000c.

———. "Unsaturated Zone Flow and Transport Model Process Model Report." TDR–NBS–HS–000002. Revision 00 ICN 02. Las Vegas, Nevada: CRWMS M&O. 2000d.

———. "Natural Analogs for the Unsaturated Zone." ANL–NBS–HS–000007. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000e.

———. "Features, Events, and Processes in SZ Flow and Transport." ANL–NBS–MD–000002. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000f.

———. "Uncertainty Distribution for Stochastic Parameters." ANL–NBS–MD–000011. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000g.

———. "Features, Events, and Processes in UZ Flow and Transport." ANL–NBS–MD–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000h.

———. "Miscellaneous Waste Form FEPs." ANL–WIS–MD–000009. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000i.

———. “Clad Degradation—FEPs Screening Arguments.” ANL–WIS–MD–000008. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000j.

———. “Clad Degradation—Local Corrosion of Zirconium and Its Alloys Under Repository Conditions.” ANL–EBS–MD–000012. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000k.

———. “Total System Performance Assessment for the Site Recommendation.” TDR–WIS–PA–000001. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000l.

———. “Repository Safety Strategy: Plan to Prepare the Safety Case to Support Yucca Mountain Site Recommendation and Licensing Considerations.” TDR–WIS–RL–000001. Revision 04 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000m.

———. “Drift Degradation Analysis AMR.” ANL–EBS–MD–000027. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000n.

———. “EBS Radionuclide Transport Abstraction AMR.” ANL–WIS–PA–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000o.

———. “EBS Physical and Chemical Environmental Model AMR.” ANL–EBS–MD–000033. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000p.

———. “Colloid-Associated Concentration Limits: Abstraction and Summary.” ANL–WIS–MD–000012. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000q.

———. “Waste Package Operations Fabrication Process Report.” TDR–EBS–ND–000003. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000r.

———. “Design Analysis for the Ex-Container Components.” ANL–XCS–ME–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000s.

———. “Features, Events, and Processes: System-Level and Criticality.” ANL–WIS–MD–000019. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000t.

———. “Probability of Criticality in 10,000 Years.” CAL–EBS–NU–000014. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000u.

———. “AMR Fault Displacement Effects on Transport in the Unsaturated Zone.” ANL–NBS–HS–000020. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000v.

———. “Unsaturated Zone and Saturated Zone Transport Properties.” ANL–NBS–HS–000019. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000w.

———. “Drift-Scale Coupled Processes (DST and THC Seepage) Models.” MDL–NBS–HS–000001. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000x.

———. “Engineered Barrier System Degradation, Flow, and Transport Process Model Report.” TDR–EBS–MD–000006. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000y.

———. “Mountain Scale Coupled Processes.” MDL–NBS–HS–000007. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000z.

———. “Disruptive Event Biosphere Dose Conversion Factor Analysis.” ANL–MGR–MD–000003. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000aa.

———. “Evaluation of the Applicability of Biosphere-Related Features, Events, and Processes (FEP)”. ANL–MGR–MD–000011. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2001a.

———. “FEPs Thermal Hydrology and Coupled Processes.” ANL–NBS–MD–000004. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2001b.

———. “FEPs Screening of Processes and Issues in Drip Shield and Waste Package Degradation.” ANL–EBS–PA–000002. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2001c.

———. “Features, Events, and Processes in UZ Flow and Transport.” ANL–NBS–MD–000001 Revision 01. Las Vegas, Nevada: CRWMS M&O. 2001d.

———. “Features, Events, and Processes in SZ Flow and Transport.” ANL–NBS–MD–000002. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2001e.

———. “EBS FEPs/Degradation Modes Abstraction.” ANL–WIS–PA–000002. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2001f.

———. “Yucca Mountain FEP Database.” TDR–WIS–MD–000003. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2001g.

Hsiung S.M. and G.H. Shi. 2001. “Simulation of Earthquake Effects on Underground Excavations Using Discontinuous Deformation Analysis (DDA).” Rock Mechanics in the National Interest. Vol. II. Lisse, The Netherlands: A.A. Balkema Publishers. pp. 1,413–1,420. 2001.

Matyskiela, W. “Silica Redistribution and Hydrologic Changes in Heated Fractured Tuff.” *Geology*. Vol. 25. pp. 1,115–1,118. 1997.

NRC. “Issue Resolution Status Report, Key Technical Issue: Evolution of the Near-Field Environment.” Revision 3. Washington, DC: NRC. 2000.

———. “Issue Resolution Status Report, Key Technical Issue: Container Life and Source Term.” Revision 3. Washington, DC: NRC. 2001.

Ofoegbu, G.I. “Thermal-Mechanical Effects on Long-Term Hydrological Properties at the Proposed Yucca Mountain Nuclear Waste Repository.” CNWRA 2000-03. San Antonio, Texas: CNWRA. 2000.

Ratcliff, C.D., J.W. Geissman, F.V. Perry, B.M. Crowe, and P.K. Zeitler. “Paleomagnetic Record of a Geomagnetic Field Reversal from Late Miocene Mafic Intrusions.” *Science*. Vol. 266. pp. 412–416. 1994.

Rebak, R.B., T.S.E. Summers, and R.M. Carranza. “Mechanical Properties, Microstructure, and Corrosion Performance of C–22 Alloy Aged at 260 °C to 800 °C.” Scientific Basis for Nuclear Waste Management XXIII. Warrendale, Pennsylvania: Materials Research Society. pp. 109-114. 2001.

Triay, I.R., A. Meijer, J.L. Conca, K.S. Kung, R.S. Rundberg, and E.A. Strietelmeier. “Summary and Synthesis Report on Radionuclide Retardation for the Yucca Mountain Site Characterization Project.” LA–13262–MS. Los Alamos, New Mexico: Los Alamos National Laboratory, Chemical Science and Technology Division. 1997.

GLOSSARY

This Glossary is provided for information and is not exhaustive.

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 variability 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 suitable form for use in a total system performance assessment. Model abstraction is intended to maximize the use of limited computational resources while allowing a sufficient range of sensitivity and uncertainty analyses.

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. For example, advection in combination with dispersion controls flux into and out of the elemental volumes of the flow domain in groundwater transport models.

air mass fraction: The mass of air divided by the total mass of gas (typically air plus water vapor) in the gas phase. This expression gives a measure of the “dryness” of the gas phase, which is important in waste package corrosion models.

Alloy 22: A nickel-base corrosion resistant alloy containing approximately 22 weight percent chromium, 13 weight percent molybdenum, and 3 weight percent tungsten as major alloying elements and that may be used as the outer container material in a waste package design (see *outer barrier*).

alluvium: Detrital deposits made by streams on river beds, flood plains, and alluvial fans; especially a deposit of silt or silty clay laid down during time of flood. The term applies to stream deposits of recent time. It does not include subaqueous sediments of seas and lakes.

alternative: Plausible interpretations or designs based on assumptions other than those used in the base case that could also fit or be applicable, based on 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 repository’s probable behavior.

ambient: Undisturbed, natural conditions such as ambient temperature caused by climate or natural subsurface thermal gradients, and other surrounding conditions.

anisotropy: The condition that physical properties vary when measured in different directions or along different axes. For example, in layered rock the permeability is often greater within the horizontal layers than across the horizontal layers.

annual frequency: The number of occurrences of an event expected in one year.

aqueous: Pertaining to water, such as aqueous phase, aqueous species, or aqueous transport.

aquifer: A subsurface, saturated rock unit (formation, group of formations, or part of a formation) of sufficient permeability to transmit groundwater and yield water of sufficient quality and quantity for an intended beneficial use.

ash: Bits of volcanic rock that would be broken-up during an eruption to less than 2 mm [0.08 inches] in diameter.

basalt: A type of igneous rock that forms black, rubbly lavas and black-to-red tephras of the type commonly used as lava rocks for barbecues.

borosilicate glass: A predominantly noncrystalline, relatively homogenous glass formed by melting silica and boric oxide together with other constituents such as alkali oxides. A high-level radioactive waste matrix 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, often at the geometric edge of the model, for a given variable. An example is using a specified groundwater flux from net infiltration as a boundary condition for an unsaturated flow model.

bound: An analysis or selection of parameter values that yields pessimistic results, such that any actual result is certain to be no worse or could be worse 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 aqueous environment and eventually permits 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.

calibration: (1)The process of comparing the conditions, processes, and parameter values used in a model against actual data points or interpolations (e.g., contour maps) from measurements at or close to the site to ensure that the model is compatible with reality, to the extent feasible. (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: A cylindrical metal receptacle that facilitates handling, transportation, storage, and/or disposal of high-level radioactive waste. It may serve as (1) a pour mold and container for vitrified high-level radioactive waste or (2) a container for loose or damaged fuel rods, non-fuel components and assemblies, and other debris containing radionuclides.

carbon steel: A steel made of carbon up to about 2 weight percent and only residual quantities of other elements. Carbon steel is a tough but ductile and malleable material used as baskets to maintain the spent fuel assemblies in fixed positions in the current waste package design.

Category 1 event sequences: Those event sequences that are expected to occur one or more times before permanent closure of a geologic repository.

Category 2 event sequences: Event sequences other than Category 1 event sequences that have at least one chance in 10,000 of occurring before permanent closure.

Center for Nuclear Waste Regulatory Analyses: A Federally funded research and development center in San Antonio, Texas, sponsored by the U.S. Nuclear Regulatory Commission, to provide the U.S. Nuclear Regulatory Commission with technical assistance for the repository program.

chain reaction: A continuing series of nuclear fission events that takes place within the fuel of a nuclear reactor. Neutrons produced by a split nucleus collide with and split other nuclei causing a chain of fission events.

cladding: The metal outer sheath of a fuel rod generally made of a zirconium alloy, and in the early nuclear power reactors of stainless steel, intended to protect the uranium dioxide pellets, which are the nuclear fuel, from dissolution by exposure to high temperature water under operating conditions in a reactor.

climate: Weather conditions including temperature, wind velocity, precipitation, and other factors, that prevail in a region.

climate states: Representations of climate conditions.

code (computer): The set of commands used to solve a mathematical model on a computer.

colloid: As applied to radionuclide migration, a colloidal system is a group of large molecules or small particles, having at least one dimension with the size range of 10^{-9} to 10^{-6} meters that are suspended in a solvent. Naturally occurring colloids in groundwater arise from clay minerals such as smectites and illites. Colloids that are transported in groundwater can be filtered out of the water in small pore spaces or very narrow fractures because of the large size of the colloids.

Colloid-Facilitated, Radionuclide Transport Model: A model that represents the enhanced transport of radionuclides by particles that are colloids.

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.

common cause failure: Two or more failures that result from a single event or circumstance.

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.

consequence: A measurable outcome of an event or process that, when combined with the probability of occurrence, gives 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.

continuum model: A model that represents fluid flow through numerous individual fractures and matrix blocks by approximating it as continuous flow fields.

corrosion: The deterioration of a material, usually a metal, as a result of a chemical or electrochemical reaction with its environment.

corrosion model: A theoretical representation of a corrosion process based on the application of a combination of fundamental electrochemical (chemical) and thermodynamic principles (or laws) with empirical parameters resulting from experiments, field measurements, or data obtained through industrial experience. Models can describe the penetration of a pit or a crack through a container wall as a function of time.

corrosion resistant alloy: An alloy that exhibits extremely high resistance to general or uniform corrosion in a given environment as a result of the formation of a protective film on its surface. Alloy 22, and other similar nickel-chromium-molybdenum alloys, are considered corrosion resistant alloys because they are extremely resistant to general corrosion in severe aqueous environments (e.g., high temperature brines containing acidic sulfur species).

coupling: The ability to assemble separate analyses or parameters in a performance assessment so that information can be passed among them to develop an overall analysis of system performance.

crevice corrosion: Localized corrosion of a metal surface at, or immediately adjacent to, an area that is shielded from full exposure to the environment because of close proximity between the metal and the surface of another material.

critical event: See *criticality*.

criticality: (1) A condition that would require the original waste form, which is part of the waste package, to be exposed to degradation, followed by conditions that would allow concentration of sufficient nuclear fuel, the presence of neutron moderators, the absence of neutron absorbers, and favorable geometry. (2) The condition in which nuclear fuel 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.

criticality accident: The release of energy as a result of accidental production of a self-sustaining or divergent neutron chain reaction.

data: Facts or figures measured or derived from site characteristics or standard references from which conclusions may be drawn. Parameters that have been derived from raw data are sometimes, themselves, considered to be data.

U.S. Department of Energy: A Cabinet-level agency of the U.S. federal government charged with the responsibilities of energy security, national security, and environmental quality.

design concept: An idea of how to design and operate the above-ground and below-ground portions of a repository.

diffusion: (1) The spreading or dissemination of a substance caused by concentration gradients. (2) The gradual mixing of the molecules of two or more substances because of random thermal motion.

diffusive transport: Movement of solutes because of their concentration gradient. The process in which substances carried in groundwater move through the subsurface by means of diffusion because of a concentration gradient.

dike: A tabular body of igneous rock that cuts across the structure of adjacent rocks or cuts massive rocks.

dimensionality: Modeling in one, two, or three dimensions.

direct exposure: The manner in which an individual receives dose from being in close proximity to a source of radiation. Direct exposures present an external dose pathway.

dispersion (hydrodynamic dispersion): (1) The tendency of a solute (substance dissolved in groundwater) to spread out from the path it is expected to follow if only the bulk motion of the flowing fluid were to move it. The tortuous path the solute follows through openings (pores and fractures) causes part of the dispersion effect in the rock. (2) The macroscopic outcome of the actual movement of individual solute particles through a porous medium. Dispersion causes dilution of solutes, including radionuclides, in groundwater, and is usually an important mechanism for spreading contaminants in low flow velocities.

disposal container: A cylindrical metal receptacle designed to contain spent nuclear fuel and high-level radioactive waste that will become an integral part of the waste package when loaded with spent nuclear fuel or high-level radioactive waste. In the current waste package design, the inner container will have spacing structures or baskets to maintain fuel assemblies, shielding components, and neutron absorbing materials in position to control the possibility of criticality.

disruptive event: An unexpected event that, in the case of the potential repository, includes volcanic activity, seismic activity, and nuclear criticality. Disruptive events have two possible effects: (1) direct release of radioactivity to the surface, or (2) alteration of the nominal behavior of the system. For the purposes of screening features, events, and processes for the total system performance assessment, a disruptive event is defined as an event that has a significant effect on the expected annual dose and that has a probability of occurrence during the 10,000-year period of performance less than 1.0, but greater than a cutoff of 0.0001.

disruptive event scenario class: The scenario, or set of related scenarios, that describes the behavior of the system if perturbed by disruptive events. The disruptive scenarios contain all disruptive features, events, and processes that have been retained for analysis.

dissolution: (1) Change from a solid to a liquid state. (2) Dissolving a substance in a solvent.

distribution: The overall scatter of values for a set of observed data. A term used synonymously with frequency distribution or probability distribution function. Distributions have structures that are the probability that a given value occurs in the set.

drift: From mining terminology, a horizontal underground passage. The nearly horizontal underground passageways from the shaft(s) to the alcoves and rooms. Drifts include excavations for emplacement (emplacement drifts) and access (access mains).

drift scale: The scale of an emplacement drift, or approximately 5 meters in diameter.

Drift-Scale Heater Test: A test being conducted in the Exploratory Studies Facility to investigate thermal-hydrologic, thermal-chemical, and thermal-mechanical processes.

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.

edge effects: Conditions at the edges of the potential repository that are cooler and wetter because heat dissipates more quickly there than at the center of the repository.

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 experience or experiment rather than on an understanding of the fundamental processes as related to the laws of nature.

emplacement drift: See *drift*.

enrichment: The act of increasing the concentration of ^{235}U from its value in natural uranium. The enrichment (typically reported in atom percent) is a characteristic of nuclear fuel.

equilibrium: The state of a chemical system in which the phases do not undergo any spontaneous change in properties or proportions with time; a dynamic balance.

events: (1) Occurrences that have a specific starting time and, usually, a duration shorter than the time being simulated in a model. (2) uncertain occurrences that take place within a short time relative to the time frame of the model. For the purposes of screening features, events, and processes for the total system performance assessment, an event is defined to be a natural or human-caused phenomenon that has a potential to affect disposal system performance and that occurs during an interval that is short compared with the period of performance.

event tree: A modeling tool that illustrates the logical sequence of events that follow an initiating event.

expert elicitation: A formal process through which expert judgment is obtained.

Exploratory Studies Facility: An underground laboratory at Yucca Mountain that includes a 7.9-kilometer [4.9-mile] main loop (tunnel); a 2.8-kilometer [1.75-mile] cross-drift; and a research alcove system constructed for performing underground studies during site characterization. The data collected will contribute toward determining the suitability of the Yucca Mountain site for a repository. Some or all of the Exploratory Studies Facility may eventually be incorporated into the potential repository.

fault (geologic): A planar or gently curved fracture across which there has been displacement parallel to the fracture surface.

fault tree: A graphical logic model that depicts the combinations of events that result in the occurrence of an undesired event.

features: Physical, chemical, thermal, or temporal characteristics of the site or potential repository system. For the purposes of screening features, events, and processes for the total system performance assessment, a feature is defined to be an object, structure, or condition that has a potential to affect disposal system performance.

ferritic steel: A subclass of carbon steels characterized by a relatively low strength but good ductility as a result of the ferrite microstructure. A type of ferritic steel, mild steel, or low-carbon steel containing up to about 0.1 weight percent carbon is the metallic material most commonly used for construction purposes.

film flow: Movement of water as a film along a surface such as a fracture plane.

finite element analysis: A commonly used numerical method for solving mechanical deformation problems. A technique in which algebraic equations are used to approximate the partial differential equations that comprise mathematical models to produce a form of the problem that can be solved on a computer. For this type of approximation, the area being modeled is formed into a grid with irregularly shaped blocks. This method provides an advantage in handling irregularly shaped boundaries, internal features such as faults, and surfaces of engineered materials. Values for parameters are frequently calculated at nodes for convenience, but are defined everywhere in the blocks by means of interpolation functions.

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 (including radionuclides) would follow in a given groundwater velocity field, governed principally by the hydraulic gradient.

fracture: A planar discontinuity in rock along which loss of cohesion has occurred. It is often caused by the stresses that cause folding and faulting. A fracture along which there has been displacement of the sides relative to one another is called a fault. A fracture along which no appreciable movement has occurred is called a joint. Fractures may act as fast paths for groundwater movement.

fracture aperture: The space that separates the sides of a fracture, and the measured width of the space separating the sides of a fracture.

fracture permeability: The capacity of a rock to transmit fluid that is related to fractures in the rock.

frequency: The number of occurrences of an observed or predicted event during a specific time period.

galvanic: Pertains to an electrochemical process in which two dissimilar electronic conductors are in contact with each other and with an electrolyte, or in which two similar electronic conductors are in contact with each other and with dissimilar electrolytes.

galvanic corrosion: Accelerated corrosion of a metal resulting from electrical contact with a more noble metal or non metallic conductor in a corrosive electrolyte.

geochemical: The distribution and amounts of the chemical elements in minerals, ores, rocks, soils, water, and the atmosphere; and the movement of the elements in nature on the basis of their properties.

geologic-framework model: A digital, scaled, geometrically congruent , three-dimensional model of the geologic system.

groundwater: Water contained in pores or fractures in either the unsaturated or saturated zones below ground level.

half-life: The time required for a radioactive substance to lose have 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 such as porosity, which is an attribute of an entity of interest such as the tuff rock containing the potential repository, varies over the space an entity occupies, such as the area around the repository, or with the passage of time.

high-level radioactive waste glass: A waste form produced by melting a mixture of high-level radioactive waste and components of borosilicate glass at a high temperature (approximately 1,100 degrees centigrade).

hydrologic: Pertaining to the properties, distribution, and circulation of water on the surface of the land, in the soil and underlying rocks, and in the atmosphere.

igneous: (1) A type of rock that has formed from a molten, or partially molten, material. (2) A type of activity related to the formation and movement of molten rock either in the subsurface (intrusive) or on the surface (volcanic).

infiltration: The process of water entering the soil at the ground surface. Infiltration becomes percolation when water has moved below the depth at which it can be removed (to return to the atmosphere) by evaporation or transpiration. See *net infiltration*.

inner barrier: The inner container in the current design of the waste package. Type 316NG stainless steel is the DOE preferred material of construction.

invert: A constructed surface that would provide a level drift floor and enable transport and support of the waste packages.

isothermal: Having a constant temperature.

license application: An application, to the U.S. Nuclear Regulatory Commission for a license to construct and operate a repository.

localized corrosion: Corrosion at discrete sites (e.g., pitting and crevice corrosion).

magma: Molten or partially molten rock that is naturally occurring and is generated within the earth. Magma may contain crystals along with dissolved gasses.

Mathematical Model: A mathematical description of a conceptual model.

matrix: Tuff rock material and its pore space exclusive of fractures. As applied to Yucca Mountain tuff, the ground mass of an igneous rock that contains larger crystals.

matrix diffusion: As used in the Total System Performance Assessment for the Site Recommendation conceptual models, the process by which molecular or ionic solutes, such as radionuclides in groundwater, move from areas of higher concentration to areas of lower concentration. This movement is through the pore spaces of the rock material as opposed to movement through the fractures.

matrix permeability: The capability of the matrix to transmit fluid.

mean (arithmetic): For a statistical data set, the sum of the values divided by the number of items in the set. The arithmetic average.

mechanical disruption: Damage to the drip shield or waste package because of external forces.

median: A value such that one-half of the observations are less than that value and one-half are greater than the value.

meteorology: The study of climatic conditions such as precipitation, wind, temperature, and relative humidity.

microbe: An organism too small to be viewed with the unaided eye. Examples of microbes are bacteria, protozoa, and some fungi and algae.

microbial influenced corrosion: Deterioration of metals as a result of the metabolic activity of microorganisms.

migration: Radionuclide movement from one location to another within the engineered barrier system or the environment.

mineral model: A description of the kinds and relative abundances of minerals that is used to approximate the true mineralogical system.

mineralogical: Of or relating to the chemical and physical properties of minerals, their occurrence, and their classification.

model: A depiction of a system, phenomenon, or process, including any hypotheses required to describe the system or explain the phenomenon or process.

near field: The area and conditions within the potential repository including the drifts and waste packages and the rock immediately surrounding the drifts. The region around the potential repository where the natural hydrogeologic system has been significantly impacted by the excavation of the repository and the emplacement of waste.

net infiltration: The amount of infiltration that escapes the zone of evapotranspiration, which is generally the zone below the zone of plant roots. See *infiltration*.

nominal behavior: (1) Expected behavior of the system as perturbed only by the presence of the potential repository. (2) Behavior of the system in the absence of disruptive events.

nominal features, events, and processes: Those features, events, and processes expected, given the site conditions as described from current site characterization information.

nominal scenario class: The scenario, or set of related scenarios, that describes the expected or nominal behavior of the system as perturbed only by the presence of the potential repository. The nominal scenarios contain all expected features, events, and processes that have been retained for analysis.

nuclear criticality safety: Protection against the consequences of a criticality accident, preferably by prevention of the accident.

U.S. Nuclear Regulatory Commission: An independent agency, established by the U.S. Congress under the Energy Reorganization Act of 1974, to ensure adequate protection of the public health and safety, the common defense and security, and the environment, in the use of nuclear materials in the United States. The U.S. Nuclear Regulatory Commission scope of responsibility includes regulation of the transport, storage, and disposal of nuclear materials and waste.

Nuclear Waste Policy Act (42 U.S.C. 10101 et seq.): The Federal statute enacted in 1982 that established the Office of Civilian Radioactive Waste Management and defined its mission to develop a federal system for the management, and geologic disposal, of commercial spent nuclear fuel and other high-level radioactive wastes. The Act also: (1) specified other federal responsibilities for nuclear waste management; (2) established the Nuclear Waste Fund to cover the cost of geologic disposal; (3) authorized interim storage under certain circumstances; and (4) defined interactions between federal agencies and the states, local governments, and Indian tribes. The act was substantially amended in 1987.

Nuclear Waste Policy Amendments Act of 1987: Legislation that amended the Nuclear Waste Policy Act to: (1) limit repository site characterization activities to Yucca Mountain, Nevada; (2) establish the Office of the Nuclear Waste Negotiator to seek a state or Indian tribe willing to host a repository or monitored retrievable storage facility; (3) create the Nuclear Waste Technical Review Board; and (4) increase state and local government participation in the waste management program.

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.

Office of Civilian Radioactive Waste Management: A U.S. Department of Energy office created by the Nuclear Waste Policy Act of 1982 to implement the responsibilities assigned by the Act.

outer barrier: The outer container in the current design of the waste package. Alloy 22 is the U.S. Department of Energy preferred material of construction.

oxidation: (1) A corrosion reaction in which the corroded metal forms an oxide, usually applied to reaction with a gas containing elemental oxygen, such as air. (2) An electrochemical reaction in which there is an increase in the valence of an element resulting from the loss of electrons.

parameter: Data, or values, such as those that are input to computer codes for a total system performance assessment calculation.

patch: A circumscribed area of a surface. In the DOE modeling of waste package corrosion, it is the minimal surface area of the outer container over which uniform corrosion occurs, as opposed to localized corrosion in pits.

pathway: A potential route by which radionuclides might reach the accessible environment and pose a threat to humans. For example, direct exposure is an external pathway, and inhalation and ingestion are internal pathways.

permeability: The ability of a material to transmit fluid through its pores when subjected to a difference in head (pressure gradient). Permeability depends on the substance transmitted (oil, air, water, etc.) and on the size and shape of the pores, joints, and fractures in the medium and the manner in which they are interconnected.

phase: A physically homogeneous and distinct portion of a material system, such as the gaseous, liquid, and solid phases of a substance. In liquids and solids, single phases may coexist.

phase stability: A measure of the ability of a particular phase to remain without transformation.

pit: A small cavity formed in a solid as a result of localized dissolution.

pitting corrosion: Localized corrosion of a metal surface, confined to a small area, that takes the form of cavities named pits.

porosity: The ratio of openings, or voids, to the total volume of a soil or rock expressed as a decimal fraction or as a percentage. See also *effective porosity*.

pre-startup and startup testing: Activities to evaluate the readiness to receive, possess, process, store, and dispose of high-level radioactive waste.

probabilistic: (1) Based on or subject to probability. (2) Involving a variate, such as temperature or porosity. At each instance of time, the variate may take on any of the values of a specified set with a certain probability. Data from a probabilistic process are an ordered set of observations, each of which is one item from a probability distribution.

probabilistic risk assessment: (1) A systematic process of identifying and quantifying the consequences of scenarios that could cause a release of radioactive materials to the environment. (2) Using predictable behavior to define the performance of natural, geologic, human, and engineered systems for thousands of years into the future including probability distributions to account for uncertainty and variability.

probability: The chance that an outcome will occur from the set of possible outcomes. Statistical probability examines actual events and can be verified by observation or sampling. Knowing 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.

processes: Phenomena and activities that have gradual, continuous interactions with the system being modeled. For the purposes of screening features, events, and processes for the total system performance assessment, a process is defined as a natural or human-caused phenomenon that has a potential to affect disposal system performance and that operates during all or a significant part of the period of performance.

process model: A depiction or representation of a process, along with any hypotheses required to describe or to explain the process.

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 (i.e., uranium) of spontaneously emitting radiation (e.g., alpha particles, beta particles, or gamma rays) by the disintegration of atomic nuclei.

radiolysis: Chemical decomposition by the action of radiation.

radionuclide: Radioactive type of atom with an unstable nucleus that spontaneously decays, usually emitting ionizing radiation in the process. Radioactive elements are characterized by their atomic mass and atomic number.

range (statistics): The numerical difference between the highest and lowest value in any set.

receptor: An individual for whom radiological doses are calculated or measured.

relative permeability: The ability of a material to transmit fluid through its pores when subjected to a pressure gradient under unsaturated conditions. Relative permeability is a function of permeability (has a value between 0 and 1).

repository footprint: The areal extent of the underground repository facility.

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 large colloidal molecules in small pore spaces or dead ends of microfractures.

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. This assessment focuses on potential negative impacts on human health or the environment.

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. See *runon*.

runon: Lateral movement of water along the ground surface from some upstream location that becomes available for infiltration. See *runoff*.

safety question: A question regarding the adequacy of structures, systems, and components important to safety and engineered or natural barriers important to waste isolation.

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 potential 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 the purposes of screening or analysis. The number and breadth of scenario classes depend on the resolution at which scenarios have been defined. Coarsely defined scenarios result in fewer, broad scenario classes, whereas narrowly defined scenarios result in many narrow scenario classes. Scenario classes (and scenarios) should be aggregated at the coarsest level at which a technically sound argument can be made while still retaining adequate detail for the purposes of the analysis.

seepage: The inflow of groundwater moving in fractures or pore spaces of permeable rock to an open space in the rock such as a drift. Seepage rate is the percolation flux that enters the drift. Seepage is an important factor in waste package degradation and mobilization and migration of radionuclides out of the potential repository.

seismic: Pertaining to, characteristic of, or produced by earthquakes or earth vibrations.

shallow infiltration: The amount of infiltration that escapes the root zone and percolates downward into the unsaturated zone. See *net infiltration*.

site recommendation: A recommendation by the Secretary of Energy to the President that the Yucca Mountain site is suitable for development as the Nation's first high-level radioactive waste repository.

sorb: To undergo a process of sorption.

sorption: The binding, on a microscopic scale, of one substance to another. A term that includes both adsorption and absorption. The sorption of dissolved radionuclides onto aquifer solids or waste package materials by means of close-range chemical or physical forces is potentially an important process in a repository. Sorption is a function of the chemistry of the radioisotopes, the fluid in which they are carried, and the mineral material they encounter along the flow path.

sorption coefficient (K_d): Coefficient for a term for the various processes by which one substance binds 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. Spent fuel that has been burned (irradiated) in a reactor to the extent that it no longer makes an efficient contribution to a nuclear chain reaction. This fuel is more radioactive than it was before irradiation, and releases significant amounts of heat from the decay of its fission product radionuclides. See *burnup*.

stratigraphy: The science of rock strata. It is concerned with all characters and attributes of rocks as *strata* and their interpretation in terms of mode of origin and geologic history.

stress corrosion cracking: A cracking process that requires the simultaneous action of a corrodent and sustained (residual or applied) tensile stress. Stress corrosion cracking excludes both the fracture of already corroded sections and the localized corrosion processes that can disintegrate an alloy without the action of residual or applied stress.

structure: In geology, the arrangement of the parts of the geologic feature or area of interest such as folds or faults. This includes features such as fractures created by faulting and joints caused by the heating of rock.

tectonic: Pertaining to geologic forms or effects created by deformation of the earth's crust.

tephra: A collective term for all clastic materials ejected from a volcano and transported through the air. It includes volcanic dust, ash, cinders, lapilli, scoria, pumice, bombs, and blocks.

thermal-chemical: Of or pertaining to the effect of heat on chemical conditions and reactions.

thermal-hydrologic: Of or pertaining to changes in groundwater movement due to the effects of changes in temperature.

thermal-hydrologic processes: Processes that are driven by a combination of thermal and hydrologic factors. These processes include evaporation of water near the potential repository when it is hot and subsequent redistribution of fluids by convection, condensation, and drainage.

thermal hydrology: The study of a system that has both thermal and hydrologic processes. A thermal-hydrologic condition, or system, is expected to occur if heat-generating waste packages are placed in the potential repository at Yucca Mountain.

thermal-mechanical: Of or pertaining to changes in mechanical properties of rocks from effects of changes in temperature.

thermodynamics: A branch of physics that deals with the relationship and transformations between work as a mechanical action and heat.

total system performance assessment: A risk assessment that quantitatively estimates how the potential 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, what the outcome was, and why.

transpiration: The removal of water from the ground by vegetation (roots).

transport: A process that allows substances to be carried in a fluid through (1) the physical mechanisms of convection, diffusion, and dispersion; and (2) the chemical mechanisms of sorption, leaching, precipitation, dissolution, and complexation. Types of transport include advective, diffusive, and colloidal.

tuff: A general term for all consolidated pyroclastic rocks. The most abundant type of rock at the Yucca Mountain site.

uncertainty: How much a calculated or measured value varies from the unknown true value.

uniform corrosion: A type of corrosion attack (deterioration) more or less uniformly distributed over a metal surface. Corrosion that proceeds at approximately the same rate over a metal surface. Also called general corrosion.

unsaturated zone flow: The movement of water in the unsaturated zone driven by capillary, viscous, gravitational, inertial, and evaporative forces.

variable: A non-unique property or attribute.

variability (statistical): A measure of how a quantity varies over time or space.

volcanism: Pertaining to volcanic activity.

watershed: The area drained by a river system including the adjacent ridges and hillslopes.