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Rev. 1

Integrated Issue Resolution Status Report

Chapters 1 through 8

U.S. Nuclear Regulatory Commission
Office of Nuclear Material Safety and Safeguards
Washington, DC 20555-0001

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

This Integrated Issue Resolution Status Report provides background information about 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 has, for many years, engaged in prelicensing interactions with DOE and various stakeholders. In recent years, DOE and NRC have reached a number of agreements related to key technical issues important to repository performance after permanent closure and items important to safety during the period before permanent closure. During the prelicensing period, the NRC staff also have undertaken a risk insights initiative to enhance the use of available risk information and develop, as a common basis for understanding, the significance of features, events, and processes that may affect the performance of potential engineered and natural barriers at Yucca Mountain.

This report provides an overview of available information and status (as of March 2004, with exceptions as noted) of the Key Technical Issue agreements reached between DOE and NRC. The report also documents the risk insights (Appendix D) and information considered by the NRC staff in formulating their views, including the results of in-depth reviews of available DOE and contractor documents; the independent confirmatory 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 in understanding the technical rationale used by the NRC staff to identify certain information as being necessary for a quality license application. The staff has not made any determination about compliance with regulations applicable to a potential repository at Yucca Mountain. If DOE submits a license application for a potential repository at Yucca Mountain, the staff will review the information provided by DOE and make determinations based on information provided at that time.

CONTENTS

CONTENTS (continued)

Page

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

Page

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

Page

CONTENTS (continued)

CONTENTS (continued)

Page

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

Page

CONTENTS (continued)

CONTENTS (continued)

FIGURES

TABLES

TABLES (continued)

Page

EXECUTIVE SUMMARY

Introduction

This Integrated Issue Resolution Status Report provides 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 has, 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 potential geologic repository.

To address and document the key technical issues, the NRC staff initiated a formal issue resolution process that includes reviewing the technical information presented in DOE documents; conducting independent confirmatory analyses, experiments, and field work; interacting with DOE in public technical meetings; and identifying the information DOE will need to provide in any potential license application. During the past several years, NRC documented the status of issue resolution through individual status reports for each of the key technical issues to address questions concerning technical information. More recently, the NRC staff intensified their prelicensing interactions with DOE, conducting a series of technical exchanges to address and resolve the remaining questions and concerns. These public meetings discussed the status of issue resolution and reached agreements documenting the additional information DOE needs to provide in a potential license application.

NRC previously documented the status of issue resolution in NUREG–1762 (NRC, 2002). This report updates the earlier report, with a staff assessment of information available as of the end of March 2004 (with exceptions as noted). The status of items covered in this report predates the issuance of the July 2004 D.C. Circuit Court opinion that, among other things, vacated portions of the regulations in 10 CFR Part 63. The report is based on the structure and the review methods contained in NUREG–1804 (NRC, 2003). Discussion of each technical issue also reflects the risk insights currently being developed by NRC to focus its preparations to review a potential DOE license application. The report documents the risk insights (Appendix D) and information considered by the NRC staff in formulating their views, including the results of the in-depth reviews of available DOE and contractor documents; the independent confirmatory work of NRC and its contractor, the Center for Nuclear Waste Regulatory Analyses (CNWRA); published literature; and other publicly available information.

This report 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 submitted, and the review will be documented in a safety evaluation report.

The information in this report may be of value to stakeholders interested in understanding the staff technical rationale for identifying certain information as necessary to a high-quality license application.

Background

The U.S. Congress, in the Nuclear Waste Policy Act (1982), directs DOE to submit information to NRC about site characterization activities before submitting 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, DOE and NRC made issue resolution a major part of the prelicensing interaction specified in the Nuclear Waste Policy Act (1982). The NRC staff issue resolution process includes reviewing the DOE technical 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 that focus on technical or regulatory issues. During prelicensing interactions, issues are considered resolved when there are no further questions at the staff level; however, issue resolution does not signify that a licensing decision has been reached. If DOE submits a license application for a potential repository at Yucca Mountain, staff will review the information provided by DOE and make determinations based on information provided at that time.

The NRC 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 development of what the NRC staff termed key technical issues identified as important to the performance of a potential repository. The NRC staff continued to emphasize these key technical issues in the prelicensing interactions with DOE.

The NRC 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. As a result, the individual key technical issues were refined into subissues that more clearly specified important areas 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 DOE would need to submit a license application. The NRC staff has consistently emphasized that the extent to which DOE addresses the key technical issues for Yucca Mountain provides assurance that DOE can submit a high quality license application for NRC review.

Starting in August 2000, the DOE and NRC staffs engaged in a series of public technical exchanges to identify the information necessary to ensure the key technical issues are addressed in a potential license application for Yucca Mountain. As a result of these technical exchanges, DOE and NRC reached 293 agreements to ensure a high-quality license application. In June 2003, the NRC staff provided the Commission with its ranking of the significance of the 293 high-level waste key technical issue agreements between DOE and NRC. The staff noted that evaluating the significance of the key technical issue agreements was part of a larger effort, referred to as the high-level waste risk insights initiative, and that the agreement risk rankings were based on the risk insights baseline.

In previous years, NRC reported status of issue resolution through individual status reports for each of the key technical issues. Beginning in fiscal year 2001, the NRC staff decided the issue resolution process was mature enough to develop a single Integrated Issue Resolution Status Report to clearly and consistently reflect the interrelationships among the various key technical issue subissues and the overall resolution status. At the same time, NRC began to develop the Yucca Mountain Review Plan to document the review methods and acceptance criteria for the detailed technical review of the DOE license application (NRC, 2003).

Report Structure

This update to the Integrated Issue Resolution Status Report is organized to reflect the structure of the Yucca Mountain Review Plan (NRC, 2003; NUREG–1804) and the results of the NRC risk insights initiative. This report captures the status of progress towards issue resolution through March 2004 (with exceptions as noted).

Based on 10 CFR Part 63 and review of the DOE reports (CRWMS M&O, 2001, 2000), and other support documents, the 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 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. The NRC staff is developing the risk insights to prioritize review of the preclosure aspects of the potential DOE license application. The type of risk information to be used in developing these insights will include available DOE design documents, previous operational experience, and independent confirmatory preclosure safety analyses.

The postclosure section of this report is organized according to a set of integrated subissues as described in NUREG–1804 (NRC, 2003). The NRC staff used an integrated subissue approach, adapted from independent performance assessments conducted by DOE and NRC, in preparing information for many of the key technical issue technical exchanges beginning in August 2000. This approach provides an integrated, transparent structure to review the DOE information pertaining to the key technical issues (Figure 1). The structure is primarily based on the natural progress of moisture downward to the repository level, various processes in the

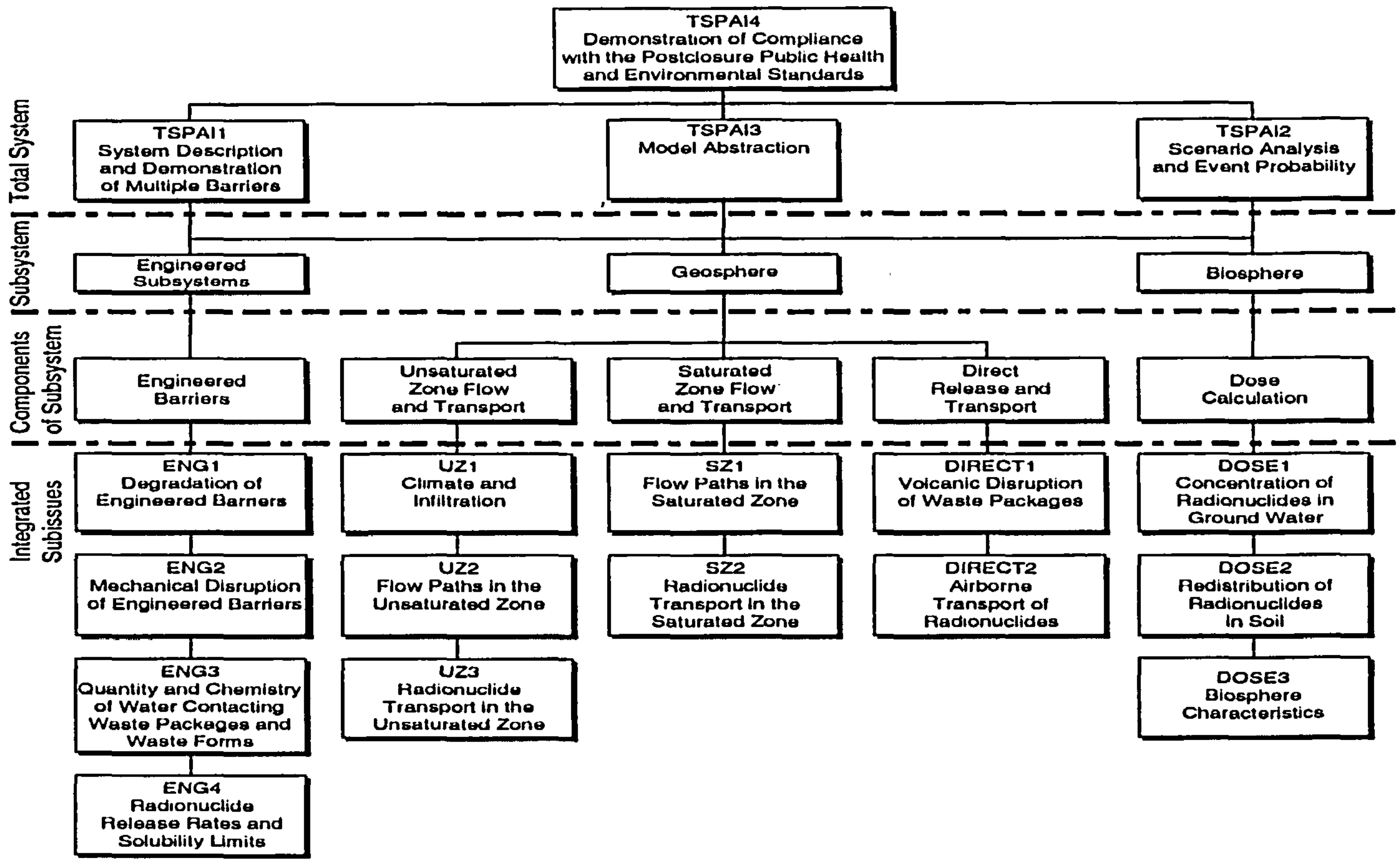

Figure 1. Components of Postclosure Performance Assessment Review

vicinity of the engineered barrier system and 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 also are considered. The topics (14) at the most detailed level 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. In addition, information presented in this report is prioritized to reflect the risk information used as part of the NRC risk insights initiative.

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. Prelicensing activities on preclosure topics will continue, including interactions between DOE and NRC, until the submittal of a potential license application. During prelicensing, the NRC will continue to conduct independent confirmatory preclosure safety analyses, as needed, to better risk inform prelicensing activities.

Postclosure Summary

Consistent with the issue resolution process, the NRC staff intensified prelicensing interactions with DOE to develop information in the areas of the key technical issues. Since August 2000, DOE and NRC have held numerous public technical exchanges focused specifically on the status of issue resolution related to these questions. Results from this increased prelicensing interaction have been documented in formal letters to DOE and in agreements reached in the public meetings between DOE and NRC. In addition, the NRC staff has used the results from its risk insights initiative to focus review on those features, events, and processes most significant to waste isolation.

Prelicensing activities on postclosure topics will continue, including interactions between DOE and NRC, until the submittal of a potential license application. During prelicensing NRC will continue to conduct independent confirmatory postclosure safety analyses, as needed, to better risk inform prelicensing activities.

Summary

This report provides the status of issue resolution between DOE and NRC for a potential high-level waste repository at Yucca Mountain, through March 2004. The issue summaries include updated, risk-informed assessments of the technical bases presented by DOE in the areas of the key technical issues identified for a potential Yucca Mountain repository.

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PREFACE

The Integrated Issue Resolution Status Report documents the status of preclosure and postclosure technical issues that have been the focus of prelicensing interactions related to the potential 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 confirmatory investigations conducted by the U.S. Nuclear Regulatory Commission (NRC) and its contractor, the Center for Nuclear Waste Regulatory Analyses; and (iv) available from a variety of open literature sources. The prelicensing consultations between NRC and DOE are consistent with the Nuclear Waste Policy Act (1982).

This update to the Integrated Issue Resolution Status Report tracks progress toward the resolution of issues and provides this information in a single document to interested parties. Because of the broad scope of this report, however, publication will lag a few months behind availability of the information. For example, although DOE is revising its technical basis to address the key technical issue agreements, this update of the report includes the NRC assessment of status based on information available through March 2004 (with exceptions as noted). The primary organization of this report is based on the structure and review methods developed in NUREG–1804 (NRC, 2003). In addition, information presented in this report is prioritized to reflect the use of risk information as part of the NRC risk insights initiative.

Some sections are absent from this report and others are incomplete. For example, only certain sections are included in Chapter 7, which is devoted to administrative and programmatic requirements for a potential license application.

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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). Staff from both organizations provided information, prepared text, and served as technical, editorial, and programmatic reviewers. The report was coordinated by James L. Rubenstone and Gregory P. Hatchett at NRC and David R. Turner at CNWRA. They thank all participants for their hard work and diligence in preparing this revision to the integrated issue resolution product.

This report was produced in accord with 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

On February 14, 2002, the Secretary of the U.S. Department of Energy (DOE) recommended the Yucca Mountain site to the President for development of a repository. The site recommendation was accompanied by a total system performance assessment and a final environmental impact statement. The president subsequently recommended the site to the U.S. Congress. The governor of Nevada disapproved the site recommendation on April 8, 2002, but the U.S. Congress overrode Nevada's disapproval and approved the recommendation on July 9, 2002. The president signed House Joint Resolution 87 on July 23, 2002, authorizing DOE to prepare a license application to submit to the U.S. Nuclear Regulatory Commission (NRC) for construction of a potential repository at Yucca Mountain, Nevada.

The U.S. Congress in the Nuclear Waste Policy Act (1982), directed DOE to submit information to NRC about site characterization activities before submitting a license application for a potential high-level waste geologic repository at Yucca Mountain, Nevada. The U.S. Congress also directed that NRC issue a final decision approving or disapproving the issuance of a construction authorization no later than 3 years after the date of the submission of such application (except that NRC may extend such deadline by not more than 12 months).

Because of the short time available to review the potential DOE license application, the NRC, consistent with the Nuclear Waste Policy Act (1982), made early identification of issues and issue resolution a major part of the prelicensing interactions. Prelicensing interactions include public meetings at which all stakeholders including the 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 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. Using risk insights information, the NRC staff has developed a risk insights baseline to focus reviews on issues most important to repository performance. The prelicensing consultations and the issue resolution process are in conformance with the NRC efforts to streamline its high-level waste program (NRC, 1999) and prepare for an efficient and competent review of any potential license application DOE may submit.

DOE has the responsibility to present a license application that will demonstrate compliance with all NRC regulatory requirements. Therefore, DOE must appropriately address all aspects of repository performance in its license application. The NRC acceptance review will determine if a potential license application contains sufficient information to be docketed. Prelicensing activities focus on the completeness of DOE information to ensure that DOE is able to submit a high quality license application for NRC review.

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

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 the Oxidation and Dissolution of Spent Nuclear 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 potentialrepository 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	—	—	—	—	—	—

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 2.1. Technical issues related to preclosure safety were not defined in the mid-1990s, but are included in this report.

Status of the NRC staff review of all 10 key technical issues has been documented previously (Sagar, 1997). In fiscal year 1997, it was decided to document issue resolution for each key technical issue in individual reports, and Revision 0 of these Issue Resolution Status Reports was issued in 1997–1998. Revision 0 did not include the Radionuclide Transport Key Technical Issue, work on which was delayed, or the Activities Related to the Development of the 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 program and new information provided in the DOE documents, these reports about individual key technical issues were updated every year, reaching Revision 3 in 2000. In the latter part of fiscal year 2000, DOE and NRC agreed to conduct technical exchanges and management meetings specifically focused on issue resolution and to reach agreements about what additional information DOE needed to provide to resolve the key technical issues.

In fiscal year 2001, the NRC staff decided the issue resolution process was mature enough to develop a single Integrated Issue Resolution Status Report to 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 included. In this way, an Integrated Issue Resolution Status Report would capture the status of the majority of the NRC prelicense application reviews related to the potential repository at the Yucca Mountain site. As a result of implementing that integration initiative, Revision 0 of the Integrated Issue Resolution Status Report was published as NUREG–1762 (NRC, 2002). Following the selection of Yucca Mountain as a potential site for the repository, NRC determined in 2003 that this report would be updated to reflect changes in the DOE program. This report is the update of NUREG–1762. With a few exceptions noted, this report considers information available from DOE as of the end of March 2004. The status of items covered in this report predates the issuance of the July 2004 D.C. Circuit Court opinion that, among other things, vacated portions of the regulations in 10 CFR Part 63.

In the issue resolution status reports for individual key technical issues, issue resolution is 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 are shared among key technical issues, making discussion and resolution cumbersome. As the NRC and CNWRA staffs conducted independent performance assessment exercises through the years and reviewed similar work by the DOE Yucca Mountain Project, Electric Power Research Institute, the DOE Waste Isolation Pilot Plant, 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.

To clarify the issue structure, charts were constructed to depict components of a safety review (Figure 1.1-1) and the relationships among various principal components of a postclosure performance assessment for the potential repository at Yucca Mountain (Figure 1.1-2). These charts show 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 based primarily on the natural progress of potential radionuclide release and transport to a reasonably maximally exposed individual at the Yucca Mountain site. The topics (14) at the most detailed level of review 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 framework for assessing the DOE postclosure performance assessment.

The integrated subissue format and the interdisciplinary questions posed for each of the integrated subissues should integrate more formally the contributions of the key technical issue subissues. Therefore, this structure was adopted to develop the postclosure portions of the standard review plan [the Yucca Mountain Review Plan (NRC, 2003, NUREG–1804)] applicable to the potential repository at Yucca Mountain. NUREG–1804 provides guidance to staff for the review of any license application submitted by DOE, and presents the methods to be used for review, the criteria to be applied for accepting the DOE analyses, and the language suggested for staff findings (NRC, 2003). To create traceability and transparency through better correlation of current assessments with future reviews of the potential license application, the same structure is also followed for the postclosure portion of this document. The structure of this document is based on NUREG–1804 (NRC, 2003).

Chapter 2 of this revision to the Integrated Issue Resolution Status Report contains a brief summary of the regulations that apply to licensing the potential high-level waste repository at Yucca Mountain. It also describes how NRC has used risk insights to evaluate information related to key contributors to repository safety and waste isolation. The generic review methods that form the basis for developing this report are taken from NUREG–1804 (NRC, 2003) and also are described in Chapter 2.

In addition to a safety analysis report, the NRC regulations in 10 CFR 63.21(b) require DOE to include general information as part of its license application. As described in NUREG–1804 (NRC, 2003), the general information in the license application allows DOE to provide an overview of its engineering design concept for the potential repository in the context of the Yucca Mountain site and its environs. The overview material is intended to be generally informational, with detailed technical discussions and descriptions found elsewhere in the safety analysis report section of the license application. Much of the information will consist of plans, programs, and schedules that have not been published by DOE. For this reason, this update of the Integrated Issue Resolution Status Report includes only brief statements in Chapter 3 (General Information) summarizing the prelicensing activities, if any, in those specific areas.

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, 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 to support the site recommendation. Chapter 4 provides a status of the preclosure issues.

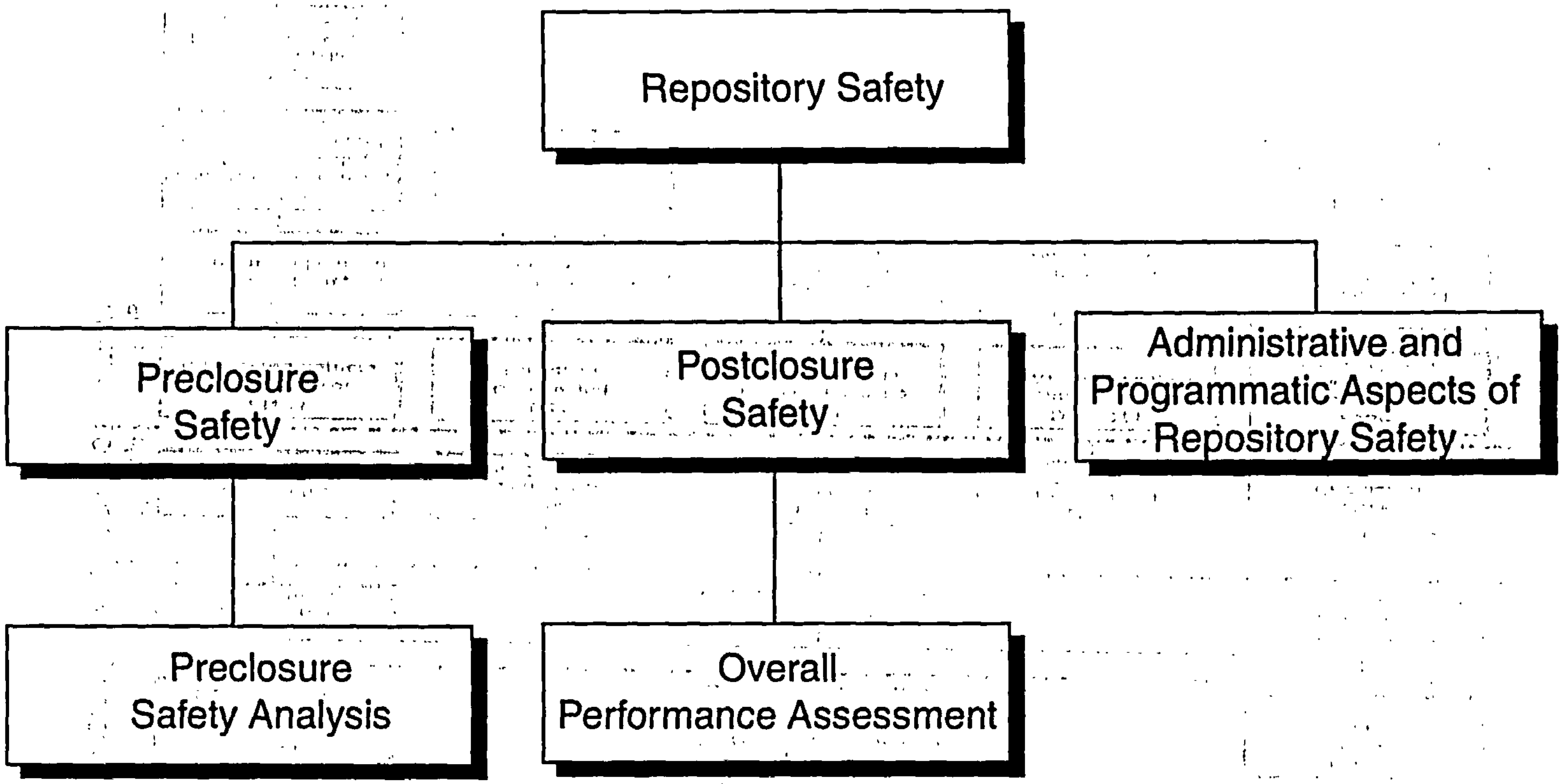

Figure 1.1-1. Review Components of Repository Safety

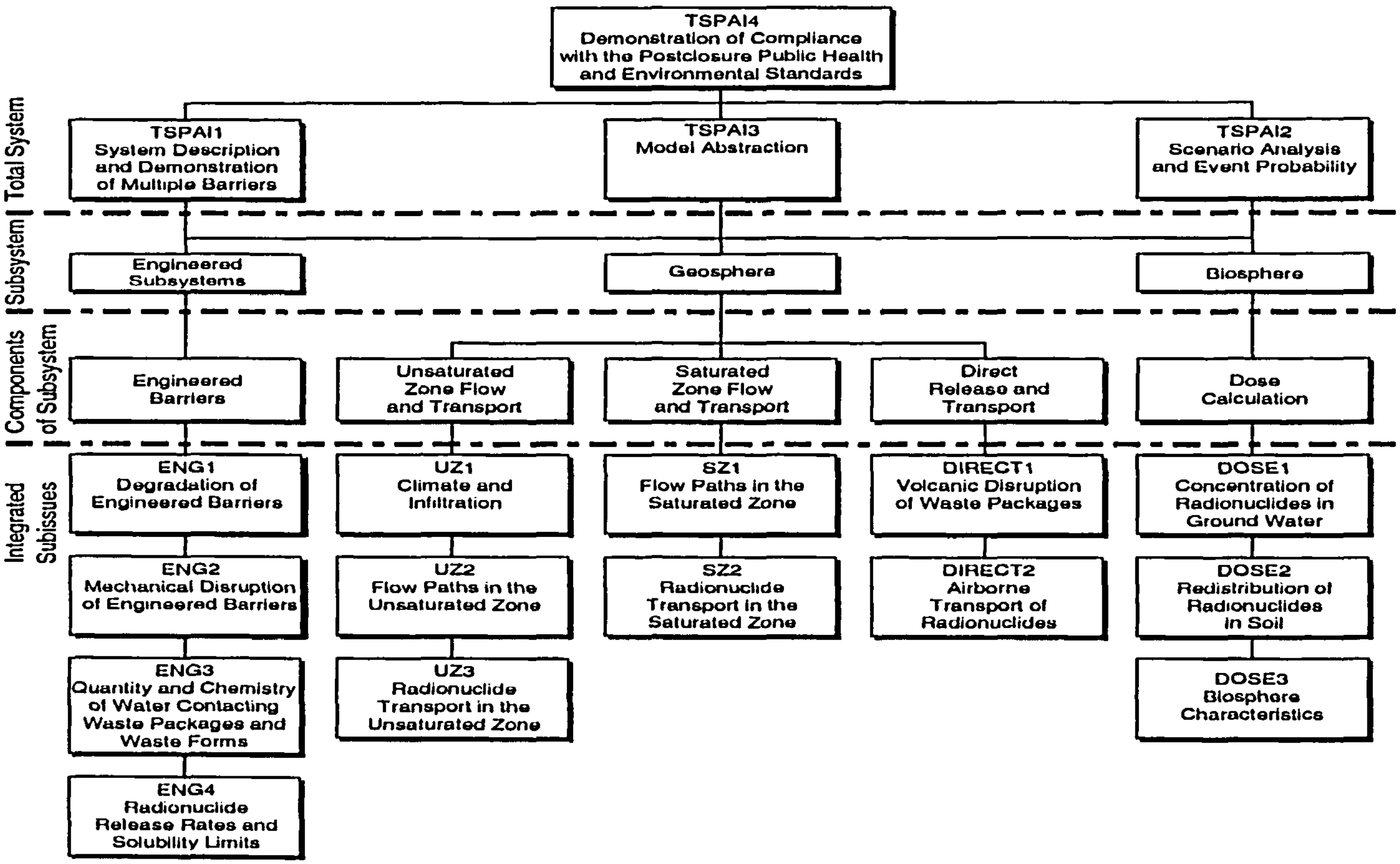

Figure 1.1-2. Components of Postclosure Performance Assessment Review

Based on NUREG–1804 (NRC, 2003), 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; (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 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 (Reamer, 2001).

Chapter 5 of this report documents the status of issue resolution for the 14 integrated subissues associated with model abstraction for postclosure performance. To put 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 also are discussed in Chapter 5. The discussion of TSPAI3—Model Abstraction (Section 5.1.3) covers the 14 integrated subissues. Each integrated subissue draws information from various key technical issue subissues, which are clearly identified in the text; these relationships also are 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 in a safe manner during the preclosure and postclosure periods. Chapter 6 discusses this aspect of the repository program. The DOE research and development programs to resolve any safety questions also are discussed in Chapter 6.

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 according to the Quality Assurance program required by NRC and akin to that set forth in Appendix B of 10 CFR Part 50. 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. Quality assurance and other administrative and programmatic aspects of the Yucca Mountain project are discussed in Chapter 7.

Finally, Chapter 8 provides a summary and conclusions. The DOE and NRC key technical issue agreements are listed in Appendix A.

This report documents the current prelicensing resolution status of preclosure and postclosure issues. This report provides additional background information pertaining to the most recent staff interactions with DOE (through March 2004, with exceptions as noted). The report also documents the information staff considered in formulating their views, including results of the in-depth review of DOE and contractor documents; the independent work of NRC and its contractor, CNWRA; published literature; and other publicly available information.

Table 1.1-2. Relationships Between Integrated Subissues and Key Technical Issues														
Key Technical Issue Subissue	Integrated Subissue													
	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–Concentration of Radionuclides in Ground Water
Dose2 BIO–Redistribution of Radionuclides in Soil
Dose3 BIO–Biosphere Characteristics

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

The report also provides a risk-informed context for the assessment by the NRC staff of the current information available to support a potential DOE license application. Review of the issues is intended to increase the likelihood that DOE will have information available to submit a high quality license application for a potential high-level waste repository at Yucca Mountain. The NRC acceptance review will determine if a potential license application contains the information necessary for the NRC to docket the application and begin its technical review. Potential docketing of an application is not an NRC judgment regarding whether, for example, a construction authorization should be issued. Licensing decisions will only be made after review of any license application.

It is emphasized 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 submitted, and the review will be documented in a safety evaluation report.

1.2 Prelicensing Issue Resolution Process

The NRC strategic plan (2000) calls for early identification and resolution of issues at the staff level before 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 mandated by the Nuclear Waste Policy Act (1982).

The purpose of issue resolution is to ensure that sufficient information is available to enable the NRC staff to review a potential 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 staff review of any license application. During prelicensing, issue resolution at the staff level is achieved when the staff has no further questions or comments at a point in time regarding how DOE is addressing an issue. The agreement items reached during the technical exchanges with DOE reflect the understanding by the NRC staff of issues most important to repository performance. This understanding is based on limited, focused, and risk-informed reviews of selected portions of information made publicly available by DOE. Depending on the DOE responses, agreement items are either closed or needs for additional information are identified by the staff. The availability of new or 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 NRC: closed, closed-pending, and 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 identifies 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 recent technical exchanges, DOE and NRC reached agreements pertaining to a subset of the nine postclosure key technical issues and the associated subissues and preclosure issues. The status of each key technical issue subissue is presented in Table 1.2-1. The agreements reached during the technical exchanges are included in Appendix A.

Table 1.2-1. 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	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

The NRC staff considers all issues open for a potential licensing decision unless and until DOE submits a license application, the staff completes its independent safety review and issues a safety evaluation report, the NRC provides an oppor'unity for a hearing on issues raised by the parties, and the NRC makes its final determination on whether the DOE license application meets the NRC regulations. Any NRC decision will be based on 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). 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 References

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

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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. <www.nrc.gov/waste/hlw-disposal/public-involvement/mtg-archive.html#KTI>

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 RISK-INFORMED REVIEW PROCESS

2.1 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 2.1-1 provides a timeline for pertinent rulemaking (adapted from CRWMS M&O, 2000).

The Nuclear Waste Policy Act (1982) established the national policy and defined the responsibilities of various Federal agencies for the safe disposal of spent nuclear fuel and high-level waste generated mainly as a result of commercial power production and defense activities. As mandated by statute, the U.S. Department of Energy (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 according to other laws; and the U.S. Nuclear Regulatory Commission (NRC) must implement the EPA standards by incorporating them into NRC regulations, issue technical criteria for licensing a repository, and 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 implementing regulations in 10 CFR Part 60. These standards and regulations were intended to apply to all appropriate facilities in the United States, including the potential high-level waste repository at Yucca Mountain, Nevada. In 1987, the U.S. Court of Appeals for the First Circuit Court invalidated the standards and remanded them to EPA (Natural Resources Defense Council, Inc., 1987). Also in 1987, the Nuclear Waste Policy Act (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 Standards (National Research Council, 1995). EPA issued its final standards applicable to Yucca Mountain in 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. After considering public comments on the draft rule, 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.

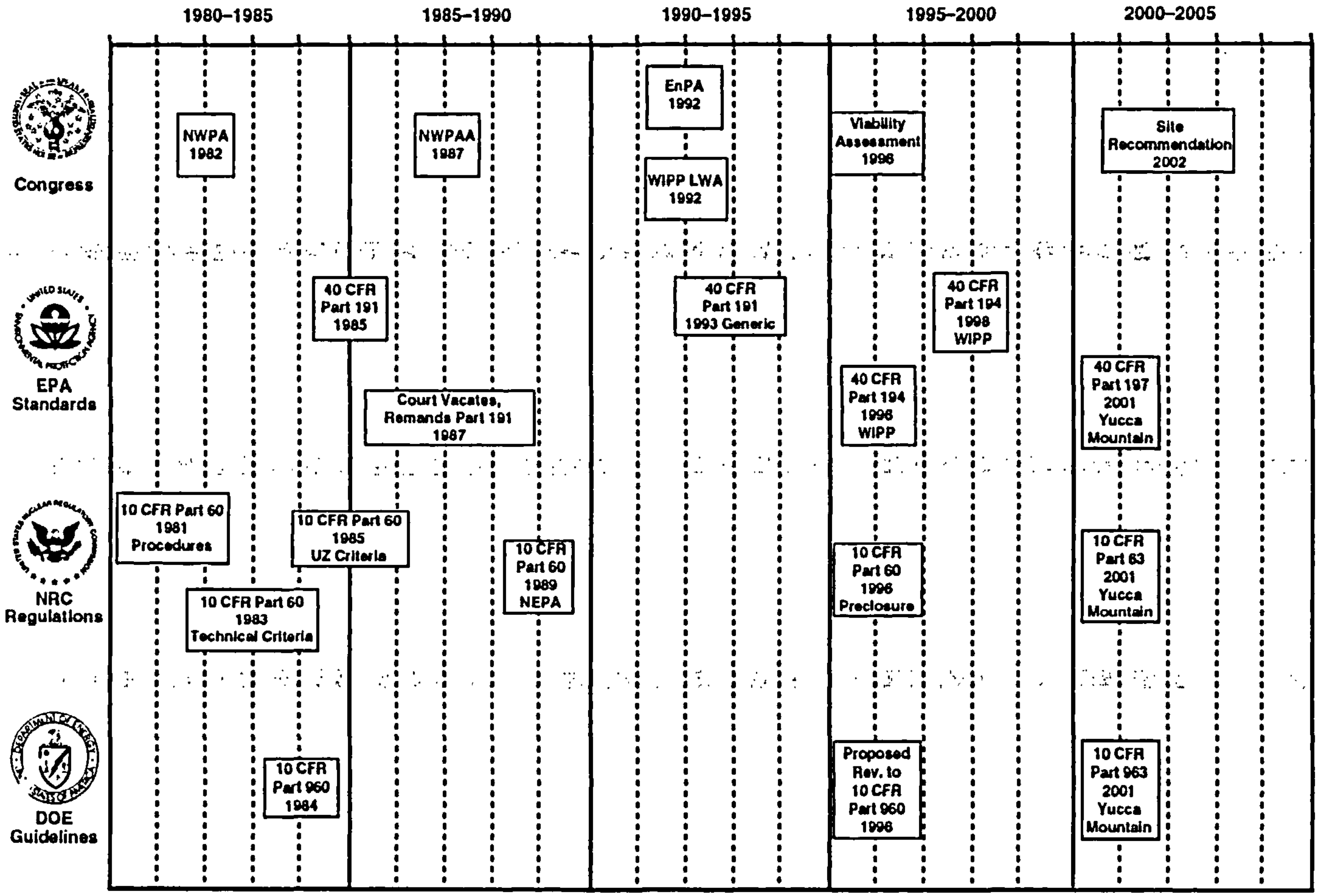

Figure 2.1-1. Timeline of Legislative and Regulatory Events, 1980–2005

EPA Standards

A brief summary of key aspects of the EPA standards is provided next. As previously noted, the discussion in this report predates the issuance of the July 2004 D.C. Circuit Court opinion that, among other things, vacated portions of the regulations in 10 CFR Part 63, and in 40 CFR Part 197.

Radiation Standards: On June 13, 2001, EPA promulgated its final public health and environmental radiation standards (40 CFR Part 197) 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 specified 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 and 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 also has 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 assume 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 is 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 throughout a 10,000-year period.

Performance Assessments: 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 the effects on performance; and estimates the potential radiological consequences. DOE is required to use performance assessment to show compliance with the postclosure performance objectives. 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 1 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, and 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 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 and comment on the proposed requirements. On November 2, 2001, the NRC published its final regulations for disposal of high-level waste in a potential geologic repository at Yucca Mountain, Nevada. The regulations address the performance of the repository system in addition to licensing procedures, records and reporting, monitoring and testing programs, performance confirmation, quality assurance, personnel training and certification, physical protection, 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 these analyses 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.

Preclosure safety analysis is a systematic examination of the site, the design, and the potential hazards and initiating events as well as the 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: Category 1 and Category 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. Consistent with the EPA final standards, Category 1 event 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 1 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] at or beyond the area boundary. The analysis of a specific Category 2 design basis event would include an initiating event 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) that results in (i) failure of a crane lifting a spent nuclear fuel waste package inside a waste handling building, (ii) damage to a building ventilation (filtration) system, (iii) drop and breach of a waste package, (iv) damage to spent nuclear fuel, (v) partitioning of a fraction of the radionuclide inventory to a building atmosphere, (vi) release of some radioactive material through a 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 for safe operations and 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 conduct a preclosure safety analysis to address specific topics. 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; abilities 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 and means to inspect, test, and maintain structures, systems, and components important to safety to ensure continued functioning and readiness.

The EPA final standards require DOE to show compliance with the postclosure performance objectives using a performance assessment subject to certain constraints (see previous discussion of the EPA standards). Evaluation of repository performance is complicated by uncertainties because of the first-of-a-kind nature of the repository and the extremely long time period for the analysis. 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 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 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 (consideration of potentially disruptive events with a probability of occurrence as low as 1 chance in 10,000 of occurring during 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 considering alternative conceptual models of features and processes that are consistent with available data and current scientific understanding. DOE also must 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 the waste or the movement of radionuclides in the geosphere (e.g., waste package and radionuclide sorption capacity of specific hydrogeologic units) are important to isolation of the waste and are termed barriers. An important focus for the performance assessment is the identification of barriers relied on to isolate radioactive waste and the characterization of each barrier's 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.

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

The Quality Assurance program comprises all those planned and systematic actions necessary to provide adequate confidence 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 the 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, and analyzing 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. General requirements for the performance confirmation program state 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, 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 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 to retrieve waste. Emergency planning is intended to ensure DOE is prepared to respond, both onsite and offsite, 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 the repository to preserve the option for waste retrieval. Waste retrieval is intended to be an unusual event only to be undertaken to protect public health and safety. For example, if information becomes available during the performance confirmation program that indicates 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

The purpose of preclosure issue resolution is to ensure sufficient information is available on an issue to enable the NRC staff to review a potential license application and make a licensing decision. The DOE and NRC interactions on the key technical issues and the issue resolution process coincide with the NRC efforts to implement its high-level waste regulatory 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 attempts to identify issues that may need

Commission guidance. These issues may require NRC rule changes, Commission direction, or Commission interpretations of existing policies.

Since August 2000, DOE and NRC have conducted technical exchanges on all the key technical issues and preclosure safety. These technical exchanges discussed issue resolution activities. Agreements were reached between DOE and NRC on additional information needed from DOE in a potential license application. No specific 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.

2.2 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, 1999) 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 decisionmaking, 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, 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. The staff has developed a baseline of risk insights (Appendix D) to risk-inform their review. In its preclosure integrated safety analyses and postclosure performance assessments, DOE demonstrates major potential contributors to safety. Combined with the NRC staff independent analyses, these DOE analyses provide a reasonable framework for selecting items of high importance to system safety and waste isolation and, therefore, that should be subjected to a more thorough NRC review. This approach of risk-informing reviews directly helps to meet the NRC strategic goal to enhance effectiveness, efficiency, realism, and timeliness.

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 (1982), however, permits prelicensing consultation between DOE and NRC.
- The NRC role is not to monitor all DOE repository activities but to oversee and audit them. As 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 subsequently included in a potential 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 for 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. In-depth, detailed analyses can be limited to a 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 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.

The NRC staff has documented available risk information and synthesized and integrated the knowledge gained from this information. This effort has been used to develop risk insights to ascertain which components are most important to waste isolation and to understand why. These insights are, in turn, used to provide staff with an independent baseline understanding of how the components of a potential repository system at Yucca Mountain might function together to isolate waste and, thus, affect risk to public health and safety.

The NRC staff compiled a set of system-level and detailed risk insights to form the risk insights baseline for the postclosure performance of the potential geologic repository system at Yucca Mountain (Appendix D). The risk insights are based on the experience of the NRC staff in conducting and reviewing performance assessments. The risk insights baseline was developed by synthesizing the results of total system performance assessments, subsystem analyses, and auxiliary calculations. The NRC staff did not attempt to address all the components of a potential repository system at Yucca Mountain in the risk insights baseline, focusing instead, on those components estimated to be most important.

The risk insights baseline (Appendix D) presents the current perspective of the NRC staff on the important parameters, models, and assumptions. The risk insights also reflect uncertainties in understanding the features, events, and processes relevant to waste isolation at Yucca Mountain. Generally, important uncertainties are addressed in a total system performance assessment through a variety of approaches such as parameter ranges (e.g., range of retardation factors of radionuclides in alluvium) and conservative modeling (e.g., assume southerly blowing wind direction for igneous activity). The risk insights provide a basis for focusing on the more important technical issues relative to risk and indicate where staff can benefit most from additional information (e.g., reduction of uncertainty in dose estimates).

Risk insights are rated by considering the contribution to, or adverse effect on, the waste isolation capabilities of the repository system. The staff rated the significance of a risk insight as high if the feature, event, or process addressed by the insight could significantly affect the waste isolation capabilities of the repository system. The significance of a risk insight was rated as medium if there could be some effect. The significance was rated as low if there would likely be negligible effect. The magnitudes of the effects are quantified through performance assessment analyses, and their impacts on waste isolation are evaluated by considering potential effects on

- The waste package integrity
- The radionuclide release from the wasteform and waste package
- The transport of radionuclides through the geosphere and biosphere

The risk insights initiative helps promote a clearer and more consistent position of the NRC staff regarding the relative risk significance of technical issues in the high-level waste program. The NRC staff is using the risk insights baseline and the risk ranking of the agreements reached during the DOE and NRC technical exchanges to identify and focus attention on the more important aspects of each topical area.

2.3 Preclosure and Postclosure Assessment Processes

A demonstration of compliance for a geologic repository system is expected to consider engineered and natural features to meet preclosure and postclosure performance objectives. Mathematical modeling and computer simulations are expected to be an important part of any DOE demonstration of repository safety and waste isolation. Other lines of evidence (e.g., natural analogs for postclosure and empirical observations of other nuclear and nonnuclear facilities for preclosure) also are expected to be a part of the DOE safety case. Identification of issues, review of technical information, determination of status, and suggestions about the path forward for resolving specific technical issues are presented in Chapters 4 and 5 for preclosure and postclosure topics.

Detailed review methods are presented in NUREG–1804 (NRC, 2003). For example, in assessing repository safety after permanent closure, 5 generic review methods are applied to each of 14 postclosure model abstractions in the total system performance assessment (Section 5.1.3). The questions associated with each of the following five generic review methods are those for which a review seeks answers.

(1) <u>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?

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

- Has sufficient justification been provided for any excluded coupling?

(2) Data and 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

- 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 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 with those in published literature or those obtained independently by the staff?

- What is the sensitivity of the system safety measures to the parameters?

(4) Model Uncertainty

- 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 Support

- Has DOE demonstrated there is a reasonable physical basis to explain the output of the models or results of other calculations used to draw safety-related conclusions?
- Has DOE assembled other sufficient evidence to support model results?

These generic review questions are customized for the review of each model abstraction and are further refined using risk information to evaluate the most significant features, events, and processes. A similar approach would be used to assess the other preclosure and postclosure sections of any DOE license application, applying the specific review methods from NRC (2003).

2.4 Updating the Integrated Issue Resolution Status Report

The NRC staff is incorporating the risk insights baseline for the postclosure period (Appendix D) into this update of the Integrated Issue Resolution Status Report. For each model abstraction described in NUREG–1762 (NRC, 2002), staff is using the risk insights baseline to develop discussion of its importance to repository performance. By incorporating risk information into the assessment of each model abstraction, the staff can identify those pieces of information most necessary to evaluate the key contributors to waste isolation. Because prelicensing interactions between DOE and NRC for the preclosure period have been less extensive than for postclosure, the NRC staff has not developed an explicit risk insights baseline report for the preclosure period. The NRC has extensive experience in conducting licensing reviews of nuclear facilities, and this experience will contribute to risk informing the review in the preclosure area.

It is the responsibility of DOE to demonstrate compliance with the regulations at 10 CFR Part 63. The NRC will review the entire application to determine if DOE has satisfied the regulatory requirements. The Yucca Mountain Review Plan (NUREG–1804, NRC, 2003), the risk insights baseline (Appendix D), and the information contained in this update of this report will form the bases for the NRC staff to conduct a risk-informed review of a DOE license application for a potential repository at Yucca Mountain. The risk insights baseline will help focus the staff review, by guiding the depth of the staff review in particular areas, and helping develop requests for additional information. The relevance of the risk insights baseline is dependent on the DOE repository design and performance assessment approaches presented in the license application; however, the staff independent analyses provide additional confidence for review of the strengths and limitations of the DOE demonstration of compliance. This approach is consistent with the NRC policy regarding risk-informed, performance-based regulations in which risk insights, engineering analysis, expert judgment, the principle of defense-in-depth, and safety margins are incorporated in licensing decisions.

It is emphasized that this update to the Integrated Issue Resolution Status Report tracks progress toward issue resolution during prelicensing interactions with the NRC staff. With a few exceptions as noted, the review is based on information available by March 2004. 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 potential license application is submitted, and the NRC staff review will be documented in a safety evaluation report.

2.5 References

CRWMS M&O. "Total System Performance Assessment for the Site Recommendation." TDR-WIS-PA-000001. Rev. 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000.

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

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

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

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

3 GENERAL INFORMATION

3.1 General Description

The general information required to be submitted as part of a license application has not been the subject of the U.S. Department of Energy (DOE) and the U.S. Nuclear Regulatory Commission (NRC) prelicensing discussions and no issues have been identified.

3.2 Proposed Schedules for Construction, Receipt, and Emplacement of Waste

Proposed schedules for construction, receipt, and emplacement of waste have not been the subject of DOE and NRC prelicensing discussions and no issues have been identified.

3.3 Physical Protection Plan

The physical protection plan was addressed during one meeting between DOE and NRC in February 2004. At this meeting, NRC outlined the requirements in this area that would apply to a potential repository at Yucca Mountain. DOE made no presentation at the meeting and has not provided any information on its physical protection plan. No issues have been identified in this area.

3.4 Material Control and Accounting Program

The material control and accounting program was addressed during one meeting between DOE and NRC in February 2004. At this meeting, NRC outlined the requirements in this area that would apply to a potential repository at Yucca Mountain. DOE made no presentation at the meeting and has not provided any information on its material control and accounting program. No issues have been identified in this area.

3.5 Description of Site Characterization Work

Detailed assessment of site characterization is discussed in Chapters 4 and 5 of this report. The general description of site characterization work has not been the subject of DOE and NRC prelicensing discussions and no issues have been identified.

4 REPOSITORY SAFETY BEFORE PERMANENT CLOSURE

4.1 Preclosure Safety Analysis

In accordance with 10 CFR 63.21(c)(5), a license application is required to include a preclosure safety analysis to ensure compliance with the performance objectives of 10 CFR 63.111(a), as required by (c).

The preclosure safety analysis, as stated in 10 CFR 63.112, must include

- A general description of the structures, systems, components, equipment, and process activities at the geologic repository operations area.[1]

- An identification and systematic analysis of naturally occurring and human-induced hazards at the geologic repository operations area, including a comprehensive identification of potential event sequences.

- Data pertaining to the Yucca Mountain site, and the surrounding region to the extent necessary, used to identify naturally occurring and human-induced hazards at the geologic repository operations area.

- The technical basis for either inclusion or exclusion of specific, naturally occurring, and human-induced hazards in the safety analysis.

- An analysis of the performance of the structures, systems, and components to identify those that are important to safety. This analysis identifies and describes the controls relied on to limit or prevent potential event sequences or mitigate their consequences. This analysis also identifies measures taken to ensure the availability of safety systems.

- A description and discussion of the design, both surface and subsurface, of the geologic repository operations area, including the relationship between design criteria and the requirements specified at 10 CFR Part 63.111(a) and (b); and the design bases and their relation to the design criteria.

The design objectives as stated in 10 CFR 63.111(b) for the geologic repository operations area are

1. The geologic repository operations area must be designed so that, taking into consideration Category 1 event sequences[2] and until permanent closure has been

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[1]*Geologic repository operations area* means a high-level waste facility that is part of a geologic repository, including both surface and subsurface areas where waste handling activities are conducted.

[2]*Event sequence* means 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. 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. 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 permanent closure are referred to as Category 2 event sequences.

completed, the aggregate radiation exposures and the aggregate radiation levels in both restricted and unrestricted areas, and the aggregate releases of radioactive materials to unrestricted areas, will be maintained within the limits specified in 10 CFR Part 20. In addition, the U.S. Department of Energy (DOE) must ensure no member of the public in the general environment receives more than the annual dose of 15 mSv [15 mrem] from a combination of the management and storage of radioactive material onsite and within the potential Yucca Mountain repository.

2. The geologic repository operations area must be designed so that, taking into consideration any single Category 2 event sequence and until permanent closure has been completed, no individual located on or beyond any point on the boundary of the site will receive, as a result of the single Category 2 event sequence, the more limiting of a total effective dose equivalent of 0.05 Sv [5 rem] or the sum of the deep dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.5 Sv [50 rem]. The lens dose equivalent may not exceed 0.15 Sv [15 rem], and the shallow dose equivalent to skin may not exceed 0.5 Sv [50 rem].

The preclosure safety analysis must demonstrate that the proposed design and operations in the geologic repository operations area meet the requirements of 10 CFR 63.111(b). The preclosure safety analysis must include a systematic examination of the site, the design, the potential hazards, and the initiating events and their resulting event sequences and the potential radiological exposures to workers and the public.

4.1.1 Site Description As It Pertains to Preclosure Safety Analysis

4.1.1.1 Areas of Review

This section provides review of the site description as it pertains to the geologic repository operations area design. The applicable requirements are

- 10 CFR 63.21(c)(1)(i)–(iii) requires a description of the Yucca Mountain site, with appropriate attention to those features, events, and processes of the site that might affect the design or performance of the geologic repository.

- 10 CFR 63.112(c) requires the preclosure safety analysis to include any data used to identify naturally occurring and human-induced hazards at the geologic repository operations area. These are to include site data and data from the surrounding region, to the extent necessary.

Information presented in this section is used in the context of conducting the preclosure safety analysis and to evaluate the design of the geologic repository operations area. Staff will review site description information presented by DOE as part of any potential license application. Staff is currently conducting an exercise to risk inform the review of the preclosure part of the license application. Results from this exercise will guide the review of the license application conducted by the staff.

This section of the Integrated Issue Resolution Status Report addresses assessment of the Yucca Mountain site description. Site description comprises (i) site geography, (ii) regional demography, (iii) local meteorology and regional climatology, (iv) regional and local surface and ground water hydrology, (v) site geology and seismology, (vi) igneous activity, (vii) site geomorphology, and (viii) site geochemistry. 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 2.1, Regulations Applicable to a Potential High-Level Waste Repository at Yucca Mountain, of this report discusses the methodology used by staff for this review.

4.1.1.2 Staff Review of Available Information

The DOE site description is primarily documented in Civilian Radioactive Waste Management System Management and Operating Contractor (CRWMS M&O) (2000a) and in DOE (1999a). Since then, DOE has published a detailed description of the geotechnical data used for preclosure earthquake ground motion analysis (Bechtel SAIC Company, LLC, 2002a). A summary evaluation of the geotechnical data from Bechtel SAIC Company, LLC (2002a) is provided in Section 4.1.1.2.5, Site Geology and Seismology. These reports, plus additional supporting DOE documents identified in the appropriate subsections that follow, are reviewed to the extent 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 report. In addition, this preclosure review incorporates information previously evaluated within the key technical issue framework, including these 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.

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 largely reflects the extensional tectonics that controlled the geologic history of the region throughout the past 65 million years. Strike-slip deformation is also present. Regional topography is characterized by exhumed blocks of crust that form subparallel, north-south-striking ranges separating elongated and internally drained basins. Occasionally, the ranges are dissected by north-northwest-trending dextral strike-slip faults. 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 active. Climate is arid to semiarid, and natural water flow is generally restricted to ground water several hundred meters (500+ ft) below the surface, with occasional surface runoff in washes and across alluvial fan drainages after rainstorms. Ground water flows in several regional and local aquifers contained within alluvial valley-fill sedimentary strata, volcanic rocks, and underlying carbonate strata. The potential repository is to be located in the silicic volcanic rocks, mainly tuffaceous 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 potential 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}.

Review of the site description is organized according to the review methods and associated review criteria identified in U.S. Nuclear Regulatory Commission (NRC) (NRC, 2003). These eight review methods are organized around general subsections of the site description identified in 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 Geography
- Regional Demography
- Local Meteorology and Regional Climatology
- Regional and Local Surface and Ground Water Hydrology
- Site Geology and Seismology
- Igneous Activity
- Site Geomorphology
- Site Geochemistry

4.1.1.2.1 Site Geography

4.1.1.2.1.1 Site Location

Yucca Mountain is located in Nye County, Nevada, approximately 142 km [88 mi] west-northwest of Las Vegas. The potential 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 potential high-level waste repository at Yucca Mountain, Nevada, is adequately identified in CRWMS M&O (2000a).

4.1.1.2.1.2 Significant Natural and Manmade Features

DOE describes natural features at the Yucca Mountain site in CRWMS M&O (2000a). Significant manmade features are identified in Tables 2.2-1 and 2.2-2 and in Figures 2.2-7 through 2.2-10 in CRWMS M&O (2000a). DOE has updated the design, functionality, and layout of the surface facilities since CRWMS M&O (2000a). Current information, as presented in DOE (2004), has been used to update this subsection. The location of various facilities is not final and may be revised as the design of surface facilities matures.

The restricted-access area for waste handling and packaging facilities will include buildings for receiving, packaging, and aging of the incoming radioactive waste. This area consists of one transportation cask receipt building, one transportation cask buffer area, two dry transfer facilities, one remediation building, two aging-facility pads, and one canister handling building. The surface facilities also will include buildings for handling low-level waste and receiving waste packages, and a water retention pond. Support facilities for the repository will include offices for administrative, management, and engineering staff; a fire rescue and medical building; heavy equipment maintenance building; two fire and water facilities; a small vehicle repair shop; security stations; a warehouse; a cooling tower; an electrical generators and switch house; and a fuel depot. The surface facilities could be expanded to include a shielded canister facility and waste processing buildings (DOE, 2004).

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. DOE should identify the locations of all manmade and natural features important to preclosure safety and document them in a potential license application.

4.1.1.2.1.3 Site Maps

CRWMS M&O (2000a) and DOE (2004) contain maps that adequately show (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, (iv) preclosure controlled area, and (v) potential withdrawal area. The maps and information conveyed are adequate to identify these features with regard to preclosure safety assessment in a potential license application.

4.1.1.2.2 Regional Demography

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

4.1.1.2.3 Local Meteorology and Regional Climatology

4.1.1.2.3.1 Climate and Meteorological Conditions

The modern climatic and meteorological conditions at Yucca Mountain are influenced by a broad range of atmospheric conditions including global-scale processes, regional weather patterns, seasonal variations, and local topographically controlled weather patterns (CRWMS M&O, 2000a). The current climate in Central and Southern Nevada 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.

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).

The original Yucca Mountain Radiological and Environmental Field Programs Department meteorological data acquisition network consisted of five stations (CRWMS M&O, 1997). This network was expanded to nine stations in 1992 (CRWMS M&O, 1997, Table 2-1). Five of the nine sites were subsequently reduced to precipitation measurement sites in 1999 (CRWMS M&O, 2000a). The air temperature at these stations has been measured 2 m [6.6 ft] above ground level in mechanically aspirated shields since 1993. Prior to 1993, air temperature was measured 10 m [32.8 ft] above ground level in naturally aspirated shields (CRWMS M&O, 1997).

More recent information presented in CRWMS M&O (2000a, Section 6.2.4) indicates that eight of the nine meteorological sites have towers with wind and temperature sensors mounted 10 m [32.8 ft] above ground level, and one site has a tower with these sensors also mounted at 60 m [196.9 ft] above ground level. The nine sites have temperature, atmospheric humidity, and solar radiation sensors mounted at the 2 m [6.6 ft] level, with barometric pressure and precipitation measurements made near the surface. All sites have tipping-bucket precipitation gauges. More details regarding the acquisition of Yucca Mountain meteorological data are given in CRWMS M&O (2000a, Section 6.2.4.2).

As reported in CRWMS M&O (2000a, Table 6.2-11), the extreme minimum and maximum temperatures measured at the Radiological and Environmental Programs Department site located at Yucca Mountain were −12.5 °C [9.5 °F] and 39.9 °C [103.8 °F]. The monthly mean minimum and maximum temperatures for this measuring station were 2.9 °C [37.2 °F], which was calculated for the months of January and December, and 32.3 °C [90.1 °F], which was calculated for the month of July. In addition, the yearly average number of days above 32 °C [90 °F] is 38.2 and, conversely, the average number of days below freezing {i.e., 0 °C [32 °F]} is 25.4 (CRWMS M&O, 2000a, Table 6.2-11). These temperatures are derived from measurements taken over a 12-year period, from 1986 to 1997.

The maximum 1-hour, 6-hour, and 24-hour precipitation totals measured at the Yucca Mountain Radiological and Environmental Field Programs Department station were 1.3 cm [0.50 in], 2.6 cm [1.03 in], and 4.5 cm [1.78 in], respectively.

Wind measurements taken at the Yucca Mountain Radiological and Environmental Field Programs Department station are summarized in CRWMS M&O (2000a, Table 6.2-11). The average annual mean wind speed at this site was calculated to be 4.3 m/s [9.6 mph]. The fastest 1-minute wind speed, which was derived from 1-second data averaged over 1 minute, was 30.0 m/s [67.1 mph]. And, finally, the peak 3-second gust was reported to be 38.2 m/s [85.5 mph]. Although CRWMS M&O (1997, Section 2.7) states that the Yucca Mountain Radiological and Environmental Field Programs Department and U.S. Geological Survey meteorological monitoring programs have met the appropriate quality assurance requirements to produce qualified data, the DOE response to Key Technical Issue Agreement PRE.03.02 (Ziegler, 2003, Section 6.1) indicates that data provided in CRWMS M&O (1997) are not qualified based on AP-3.15Q, *Managing Technical Product Inputs*.

Although CRWMS M&O (2000a, Section 6.2.4) indicates that insolation data are being acquired for the Yucca Mountain site, no information regarding the actual measured values for this parameter have been provided. Insolation is a site characteristic that should be taken into consideration when determining the dry storage cask decay heat removal capabilities that may be needed for the proposed spent nuclear fuel surface aging area. DOE should provide this information in a potential license application.

4.1.1.2.3.2 Precipitation and Flooding

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 105 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. The staff notes, 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. DOE should provide this information in a potential license application.

4.1.1.2.3.3 Severe Weather

Severe weather events include extreme precipitation events from storms, high winds, and tornadoes. Severe weather conditions at Yucca Mountain are described in Section 6.2 of CRWMS M&O (2000a). Additionally, DOE submitted a report titled Extreme Wind/Tornado/Tornado Missiles Hazard Analysis (Bechtel SAIC Company, LLC, 2003a) in response to the Key Technical Issue Agreement PRE.03.02. This report replaces the report with a similar title (CRWMS M&O, 1999). The staff has reviewed the document (NRC, 2004), and information on this topic is given in Section 4.1.3, Identification of Hazards and Initiating Events, of the present report.

4.1.1.2.4 Regional and Local Surface and Ground Water Hydrology

A review of the integration of surface and ground water characteristics into the design, construction, and operation of the potential repository is a necessary component of the preclosure safety analysis. The primary concerns are inundation, erosion, and deposition by water and debris flows affecting surface facilities and components and elevated flux of water into subsurface tunnels, drifts, and ventilation shafts during the operational phase of the potential repository. To ensure that hydrological features relevant to preclosure safety and potential 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 ground water interaction with the potential repository design. The focus is proportionately on features deemed to be structures, systems, and components important to safety. Accordingly, evaluation is needed for the (i) flood potential, catastrophic erosion, and drainage design for the facilities, systems, and components; (ii) flood and catastrophic erosion near expected transportation pathways, particularly near wash channels; and (iii) design modification and standoff distances from known faults crossing emplacement drifts and access tunnels. These three items are discussed in the context of Surface Waters and Ground Water.

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 north construction portal and the south pad adjacent to the south portal of the Exploratory Studies Facility, aging-facility pads in the northern portion of Midway Valley (McDaniel, 2004), ventilation shafts for the operational period, muck areas in Midway Valley, and the transportation routes used to deliver the waste to the north pad facilities.

Documents reviewed for potential repository and facility design are McDaniel (2004), DOE (2002), and CRWMS M&O (2000b, 1999, 1998b). Documents reviewed for characterization of the natural systems are CRWMS M&O (2000a), DOE (1995), and Bullard (1986). Documents reviewed for preclosure safety are DOE (2002, 2001) and CRWMS M&O (2000c).

4.1.1.2.4.1 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. In addition to flooding, water and rock debris flows are known to occur in the Yucca Mountain area.

Large-magnitude precipitation events can lead to three natural hazard conditions for repository and operational design: (i) localized drainage of water and debris flows deposited onto facilities; (ii) drainage off facility buildings and pads, including increased loads deposited 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 potential repository design (McDaniel, 2004). The ventilation system consists of seven exhaust shafts, four inlet shafts, and the north and south portals of the Exploratory Studies Facility (McDaniel, 2004). The shafts will be vertical with sweeps near the ground surface. Based on CRWMS M&O (2000b), the shafts will not be sited over nor have direct pathways vertically to emplacement drifts. Hence, the safety concern is with operation of the ventilation systems and flooding of localized zones in the tunnels. The primary concern with the ventilation system is that the surface intersection of the shafts should avoid channels and side slopes prone to enhanced runoff and erosion. Shafts on the crest and east flanks of Yucca Mountain pose little risk because the catchment areas are small. It is difficult to determine the topographic locations of shafts northeast of Drill Hole Wash from McDaniel (2004). In addition to a clear delineation of topographic position, DOE should provide more detailed information on flood probabilities for the surface intersections of shafts northeast of Drill Hole Wash.

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 only is 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.

Direct precipitation during intense storms could lead to a flooding of facilities buildings and components. 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 (DOE, 2001). The drainage system for the remainder of the facility will be designed to handle the 100-year flood. CRWMS M&O (2000c) mentions roofs will be designed to withstand a 100-year precipitation event. The north portal itself will be protected by construction of open channels to divert water (DOE, 2001). 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 area). Justification or clarification is needed for the implied use of the 100-year precipitation event for the critical buildings on the pad, specifically the Waste Handling and Waste Treatment Buildings.

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 then can 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 should 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, recurrence interval is not associated with a probable maximum precipitation or flood.

In Bullard (1986) the approach for estimating a probable maximum flood uses a synthetic unit hydrograph coupled to the probable maximum precipitation event and 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 was 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 the Bullard (1994, 1986) 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 (2000b) and DOE (1995) refer to the results of Bullard (1992) and the addition of the bulking factor by Blanton (1992) in discussing probable maximum floods that might affect repository facilities. Portions of the north portal pad are within the area of the probable maximum flood. CRWMS M&O (2000c) and DOE (2001) note that critical buildings and systems, such as the Carrier Preparation Building, the Waste Handling Building, and the Waste Treatment Building, will be above the 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, which is implied to be approximately 2.5 feet below the probable maximum flood level (DOE, 2001). The choice of the 100-year flood leaves flooding as borderline between a category 1 or 2 design consideration (CRWMS M&O, 2000c) using probability criteria, but category 2 is selected (DOE, 2001). However, 10 CFR Part 63 specifies the use of event sequences for categorization of initiating events. Thus, additional description of potential event sequences for flooding is needed for the portion of the north portal area below the probable maximum flood.

Muck piles developed during excavation of the drifts are currently sited in Midway Valley (McDaniel, 2004; CRWMS M&O, 1999, 1998b). Midway Valley has been in a state of sediment aggregation during 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. McDaniel (2004) noted several smaller muck piles south and east of the north portal of the Exploratory Studies Facility. DOE should provide criteria for siting the muck piles that include consideration of flooding.

Siting of potential onsite aging areas in the northern extent of Midway Valley (DOE, 2004; McDaniel, 2004) should consider potential flooding of any drainages leading into northern Midway Valley. Sever, Yucca, and Pageny Washes may contribute to flooding in northern Midway Valley. In addition, siting of muck piles near the north construction portal at the northern extent of Midway Valley, may warrant concern because of potential flooding in Sever and Yucca Washes and the downgradient aging areas. No discussion by DOE of criteria for siting of aging areas has been reviewed.

Transportation pathways for the radioactive waste to the north portal facility may include roadways or railways. In either case, pathways near and across large washes may be prone to catastrophic damage from flooding and debris flows. Fortymile Wash has been the most studied wash in the area. It is said to have reached a state of equilibrium (DOE, 2002, 2001), thus implying that it is not aggrading or degrading. Because a channel segment is in a state of equilibrium, however, does not mean that the sediment flux, and thus erosive or depositional damage, is not large. Information on possible flood damage along the transportation pathways should be included in an evaluation of preclosure risk.

4.1.1.2.4.2 Ground Water

Elevated rates of influx into drifts, access tunnels, and shafts during operations could occur from percolating water down faults or fracture zones from surface flooding, or perched water. Removal of water from possible condensation in exhaust shafts should also be considered.

Focused, fast pathway, downward percolation may occur along large fracture systems and faults (e.g., Drill Hole Waste fault). 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 during periods of flooding at the ground surface. If standard mining practices that would alleviate the problems are to be used, then information on these or alternative practices should be given.

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 Spring Tuff and in the Calico Hills Formation below the repository footprint, but it is unlikely to occur in the repository horizons.

4.1.1.2.4.3 Summary

CRWMS M&O (2000a) and references therein adequately describe streams, drainages, and aquifers that might affect operation of the potential repository. The staff has not reviewed all aspects of the current design (DOE, 2004; McDaniel, 2004) and their interplay with flood and erosion studies. This preliminary assessment identified eight features that warrant further clarification:

- Hydrologic issues for siting of a potential onsite aging area in northern Midway Valley
- Potential water and debris flows from hill slopes above shafts, north and south pads, and north construction portal area
- Siting criteria for ventilation and emplacement shafts
- Routing of surface water around or through the muck piles
- Water level and peak discharge rate differences between the probable maximum flood and the 100-year flood
- Potential event sequences for facility buildings and components that use 100-year flood design considerations rather than probable maximum flood
- Transportation routes to the north pad, particularly in stream channels and flood plains
- Criteria to address water influx from faults that intersect drifts

4.1.1.2.5 Site Geology and Seismology

4.1.1.2.5.1 Site Geology

Site geology includes the regional geologic and tectonic settings, Quaternary stratigraphy and surface processes, Yucca Mountain site stratigraphy and structural geology, and geoengineering properties.

4.1.1.2.5.2 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 (shear faults) and extensional deformation, active since onset of the Cenozoic (65 million years). The region remains tectonically active as indicated by numerous Quaternary (in last 2 million years) faults [including evidence for Holocene (last 10,000 years) 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).

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 Earth 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 (2,500 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 2,500 to 500 million years) silica-rich strata overlain by a thick Paleozoic (approximately 500–245 million years) section of limestones and dolomite. The regional carbonate aquifer is within these Paleozoic limestones and dolomites. 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 valley fill 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 (extensional) and strike-slip (shear) 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 in the Basin and Range Province 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 plates 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 adequately describes sufficient technical bases for the descriptive 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), additional aeromagnetic data were acquired (Blakely, et al., 2000). Although the intent of the aeromagnetic data was to assess potential ground water flow paths in the Amargosa Desert and Death Valley, these new data also provide important new information on the regional geologic setting features such as faults and possible small basaltic volcanoes now buried within the thick accumulations of alluvial material in the basins.

Interpretations of these aeromagnetic data by the United States Geological Survey (O'Leary, et al., 2002) and Center for Nuclear Waste Regulatory Analyses (CNWRA) staff (Hill and Stamatakos, 2002) indicate that as many as 24 anomalies from the Blakely, et al. (2000) survey near Yucca Mountain could originate from buried volcanoes. Although these additional volcanoes could affect estimates of the probability of a volcano forming in the repository during the postclosure period, current evaluations by the staff (as summarized in Section 5.1.2.2) indicate that direct volcanic disruption of the repository is not a concern to the preclosure safety assessment. However, a new volcano might form at some distance away from the site and produce a lava flow or tephra fall that might affect operation of surface facilities. Thus, probability estimates for a new volcano forming in the area around the potential site should evaluate current uncertainties in the number and age of past events.

4.1.1.2.5.3 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 the Yucca Mountain region explain geologic and geophysical data within established tectonic processes. To do so, discrete data sets such as the histories of volcanism, sediment deposition, seismicity, 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 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 ground water flow, igneous activity, 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 deformation 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 tectonic 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, 1999a, Appendix C–1; 1998). 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. The NRC staff considers that, in a broader sense, these five models can be considered within two general categories of deformation. 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 (composite extension plus strike-slip) related to a pull-apart basin bounded by strike-slip faults. Potential implications of the five viable models to repository performance subissues are summarized in NRC (1999a, Appendix C–1; 1998, Appendix C–3).

After the 1996 Appendix 7 meeting, the classification of these tectonic models changed. For example, the full range of tectonic models was presented to the DOE expert elicitation panel, which 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.

Staff reviewed the development and application of tectonic 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 for staff review 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 should be avoided without full justification; 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 descriptions of regional tectonic models as they relate 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 adequately describes the 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 are several new published regional reconstructions of Basin and Range extension (e.g., Snow and Wernicke, 2000). DOE should evaluate the new tectonic models as to their impact on DOE's current understanding of the site geology.

4.1.1.2.5.4 Quaternary Stratigraphy and Surficial Processes

The Quaternary stratigraphy of the Yucca Mountain region yields geological information used to assess (i) faulting activity, (ii) repeat times between large earthquakes on major faults, (iii) ongoing tectonic activity, (iv) volcanism, (v) paleoclimates, (vi) erosion rates, and (vii) sedimentary processes. Landform evolution created by surficial processes is also important to issues of land use in the vicinity of Yucca Mountain. 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. Geological 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 1.6 Ma to the present. The alluvial 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 postclosure performance assessment and preclosure safety analysis and as input to the preclosure seismic design.

The DOE summary of the Quaternary stratigraphy and surficial processes (CRWMS M&O, 2000a) adequately describes the 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 as discussed next.

For preclosure seismic design, specific information on the Quaternary alluvium at the facility site is necessary to construct site response models of earthquake-induced ground motions. DOE collected site information necessary for the site response modeling from surface seismic lines, test borings, test pits, and trenches. Those data include (i) velocities of compression and shear waves in the shallow subsurface to depths of approximately 200 m [650 ft]; (ii) densities of rock and soil units; and (iii) dynamic shear strain and damping values of various rock and soil units. A summary of staff assessment of those data is provided in the Site Geoengineering Properties subsection of this chapter.

DOE has not provided information that describe how these site data coupled with the results of the probabilistic seismic hazard assessment (CRWMS M&O, 1998a) will be used to develop preclosure seismic response spectra and preclosure seismic design values. DOE originally established a timetable for the release of two documents for that purpose. The Seismic Design Inputs Report was to be made available in September 2001 and the Seismic Topical Report 3 in fiscal year 2002 (Schlueter, 2000; Reamer, 2001). The DOE schedule for release of that information has been delayed and changed. The current plans (Ziegler, 2004) indicate that DOE will instead release three documents to support site response and seismic design. Technical Basis Document No. 14: Low Probability Seismic Events will develop the preclosure seismic design and fault displacement inputs and the seismic hazard results used for postclosure performance assessment. DOE plans indicate that detailed information on seismic design inputs will be provided in upcoming documents. Topical Report 2 (DOE, 1997a) will be revised to incorporate the new seismic information and to update the seismic design methodology to be consistent with the performance objectives in 10 CFR Part 63.

In summary, 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 report.

4.1.1.2.5.5 Site Stratigraphy

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).

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, whereas the repository horizon is within the Topopah Spring 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.

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 nomenclature (Scott and Bonk, 1984; Buesch, et al., 1996; Flint, 1998; Ortiz, et al., 1985). Table 4.5-3 of CRWMS M&O (2000a) provides a cross-correlation of these different stratigraphic units. Different compositions of the igneous 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 significant tuff units to the preclosure safety analysis are the Tiva Canyon Tuff and the Topopah Spring Tuff, which are part of the Paintbrush Group. These two tuffs make up the bulk of exposed volcanic rock 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, which overlies the Topopah Spring 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., 1998, 1997).

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 adequately describes the technical bases for the site stratigraphy used to assess the ability of the natural system to help meet preclosure safety performance objectives. However, DOE computer models of unsaturated zone flow, transport, and rockfall that reflect attributes of the stratigraphy often lump or generalize rock units and rock data (such as repository horizon, thickness, percent vitric verses percent zeolitic). This confuses between-model comparisons. For example, some models use data from thermal-mechanical unit TSw2 (Ortiz, et al., 1985) to represent the repository host horizon, but the repository host horizon includes the additional thermal-mechanical unit Tptpul. Also, the uncertainties associated with thickness and lateral continuity of key strata (e.g., Ptn, Chv, Tptpll) should be consistently explained for each model that abstracts data from these strata. For consistency, DOE should correlate the modeled stratigraphic or thermal mechanical units with the developed lithostratigraphy (e.g., Buesch, et al., 1996) for each model that assumes or uses stratigraphic data.

4.1.1.2.5.6 Site Structural Geology

The 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.

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., 1998, 1997). 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 slip on the large block-bounding fault triggers relatively widespread, but predictable, secondary deformation 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 tectonically and geomorphically active, 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. DOE should consider the effects of excavation-induced fractures or other skin effects in pre- and postclosure performance analyses, especially if risk-significant processes are effected (e.g., rockfall). 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 potential repository is optimized to ensure that large unstable blocks are minimized (i.e., drifts perpendicular to the dominant fracture orientation). DOE has not fully utilized fracture data in preclosure repository design, especially within the larger lithophysal units of the Topopah Spring Tuff. Most fracture studies have focused on the middle nonlithophysal units of the Topopah Spring Tuff (e.g., Gritto, et al., 2004).

The 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 not routinely counted. Consequently, small fractures are underrepresented in fracture characterizations. Exclusion of small fractures can 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 provide this information.

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. The staff has 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 (Schlueter, 2000). While the DOE structural geology summary has not yet adequately described the technical bases for the descriptive and process models used to assess the ability of the natural system to help meet preclosure safety performance objectives, DOE has agreed to provide the needed information. Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.2.4.1), is sufficient to expect that the information necessary to assess site structural geology with respect to preclosure safety will be available at the time of a potential license application.

4.1.1.2.5.7 Seismic Hazard Assessment

DOE calculations of the seismic hazards for both pre- and postclosure analyses were developed from a probabilistic seismic hazard analyses conducted by DOE (CRWMS M&O, 1998c; Stepp, et al., 2001). The expert elicitation described in these documents consisted of two components, seismic source characterization and ground motion attenuation. DOE (1997b) outlined the methodology used for its probabilistic hazard analyses, which was accepted by the NRC staff (Bell, 1996).

Seismic Source Characterization. In the seismic source characterization, six teams of experts were used. 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 relied on information provided by the U.S. Geological Survey; DOE; other project-specific Yucca Mountain studies; and published geological, geophysical, and seismological 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).

The expert teams considered all the viable tectonic models, and aspects of all the models were incorporated into all the expert elicitation teams' identifications of seismic sources. The teams relied, to varying degrees, on 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.

Seismic sources in CRWMS M&O (1998c) 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; Simonds, 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.

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, 1998c, 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 tbe 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, 1998c, 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, 1998c; Stepp, et al., 2001), experts relied on empirical relationships that relate surface rupture to earthquake magnitude (e.g., Wells and Coppersmith, 1994). Given these data, there is greater than an 80-percent probability that M6.5 earthquakes will rupture the surface, while there is less than a 20-percent chance that M5.5 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; 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.

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, 1998c). 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).

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).

To quantify the historical seismicity of the region, the DOE facilitation team provided a single earthquake catalog to the expert teams. This catalog was compiled from 12 regional catalogs (CRWMS M&O, 1998c, p. G-2). The initial catalog contained 271,223 earthquakes of M0.5 and larger for the period 1868–1996. This initial catalog was modified in three ways. First, all the magnitudes were converted to moment magnitude ($\mathbf{M_W}$). 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.

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, 1998c, Figures 4-2 and 4-3, example logic tree representations).

Based on the information provided in CRWMS M&O (1998c) and Stepp, et al. (2001), staff concluded that sufficient information exists on seismic source characterization for staff to review this aspect of the probabilistic seismic hazard analysis for a potential license application.

Ground Motion Attenuation. In the ground motion part of the expert elicitation, seven individual experts were used. The experts relied on information provided by the U.S. Geological Survey; DOE; other project-specific Yucca Mountain studies; and published geological, geophysical, and seismological literature. Each expert participated in three workshops, two working meetings, and one-day elicitation meeting held between April 1995 and June 1997, at which the experts exchanged information on ground motion attenuation and participated in additional discussions with other external experts. Details of the workshops are given in CRWMS M&O (1998c).

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. The ground motion attenuation models are mathematical relationships between ground motion and earthquake magnitude, distance, site conditions, and style of faulting and are used to estimate the levels of ground motion that may occur at a site. 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.

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 second. In the probabilistic seismic hazard analysis, a value of 0.0186 second was used. DOE agreed (Schlueter, 2000) that if new studies find that 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, the median attenuation model will be adjusted.

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 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. 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 experts represent the seismic hazard and its associated uncertainty at Yucca Mountain. The separation between the 15th- and 85th-percentile curves conveys the effects of the epistemic uncertainty on the calculated hazards. In general, the greatest contributors to uncertainty in the hazard were within expert uncertainty, rather than expert-to-expert uncertainty. Additionally, the total uncertainty caused by ground motion is larger than the uncertainty caused by the seismic source characterization.

In contrast to seismic source characterization, DOE has not provided complete information regarding ground motion components of the probabilistic seismic hazard assessment. DOE should provide information regarding the ground motion expert elicitation process (see the discussion in Section 7.4, Expert Elicitation). In addition, DOE is planning to revise the ground motion results because ground motions for low annual exceedence probabilities (10^{-6} to 10^{-8}) are deemed to be unrealistically large (see detailed discussion in Section 5.1.2.2.4.3 of this report). DOE has indicated that it will provide this remaining information.

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 potential repository. Inputs to preclosure safety analyses or postclosure performance assessment should include additional information and modifications to the reference hazard results. For example, for preclosure seismic analyses of the surface facilities in Midway Valley, the hazard results need to be adjusted for site-specific effects. These adjustments include amplification of the ground motions because of changes in the physical properties of the volcanic strata and alluvium beneath the proposed surface facility structures. Details of the DOE ground response analyses are provided in the following section on Site Geoengineering Properties.

4.1.1.2.5.8 Faulting Hazard Assessment

Appropriate design parameters for faulting, including setback distances, were 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, 1998c; 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 evaluate the potential hazards of an active fault intersecting vital components of the engineered barrier system, especially waste packages.

In that elicitation, the experts derived probabilistic fault displacement hazard curves for a series of demonstration points at or near Yucca Mountain. These demonstration points were selected to represent faulting and related fault deformation in the subsurface and near the proposed surface facility sites, at least as they were defined during the elicitation process. DOE is currently using the results of that expert elicitation to evaluate the potential consequences of faulting on potential repository performance. At present, DOE considers faulting within the repository to be too infrequent and fault displacements too small to impact repository performance, and as such has screened the faulting disruptive event from consideration in its total system performance assessment. Similarly, DOE does not consider faults near the surface facility sites to be active enough to impact preclosure design.

To determine whether the DOE analyses of faulting will contain sufficient information within a potential license application for Yucca Mountain, the staff reviewed the DOE probabilistic fault displacement results and associated DOE analyses of the potential consequences of faulting. Based on this review of the DOE analyses coupled with risk insights gained from an independent consequence analysis of faulting (Stamatakos, et al., 2003), staff concluded that DOE has assembled sufficient information on the issue of direct faulting for the staff to conduct a review of a potential license application. Overall, the current information is sufficient to expect that the necessary information will be available to assess the probability of faulting affecting the preclosure repository system at the time of a potential license application. However, revised design plans for the surface facilities (McDaniel, 2004) show several potentially important systems, structures, or components are to be placed atop faults in Midway Valley, including the Midway Valley splay and possibly the Bow Ridge fault. DOE should develop information regarding the potential hazards these faults pose to the surface facilities.

4.1.1.2.5.9 Site Geoengineering Properties

The scope of the review criteria on site geoengineering properties includes confirmation of site characterization data, including sufficient geomechanical properties and conditions of host rock and soil where major surface facility construction activities will occur, and verification that the rock mechanics testing data will support evaluation of preclosure safety analysis for the geologic repository operations area design. Staff review of information provided by DOE on geoengineering properties for subsurface design is discussed in Section 4.1.7 of this report.

To characterize the subsurface properties of the soil, alluvium, and bedrock beneath the Waste Handling Building Site, DOE and its contractors conducted a series of detailed geotechnical investigations. Methods and results from these studies through early 2002 are documented in Bechtel SAIC Company, LLC (2002a). Results from this study are currently undergoing detailed review by CNWRA and NRC staff.

Current DOE design plans (e.g., McDaniel, 2004) place surface facility structures above several hundred feet of alluvial soil and nonwelded to densely-welded tuff in Midway Valley. Because the physical properties of the soils and tuff contrast with those in the underlying bedrock, ground motions from earthquakes amplify as the seismic waves propagate upward through the tuff and soil. A site response analysis is therefore needed to develop appropriate ground motion response spectra for seismic design and safety assessments of the surface facility structures. DOE modeled the site response using a one-dimensional, equivalent linear analysis approach (e.g., Silva and Lee, 1987). An input bedrock motion is propagated through the soil column to determine the resulting ground motions at the soil surface. Input bedrock motions are derived from the DOE probabilistic seismic hazards assessment for Yucca Mountain (CRWMS M&O, 1998c).

The main purpose of the DOE geotechnical studies was to gather sufficient subsurface information for a technically defensible seismic site response model. Geotechnical information was also collected in the very shallow subsurface as input to design of the building and aging-facility pad foundations. Data collected at the Waste Handling Building Area site include: (i) detailed geologic data from 15 boreholes drilled to a maximum depth below ground surface of ~200 m [650 ft] and four test pits; (ii) shear wave (V_S) and compression wave (V_P) velocities as a function of depth below ground surface using conventional down-hole and wire-line suspension surveys; (iii) shear wave velocity profiles from spectral analysis of surface waves (e.g., dispersive characteristic of Rayleigh waves when traveling through a layered medium); (iv) caliper and gamma-gamma density wire-line surveys from seven boreholes; and (v) strain dependent shear modulus and damping from combined resonant column and torsional shear laboratory testing of samples collected at the site.

Results show that V_S increases from approximately 700 m/s [~2,300 ft/sec] in the alluvium to approximately 1,800 m/s [~6,000 ft/sec] in the densely welded lower nonlithophysal unit of the

Tiva Canyon Tuff. V_P increase from approximately 1,500 m/s [~5,000 ft/sec] in the alluvium to approximately 3,000 m/s [~9,800 ft/s] in the lower non-lithophysal unit of the Tiva Canyon Tuff. Density values range from approximately 1.6 Mg/m^3 [~100 lb/ft^3] in the nonwelded post-Tiva Canyon (Tuff-X) to approximately 2.3 Mg/m^3 [~145 lb/ft^3] in the middle nonlithophysal unit of the Tiva Canyon Tuff. These relatively large variations in velocity and density suggest that the site response will substantially amplify the input bedrock ground motions. The details of this amplification and their impact on design and preclosure safety assessment are, however, unknown, because DOE has not yet provided the surface design spectra and associated acceleration time histories.

Shear modulus and damping results are available for alluvial and tuff samples at strain levels in the general range of 10^{-4} to 10^{-1} percent. Shear modulus and damping curves as a function of shear strain form an integral part of the site response model. However, these curves are not well constrained by data beyond strain levels of approximately 10^{-1} percent, due to limitations of the testing method. DOE is therefore considering a suite of dynamic property curves to incorporate this uncertainty into the site response modeling, because of likely effects on ground motion amplitudes.

In addition, recent design changes to the surface facility (McDaniel, 2004) indicate that additional geotechnical characterization will be necessary beyond that provided in Bechtel SAIC Company, LLC (2002a). In the latest design configuration, DOE proposes, among other changes, to construct several aging-facility pads north and northeast of Exile Hill. DOE should collect additional information on the subsurface material there to complete site characterization for the preclosure seismic analyses. Overall, however, the available information, along with information that DOE has indicated it will collect and use to develop appropriate earthquake site response models and results is sufficient to expect that the necessary information will be available at the time of a potential license application.

4.1.1.2.6 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 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 ago, approximately 12 basaltic volcanoes are known to have formed within 30 km [19 mi] of the potential repository site. Each of these volcanoes represent igneous events that consisted of one to four scoria cones and multiple subsurface intrusions that extend for kilometers away from the volcano. In addition to these known volcanoes, recent geophysical surveys (Blakely, et al., 2000) indicate that approximately 20 magnetic anomalies could be interpreted as more basaltic volcanoes buried in alluvial basins around the potential repository site (O'Leary, et al., 2002; Hill and Stamatakos, 2002).

Basaltic scoria cone volcanoes 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, depending on the strength and direction of prevailing winds. At distances of approximately 20 km [12 mi] from the volcano, basaltic tephra fall deposits are generally 1–100 cm [0.4–39 in] thick with bulk densities of 1,000–1,700 kg/m^3 [62–106 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 checklist 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 and that this thickness represented a worst-case tephra fall event (DOE, 1999b). DOE excluded roof loading caused by tephra fall from further consideration, because it considered that 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 were considered by DOE to be bounded by sandstorms (DOE, 1999b).

Available analysis or data do not support the basis for the DOE conclusion that a 3-cm [1-in]-thick volcanic tephra deposit represents a 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 potential 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 potential 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). 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 1,000–1,700 kg/m^3 [62–106 lb/ft^3] (e.g., Hill, et al., 1998; NRC, 1999b). These deposit densities can increase by an approximate factor of two when wet, depending on average grain size and sorting of the deposit. Thus, a basaltic volcanic eruption located 1–10 km [0.6–6 mi] from Yucca Mountain represents a Category 2 event that could deposit on the order of 1–100 cm [0.4–39 in] of dry tephra on surface structures, resulting in dry loads between 10 and 1,700 kg/m^2 [2 and 348 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 may have larger amounts of fine-grained particles (i.e., ash) than wind-blown sands. Thus, the possible effects of volcanic ash on air filters and ventilation systems may be different than the effects on these systems from wind-blown sand.

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 should 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 greater than 1 chance in 10,000 to occur before permanent closure. DOE should then evaluate the potential effects of these tephra fall deposits on structures and systems important to safety.

4.1.1.2.7 Site Geomorphology

For preclosure, site geomorphology refers to geologic processes of erosion and deposition and the likelihood that extreme erosion and deposition (e.g., landslides, rock avalanches, and other mass wasting and rapid fluvial degradation in channels or interfluves) might affect site structures and operations. DOE recently submitted a revised plan for type and location of possible surface facilities (DOE, 2004). Included in the new plan are aging-facility pads within the Midway Valley drainage and north-northwest of Exile Hill along the alluvial flanks of Bleach Bone Ridge. DOE has not provided supporting technical information with regard to safety assessment of potential hazards associated with site geomorphology conditions. DOE should update the site description to assess potential hazards with respect to potential geomorphologic hazards.

4.1.1.2.8 Site Geochemistry

4.1.1.2.8.1 Geochemistry of Subsurface Waters

The unsaturated zone at Yucca Mountain contains pore waters, fracture waters, and isolated perched water (CRWMS M&O, 2000a; Bechtel SAIC Company, LLC, 2003b). Yang, et al. (1998, 1996) measured chemical compositions of ambient pore water and perched water from Yucca Mountain and its vicinity. More recent data on perched and pore waters are reported in Technical Basis Document No. 5 (Bechtel SAIC Company, LLC, 2003c, Section 3.2.1); DOE considers the latter to be more representative of the waters that may enter the drift. Pore waters have been extracted from unsaturated zone borehole core samples using high-pressure uniaxial compression and ultracentrifuge 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. The newer ambient pore water compositions include samples from the potential repository horizon in the Topopah Spring Tuff. There are 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.

The pore water analyses of Yang, et al. (1998, 1996) provide valuable characterizations of unsaturated zone ground water chemistry at Yucca Mountain, but there are indications that aspects of these data are unreliable. Yang, et al. (1998, 1996) 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. (1998, 1996) 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. (1998, 1996) 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 ground water compositional data provided by Yang, et al. (1998, 1996); Apps (1997); or Browning, et al. (2000) in any process-level models providing input into the Total System Performance Assessment–Site Recommendation. For the Total System Performance Assessment–License Application, DOE has indicated it will base characterization of unsaturated zone water chemistry on the more recent pore water data obtained by ultracentrifugation of samples from potential repository horizons in the Cross-Drift and in boreholes SD–9 and NRG–7/7A (Bechtel SAIC Company, LLC, 2003c). The analyses show a great deal of diversity in terms of major ion chemical characteristics. This formed the basis for representative waters used in calculating the range of possible seepage water compositions to support a potential license application. The unsaturated zone waters tend to be calcium-sulfate-chloride type, rather than sodium-bicarbonate type, but a range of compositions with varying (Ca^{2+} + Mg^{2+})/(Na^{+} + K^{+}) ratios is observed. As discussed in Section 5.1.3.3 of this report (Quantity and Chemistry of Water Contacting Engineered Barriers and Waste Forms), it appears DOE has obtained sufficient data on unsaturated zone water compositions. However, the report detailing the data and their interpretation was not available at the time of this assessment; therefore, the staff has not yet evaluated whether the new analyses have addressed the uncertainties attending the Yang, et al. (1998, 1996) data.

The reported ambient unsaturated zone water compositions directly affect the seepage water compositions used in DOE models for engineered barrier performance. Appendix D concludes that seepage water chemistry has high significance to waste isolation. Even for ambient conditions (Browning, et al., 2000), water compositions in the unsaturated zone will vary, depending on the types of materials encountered along a particular flow pathway and the duration of those interactions. DOE has agreed to provide a more detailed technical basis for their binning approach.

4.1.1.2.8.2 Geochemistry of Rock Strata

CRWMS M&O (2000a) and Bechtel SAIC Company, LLC (2003b) summarize data provided by DOE on geochemical composition of the rock strata at Yucca Mountain through the year 2000. X-ray diffraction techniques were used to characterize the mineralogy of core samples from boreholes in the vicinity of Yucca Mountain, as well as fracture samples collected within the Exploratory Studies Facility. 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 considered at the time that DOE should provide additional support to characterize the mineralogy of fractures and lithophysal cavities for numerical modeling efforts, such as reactive transport modeling.

More recently, as part of the Single Heater and Drift-Scale Tests, pre-test mineralogical and petrologic analyses of drill cores from the potential repository horizon were conducted (Bechtel

SAIC Company, LLC, 2002b). These data may be more indicative of the ambient geochemical environment of the potential repository. Typical fracture coating solids include stellerite (a zeolite), calcite, crystalline silica, opal, smectite, manganese phases, and feldspar. Both calcite and crystalline silica/feldspar were more abundant in vapor-phase intervals than in non-vapor-phase intervals; stellerite comprised over 40 percent of reported fracture assemblages. These data provide a baseline for comparing post-thermal test mineralogy to the ambient system. It appears DOE has collected sufficient data on ambient rock geochemical features.

4.1.1.2.8.3 Geochemical Alterations

The chemical compositions of ambient ground water from Yucca Mountain are expected to evolve significantly before contacting the engineered barrier system. Several different factors will control the composition of water as it infiltrates 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, 2000d). These models suggest that ambient ground water compositions should adequately characterize seepage compositions for a period of almost 10,000 years, but more information concerning the technical basis should be provided. It is unlikely that ambient pore water will ever drip in significant volumes from the drift crown at the potential 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 system. Thus, ambient ground water above the potential 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 for a period of thousands of years for the potential Yucca Mountain repository are thus difficult and must be accomplished by considering both analytical data and numerical models.

Section 5.1.3.3, Quantity and Chemistry of Water Contacting Engineered Barriers and Waste Forms, of this report addresses the information provided by DOE on their approach to characterizing compositions of seepage water at the drift crown and evaporated water in the engineered barrier system. The two review areas of this subissue that the staff has identified as bearing on preclosure areas involve the DOE approach toward model validation and the treatment of data and model uncertainties. These areas are discussed in detail in Section 5.1.3.3.

4.1.1.3 Summary and Status

Table 4.1.1-1 provides a summary of the status 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. Those items considered pending involve either additional review by staff or additional information from DOE.

Table 4.1.1-1. Summary of Resolution Status of Site Description Preclosure Topic			
Preclosure Item	**Status**	**Related Agreement**	**Note**
Site Geography	Pending	None	Current information sufficient, but site location information may need updates given recent changes to surface facility design.*
Regional Demography	Pending	None	Demographic information should be updated to include most recent census data.
Local Meteorology and Regional Climatology	Pending	None	Current information lacks sufficient information on site insolation. Cask decay heat removal capabilities for the aging-facility pads should be determined. Updated information to include regional data from Amargosa Farms and Nevada Test Site should be provided.
Regional and Local Surface and Ground Water Hydrology	Pending	None	Additional information should be provided to evaluate potential water and debris flows: maximum versus 100-year flood, siting criteria for ventilation shafts, aging-facility pads, and muck piles in Midway Valley, transportation near active drainages, and water influx along faults.
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. Correlation of rock and model units should be provided. Because of changes to surface facility design, additional site soil data should be provided for seismic response models and site design. Additional information on dynamic properties should also be provided to complete evaluation of DOE site response analysis.

Table 4.1.1-1. Summary of Resolution Status of Site Description Preclosure Topic (continued)			
Preclosure Item	**Status**	**Related Agreement**	**Note**
Igneous Activity	Pending	None	DOE should provide further information to support technical bases for tephra deposition at the site.
Site Geomorphology	Pending	None	Current information of site geomorphology should be updated to include recently proposed site facilities in Midway Valley and north-northwest of Exile Hill, to evaluate potential hazards associated with site geomorphologic conditions.
Site Geochemistry	Pending	None	Additional information on types of minerals present in fractures should be provided for reactive transport modeling. DOE should provide further information concerning its treatment of model validation, data, and model uncertainties.
*McDaniel, P. "Surface Facilities Design." *Presentation at the DOE/NRC Technical Exchange on Pre-Licensing Activities and Level of Design Detail, February 3–4, 2004.* Las Vegas, Nevada. 2004. <www.nrc.gov/reading-rm/adams.html>			

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

4.1.2.1 Areas of Review

This section provides review of the description of structures, systems, components, equipment, and operational process activities. The applicable requirement is

- 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 system including (i) dimensions, material properties, specifications, analytical and design methods used along with any applicable codes and standards, (ii) the design criteria used and their relationships to the preclosure and postclosure performance objectives, and (iii) the design bases and their relation to the design criteria.

This review is limited to the general description and location of surface facilities and their functions and operational activities to assess if sufficient information exists for a review of the preclosure safety analysis. A review of the specific design details of the structures, systems, components, and equipment can be found in Section 4.1.7.

4.1.2.2 Staff Review of Available Information

4.1.2.2.1 General Information

In the Yucca Mountain science and engineering report (DOE, 2001, 2002a), DOE proposed a design and process of operations in the potential geologic repository operations area. Since the publication of the report, the proposed design and process operations have been changed. DOE is currently finalizing the design of structures, systems, components, equipment, and operational process activities in the geologic repository operations area. The DOE descriptions of these items are not available, and, therefore, the staff evaluation of the available information is preliminary. The following discussion includes information provided by DOE at a DOE and NRC technical exchange (McDaniel, 2004; Board, 2004) and a presentation made to the Advisory Committee for Nuclear Waste (Harrington, 2003).

Approximately 70,000 metric tons heavy metal [77,162 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 commercial spent nuclear fuel, DOE spent nuclear fuel (including naval reactor spent nuclear fuel), vitrified high-level radioactive waste, and immobilized plutonium. The geologic repository operations area is categorized into surface and subsurface facilities. The surface facilities will be used to receive spent nuclear fuel and defense high-level waste shipments, provide capability to age waste as necessary, and prepare and package the wastes for underground emplacement (McDaniel, 2004; Board, 2004). 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 four primary functional areas: (i) the waste receiving and inspection area, where incoming trucks and rail cars are inspected, received, and temporarily staged; (ii) aging areas, where the received wastes are placed for cooling and radiological decay until ready for disposal; (iii) the surface portion of the waste handling operations area, which includes all buildings where radioactive material is handled for packaging; and (iv) the general support facilities, consisting of administrative buildings, security stations, and warehouses (McDaniel, 2004).

4.1.2.2.2 Surface Facilities

The restricted-access area for the waste handling and packaging facilities will include buildings and equipment for receiving, packaging, and aging the incoming wastes. More specifically, the area consists of one transportation cask receipt building, one transportation cask buffer area, two dry transfer facilities, one remediation building, two aging pads, and one canister handling building (McDaniel, 2004). The surface facilities will include buildings to handle low-level waste and receive the waste packages. A water retention pond also is part of the surface facilities. Support facilities for the repository will include offices for administrative, management, and engineering staffs; a fire rescue and medical building; a heavy equipment maintenance building; two fire water facilities; a small vehicle repair shop; security stations; a warehouse; a cooling tower; an electrical generators and switch house; and a fuel depot. The surface facilities could be expanded to include a shielded canister facility and waste processing buildings (McDaniel, 2004). DOE plans to construct the surface facility in several phases (Harrington, 2003). The transportation cask receipt building, a canister handling facility, a dry transfer facility, an aging facility with a capacity of accommodating 6,000 metric tons heavy metal [6,614 tons] of waste, and a portion of balance of plant facilities to support surface and subsurface operations would be constructed in the first phase. The second dry transfer facility, the remaining aging facilities, and the rest of balance of plant facilities would be constructed in the second phase.

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 the 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. The annual rate of receiving and handling 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 under 10 CFR 63.21(c)(5), the preclosure safety analysis must assume that operations will be carried out at the maximum capacity and rate of receipt of waste stated in the license application.

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 to prepare and package the wastes for underground emplacement (DOE, 1998). The transportation cask is shipped to the surface facility by either road or rail. All waste shipments will be received at the cask receipt security station where they will be inspected. After inspection, the casks may be temporarily staged in the area designated for truck staging or rail staging. Casks would then be transported to the transportation cask receipt building where each cask would be lifted from the railcar or trailer and placed on a site rail transfer cart. The site rail transfer cart is then staged in the transportation cask buffer area before moving to a dry transfer facility or canister handling facility for fuel handling, repackaging, and transporting to the aging facility or underground, as needed.

To control the heat output of the waste package, DOE is considering adding the capability to age as much as two-thirds of the commercial spent nuclear fuel (DOE, 2002b, Section 2.1.1.2.2, page 2-12). Aging would reduce the total thermal energy output to achieve the temperature management goal of the potential repository. The aging facility would include access roads, aisles, security fences, and concrete pads. DOE currently plans to develop two aging areas (Harrington, 2003; McDaniel, 2004) in the surface facility. One aging area of smaller capacity {1,000 metric tons [1,102 tons] of commercial spent fuel} would be located at the north portal operations area. The second area, with a proposed capacity of 20,000 metric tons [22,046 tons] or more of commercial spent fuel would be located northeast of the north portal operations area. Both the vertical-placement approach and horizontal modules may be used for the aging operation. Information is not available about the type and design of the aging casks, configurations of the aging facilities, and operations of the proposed aging facilities.

The dry transfer facility will be designed to process canistered and uncanistered wastes. The 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 (DOE, 1998). 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 and transferred to the waste packages. Transportation casks would arrive at the dry transfer facility from the buffer staging area on a site rail transfer cart (McDaniel, 2004). The impact limiters will be removed and the transportation casks would be unloaded from the site rail transfer cart and placed vertically onto a trolley. The trolley system, which moves on rail, consists of turntables to change directions within the facility. From the unloading area, the trolley would first move the transportation casks to the cask preparation room and then to the assembly transfer room. The empty waste packages on a trolley would be docked in the assembly transfer area. There are 10 waste package configurations for various wasteforms (Brown, 2004). Assemblies would be transferred from transportation casks to the waste packages. During the transfer, the assemblies could be temporarily staged in the fuel element and canister staging area for blending spent nuclear fuel to maintain thermal design loads for the waste packages. The waste package cover lid is welded in the closure area and taken to a loadout area where the waste package is lifted from the trolley, tilted, and placed horizontally in the underground transporter. The dry transfer facility also consists of designated areas for dual purpose canister cutting, waste package remediation, and cask dry remediation. In addition, one of the dry transfer facilities houses a pool for wet storage and remediation of spent nuclear fuel.

The canister handling facility will be designed to process the canistered waste. The canister transfer system will receive DOE spent nuclear fuel, vitrified high-level waste, and immobilized plutonium. The transportation casks would arrive in the facility on a truck carrier. The canister handling facility consists of an operating platform for cask preparation and pits for transportation cask, waste package, and canister staging for waste transfer operations. The waste packages would be transferred onto a trolley for movement to the waste package closure area and then moved to the waste package loadout area, which loads waste packages in the underground transporter. The loadout area consists of a tilt station, rotating table, collar remover, pallet lifts, and lifting fixtures.

4.1.2.2.3 Subsurface Facilities

The subsurface facilities consist of portals and access ramps, access mains, emplacement drifts, shafts to support the subsurface ventilation, and drifts to support monitoring and performance confirmation testing (CRWMS M&O, 1998). The repository host horizon is located above the water table in the unsaturated zone. The physical location and general arrangement of the subsurface facility in the unsaturated zone above the water table take advantage of the mountain natural geologic barriers and other attributes as part of the overall waste isolation strategy. Another design consideration is locating the emplacement drifts away from major faults. To facilitate construction and meet the emplacement schedule, the emplacement areas will be divided into four panels. All panels except Panel 3E will be located west of the Exploratory Studies Facility. Panel 3E will be located east of the main access connecting the north construction ramp and the north ramp of the Exploratory Studies Facility (Board, 2004). Panel 1 is the smallest panel and will be constructed first to meet the emplacement schedule.

The portals and access ramps (north portal, south portal, north ramp, and south ramp) of the existing Exploratory Studies Facility will be integrated into the potential repository and connect the surface and subsurface facilities through the access mains. The north construction portal and ramp will be built north of the north ramp of the Exploratory Studies Facility.

The access mains are a network of tunnels that define the perimeter of, and provide access to, the proposed emplacement panels. 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, to the surface facility through the access ramps (CRWMS M&O, 2000a). The access mains have a nominal diameter of 7.6 m [25 ft] and are provided with rail lines to support transportation of the waste packages to and from the emplacement panels. The east and west mains also will serve to conduct either the intake or exhaust ventilation air to or from the emplacement panels. To support ventilation, three intake shafts, three exhaust shafts, and two exhaust raises also will be constructed. The ventilation for the construction and emplacement sides of the emplacement panels will be separated by bulkheads installed in the east and west access mains (Board, 2004).

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, 2000b).

The support system for the walls and roof (known as the ground-support system) of the emplacement drifts will consist of friction-type, expandable stainless steel rock bolts and perforated sheets. The perforated sheets will be made of stainless steel material (Board, 2004). The rock bolts will be 3 m [10 ft] long, and the perforated sheet will be 3 mm [0.125 in] thick.

Inverts will be used to support the emplaced waste packages, pallet, drift rail system, and drip shield. The invert in each emplacement drift consists of three longitudinal support beams, transverse support beams at equal spacing sitting on top of the longitudinal beams, and two longitudinal rails at the ends of the transverse support beams. The invert will be placed on top of the ballast made of crushed tuff (Board, 2004).

The nonemplacement openings in the underground facility include intake and exhaust shafts, exhaust raises, and other drifts within the emplacement block that will be used for various purposes other than waste emplacement. The ground-support system for these nonemplacement openings (including the access mains and turnouts) was originally designed based on DOE (2001) to initially consist of rock bolts and welded wire fabric. A final ground support consisting of a cast-in-place concrete lining may be installed to provide long-term support for such openings during the preclosure period. Because the design of the underground facility is evolving, it is not known if the proposed ground support for these nonemplacement openings has been changed.

Construction of underground openings and waste emplacement operations will proceed concurrently, and development of underground openings will not interfere with the waste emplacement operations (CRWMS M&O, 1999b; DOE, 1998). The repository openings are constructed to serve a variety of functions. The 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 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 (CRWMS M&O, 1999a; DOE, 2001, 1998).

The subsurface transporter is used to transport the waste package to the emplacement drifts. The subsurface transporter is a shielded cask mounted on a rail car. A locomotive will be coupled to each end of the subsurface transporter at the waste handling loading facility. The two locomotives will move the transporter down the north ramp (sloping at a 2.15-percent grade) and along the access main tunnel to reach the emplacement drift turnout. 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. 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 lifting arms will engage the pallet structure to lift the pallet and waste package off the transporter. After raising the pallet to a desired elevation, the gantry would move the waste package to its emplacement location in the drift and lower the waste package and the pallet onto the drift invert. The gantry would disengage from the pallet and return to a position near the emplacement drift door. If the waste package is to be moved during or after emplacement, it will be removed from the emplacement drift by following the emplacement operations in reverse order.

4.1.2.3 Summary and Status

The staff is reviewing information provided by DOE describing the structures, systems, components, equipment, and operational process activities. This review is coordinated with the review of information to be provided in the preclosure safety analysis and will focus on the following areas:

- 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 facilities
- Descriptions of commercial spent nuclear fuel and high-level waste characteristics
- Descriptions and design details of the engineered barrier system components (e.g., waste package, drip shield, and backfill, if any)
- Descriptions of the geologic repository operations area processes activities and procedures, including material and process flow diagrams; mode of operations, remote and manual; human interactions; and interfaces and interactions between structures, systems, and components

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 the description should be detailed enough for the staff to obtain a clear understanding of the design and operations and consistent with the level of the preclosure safety analysis performed. Consequently, the sufficiency of this subsection should be evaluated in conjunction with other subsections relevant to the preclosure safety analysis, including repository design. Review and evaluation of descriptions of the structures, systems, components, equipment, and operational process activities will continue as the DOE design and preclosure safety analysis are made available.

4.1.2.4 References

Board, M. "Subsurface Facilities Design." *Presentation at the DOE/NRC Technical Exchange on Pre-Licensing Activities and Level of Design Detail, February 3–4, 2004.* Las Vegas, Nevada. 2004. <www.nrc.gov/reading-rm/adams.html>

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CRWMS M&O. "Waste Emplacement/Retrieval System Description Document." SDD–WES–SE–000001. Rev. 01. Las Vegas, Nevada: CRWMS M&O. 2000a.
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McDaniel, P. "Surface Facilities Design." *Presentation at the DOE/NRC Technical Exchange on Pre-Licensing Activities and Level of Design Detail, February 3–4, 2004.* Las Vegas, Nevada. 2004. <www.nrc.gov/reading-rm/adams.html>

4.1.3 Identification of Hazards and Initiating Events

4.1.3.1 Areas of Review

This section provides the review of the identification of hazards and initiating events. The applicable requirements are

- 10 CFR 63.21(c)(5) requires a preclosure safety analysis of the geologic repository operations area, for the period before permanent closure, to ensure compliance with 10 CFR 63.111(a), as required by 10 CFR 63.111(c).
- 10 CFR 63.112(b) requires the preclosure safety analysis include an identification and systematic analysis of naturally occurring and human-induced hazards at the geologic repository operations area.

Information and analysis presented in this section are used to identify hazards and initiating events for conducting preclosure safety analysis to identify those structures, systems, and components that have been credited to keep the radiological consequences from an event sequence below the regulatory limits of 10 CFR 63.111(a) and 63.111(b). Additionally, if DOE elects to design structures, systems, and components against the natural, human-induced, and operational hazards, information presented will form the basis for DOE to develop design bases and design criteria of these structures, systems, and components. A systematic and thorough evaluation of hazards and resulting event sequences is an essential component of identifying structures, systems, and components important to safety.

Staff will review information presented by DOE on hazards and initiating events as part of the preclosure safety analysis of the potential license application. The staff is currently conducting an exercise to risk inform the review of the preclosure part of the potential license application. Results from this exercise will guide the staff potential license application review.

4.1.3.2 Staff Review of Available Information

DOE developed a preliminary list of operational hazards and initiating events that have the potential for radiological consequences during the preclosure period (CRWMS M&O, 1999a). DOE evaluated the suitability of the Yucca Mountain site (DOE, 2001a) based on facility design and operations and preclosure safety analysis discussed in DOE (2001b). The facility design, operations, and the functions of the structures, systems, and components are also described in several system description documents and in DOE (2001c). As discussed in Section 4.1.2 of this report, the DOE facility design is being modified from the one in the site suitability report with significant changes in layout, design, and functionality (McDaniel, 2004; Board, 2004).

Major design changes to the surface facility are including a dry transfer system for handling spent nuclear fuel assemblies and surface aging facilities in place of the pool-based transfer system and adding a staging system to the site recommendation design (McDaniel, 2004). For fuel handling operations, the new design includes two dry transfer buildings, one canister transfer building, and a transporter cask buffer staging area proposed to be constructed in a phased manner. The proposed design would involve substantial changes in facility process activities and requirements of structures, systems, and components for the surface facilities. Additionally, the drift layout, construction phases, and ventilation design for the subsurface facilities have been modified; however, the subsurface operations and systems required for emplacement activities have not changed (Board, 2004). These proposed modifications in design and operations, especially for surface facilities, may significantly affect the preclosure safety analyses submitted earlier (CRWMS M&O, 1999a; DOE, 2001a).

Information on DOE identification of operational hazards and initiating events from surface and subsurface operations for the revised facility design is not available; hence, status of the DOE hazard analysis for the site recommendation design is discussed in this section of the Integrated Issue Resolution Status Report. A list of operational hazards is compiled in Table 4.1.3-1 for the following functional areas: waste receipt and cask transportation, waste handling (canister and assembly transfer), subsurface transportation, and emplacement. The table includes hazards identified in DOE (2001a). Because the aging facility was not a part of the site suitability design, hazards applicable to operations in the aging facility were not identified in DOE (2001a).

In the preliminary natural and human-induced hazards analysis, as summarized in Tables 4.1.3-2 and 4.1.3-3, DOE generated a list of potential external hazards from a generic checklist of 53 human-induced and natural phenomena hazards (CRWMS M&O, 1999b; DOE, 2001b; Bechtel SAIC Company, LLC, 2002a). 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. DOE identified these hazards and initiating events for a 100-year preclosure period using a methodology based on the following five screening criteria (CRWMS M&O, 1999b; DOE, 2001b; Bechtel SAIC Company, LLC, 2002a).

- Potential exists for this event to be applicable to the potential repository site at Yucca Mountain. Additional and separate analyses 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 analyses can justify that the process occurs at too slow a rate to pose any potential hazard to the potential repository during the 100-year preclosure period, the event will be screened out from further consideration.

- Consequence of the event is sufficiently 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 at least 1 in 10,000 of occurring 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 to be true for any natural or human-induced event, the event is included in the hazard list for the potential 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.

Table 4.1.3-1. Preliminary Operational Hazard Analysis				
No.	**Functional Area**	**Generic Hazard*†**	**Potential Event*†**	**Staff Remark**
1	Waste Receipt and Cask Transport	Collision/Crushing	Transportation cask: collision, railcar derailment involving transportation cask, overturning of truck trailer involving cask, cask drop, handling equipment drop on cask	Staff reviewed the information provided by DOE for site recommendation design and has no further questions at this time
		Chemical Contamination/ Internal Flooding	None	
		Explosion/Implosion	None	
		Fire/Thermal	Diesel fuel fire	
		Radiation/Fissile Materials	Radiation exposure to facility workers Criticality associated with cask collision, railcar derailment, or overturned truck trailer and rearrangement of cask internals	

Table 4.1.3-1. Preliminary Operational Hazard Analysis (continued)				
No.	**Functional Area**	**Generic Hazard*†**	**Potential Event*†**	**Staff Remark**
2	Waste Handling–Canister Transfer	Collision/Crushing	Transportation Cask: slap down, handling equipment drop on cask, shield door close on cask Canister: drop, slap down, collision, canister drop onto waste package, canister drop on sharp object, canister drop onto another canister in staging rack Waste Package: drop, slap down, collision	Staff reviewed the information provided by DOE and has no further questions at this time.
		Chemical Contamination/Internal Flooding	None	
		Explosion/Implosion	None	
		Fire/Thermal	None	
		Radiation/Fissile Materials	Radiation exposure to facility workers Criticality associated with staging rack and rearrangement of container internals due to collision/drop of cask/canister	

Table 4.1.3-1. Preliminary Operational Hazard Analysis (continued)

No.	Functional Area	Generic Hazard*†	Potential Event*†	Staff Remark
3	Waste Handling–Assembly Transfer	Collision/Crushing	Transportation Cask: drop, slap down, collision, handling equipment drop on cask Spent Nuclear Fuel Assembly: drop on floor, slap down, collision, drop onto spent nuclear fuel assembly staging rack, drop onto waste package Waste Package before and after closure weld: drop, slap down, drop onto sharp object, collision, handling equipment drop	Staff reviewed the information provided by DOE and has no further questions at this time.
		Chemical Contamination/Internal Flooding	Leakage, uncontrolled draw-down, or filling of remediation pool	
		Explosion/Implosion	None	
		Fire/Thermal	Spent nuclear fuel overheating resulting in excessive clad temperature and zircalloy cladding fire due to loss of water in remediation pool Fuel damage by burn-through during welding process, spent nuclear fuel overheating in waste package resulting in excessive cladding temperature and possible zircalloy cladding fire	

Table 4.1.3-1. Preliminary Operational Hazard Analysis (continued)				
No.	Functional Area	Generic Hazard*†	Potential Event*†	Staff Remark
3	Waste Handling–Assembly Transfer	Radiation/Fissile Materials	Radiation exposure of facility workers Criticality associated with cask collision/drop, rearrangement of cask internals, spent nuclear fuel assembly staging rack, misload of waste package, waste package staging area, waste package collision, drop causing rearranging of container internal	
4	Subsurface Transport	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	Staff reviewed the information provided by DOE and has no further questions at this time.
		Chemical Contamination/Internal Flooding	Flooding from water pipe break	
		Explosion/Implosion	None	
		Fire/Thermal	Fire associated with waste package transporters/locomotive or development equipment	
		Radiation/Fissile Materials	Radiation exposure of facility workers, juvenile waste package failure resulting in release of radioactive waste material Criticality associated with collision/drop of waste package, rearrangement of waste package internals	

Table 4.1.3-1. Preliminary Operational Hazard Analysis (continued)				
No.	**Functional Area**	**Generic Hazard*†**	**Potential Event*†**	**Staff Remark**
5	Emplacement	Collision/Crushing	Emplacement Gantry: derailment Waste Package: drop from emplacement gantry, rockfall, waste package/emplacement gantry collision with equipment or another waste package	Staff reviewed the information provided by DOE and has no further questions at this time.
		Chemical Contamination/Internal Flooding	None	
		Explosion/Implosion	None	
		Fire/Thermal	None	
		Radiation/Fissile Materials	Radiation exposure to facility workers	

*Bechtel SAIC Company, LLC. "Preclosure Safety Analysis Guide." TDR–MGR–RL–000002. Rev. 00. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2002.
†CRWMS M&O. "Monitored Geologic Repository Internal Hazards Analysis." ANL–MGR–SE–000003. Rev. 00. Las Vegas, Nevada: CRWMS M&O. 1999.

Table 4.1.3-2. List of Natural Hazards with DOE Assessment*†				
No.	**Hazard**	**Hazard Definition**	**DOE Assessment**	**Staff Remark**
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	Staff has reviewed information provided by DOE and has no further questions at this time.
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	Staff has reviewed information provided by DOE and has no further questions at this time.
3	Dam Failure	Failure of a large human-made barrier that creates and restrains a large body of water	Not applicable to the hazards list • No dam of sufficient size exists in proximity to Yucca Mountain	Staff has reviewed information provided by DOE and has no further questions at this time.
4	Debris Avalanche	Sudden and rapid movement of debris down steep slopes resulting from intensive rainfall	Applicable to the hazards list • Potential exists at Yucca Mountain • 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	Staff has reviewed information provided by DOE; however, staff is waiting for further information from DOE regarding this hazard. Excavated rock debris placed next to surface facilities should be examined for its effects on structures during a seismic event.
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 slow enough to affect during 100-year preclosure period	Staff has reviewed information provided by DOE and has no further questions at this time.

Table 4.1.3-2. List of Natural Hazards with DOE Assessment*† (continued)				
No.	**Hazard**	**Hazard Definition**	**DOE Assessment**	**Staff Remark**
6	Dissolution	Processes of chemical weathering by which mineral and rock materials pass 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	Staff has reviewed information provided by DOE and has no further questions at this time
7	Eperogenic Displacement	Geomorphic processes of uplift and subsidence that produce 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	Staff has reviewed information provided by DOE and has no further questions at this time.
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 is not sufficient to pose credible hazard during 100-year preclosure period	Staff has reviewed information provided by DOE and has no further questions at this time.
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	Staff has reviewed information provided by DOE and has no further questions at this time

Table 4.1.3-2. List of Natural Hazards with DOE Assessment*† (continued)

No.	Hazard	Hazard Definition	DOE Assessment	Staff Remark
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 ≥ 10^{-6} • Not included in another analysis	In the report, Extreme Wind/Tornado/ Tornado Missiles‡, DOE used a 50-year return period for design-basis wind speed, not 100 years. In addition, Yucca Mountain and surrounding region, according to SEI/ASCE 7–02§, are a special area requiring site-specific measurement of wind speed. Staff review is provided in this report.
11	Flood (Storm and 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 • Consequence of process are sufficiently high • Annual event frequency ≥ 10^{-6} • Not included in another analysis	Staff has reviewed information provided by DOE and has no further questions at this time.
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	Staff has reviewed information provided by DOE and has no further questions at this time.
13	Glacial Erosion	Lowering of Earth's surface due to grinding and scouring by glacier ice incorporated with rock fragments	Not applicable to the hazards list • No potential exists at Yucca Mountain for a glacier	Staff has reviewed information provided by DOE and has no further questions at this time.

Table 4.1.3-2. List of Natural Hazards with DOE Assessment*† (continued)				
No.	**Hazard**	**Hazard Definition**	**DOE Assessment**	**Staff Remark**
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	Staff has reviewed information provided by DOE and has no further questions at this time.
15	High Lake Level	Potential overflow or flooding of lake	Not applicable to the hazards list • No potential exists at Yucca Mountain because there is no lake nearby	Staff has reviewed information provided by DOE and has no further questions at this time.
16	High Tide	High tide in water connected with ocean having potential for flooding inland areas	Not applicable to the hazards list • No potential exists at Yucca Mountain because there is no ocean or coastal area	Staff has reviewed information provided by DOE and has no further questions at this time.
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	Staff has reviewed information provided by DOE and has no further questions at this time.
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 ANSI/ANS 2.8–92‖, site needs to be within 160–320 km [100–200 mi] from ocean for hurricane to be potential natural hazard	Staff has reviewed information provided by DOE and has no further questions at this time.

Table 4.1.3-2. List of Natural Hazards with DOE Assessment*† (continued)				
No.	**Hazard**	**Hazard Definition**	**DOE Assessment**	**Staff Remark**
19	Landslide	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	Staff has reviewed information provided by DOE and has no further questions at this time.
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	Staff has reviewed information provided by DOE and has no further questions at this time.
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	Staff has reviewed information provided by DOE and has no further questions at this time.
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	Staff has reviewed information provided by DOE and has no further questions at this time.
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 $\leq 10^{-6}$	Staff has reviewed information provided by DOE and has no further questions at this time.

Table 4.1.3-2. List of Natural Hazards with DOE Assessment*† (continued)				
No.	**Hazard**	**Hazard Definition**	**DOE Assessment**	**Staff Remark**
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	Staff has reviewed information provided by DOE and has no further questions at this time.
25	Rainstorm	Storm that produces 100-year or greater maximum rainfall rate occurring for 1 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 ≥ 10^{-6} • Bounded by debris avalanche, flooding, and landslide events for which this is an initiator	Staff has reviewed information provided by DOE and has no further questions at this time.
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 • Consequence is significant • Annual event frequency ≥ 10^{-6} • Will be addressed in fire hazard analyses	This hazard will be addressed in the fire hazard analyses of the potential facilities. DOE has not provided the fire hazard assessment for the current facility design.

Table 4.1.3-2. List of Natural Hazards with DOE Assessment*† (continued)

No.	Hazard	Hazard Definition	DOE Assessment	Staff Remark
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	Staff has reviewed information provided by DOE and has no further questions at this time.
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	Staff has reviewed information provided by DOE and has no further questions at this time.
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	Staff has reviewed information provided by DOE and has no further questions at this time.
30	Seismic Activity (Uplifting)	Structurally high area in the crust produced by positive movements for 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	Staff has reviewed information provided by DOE and has no further questions at this time.

Table 4.1.3-2. List of Natural Hazards with DOE Assessment*† (continued)				
No.	**Hazard**	**Hazard Definition**	**DOE Assessment**	**Staff Remark**
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	DOE has not yet submitted agreed-to information on either seismic inputs or seismic design.
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 exists 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 fault displacements are 1×10^{-4} and 1×10^{-5}; structures, systems, and components important to safety will be designed to avoid or withstand design-basis fault displacements (Frequency Categories 1 and 2), as appropriate • Not bounded by another analysis	DOE has not yet provided information on faulting hazard or design. DOE should consider new faults at the surface facility site, given the new layout.

Table 4.1.3-2. List of Natural Hazards with DOE Assessment*† (continued)				
No.	**Hazard**	**Hazard Definition**	**DOE Assessment**	**Staff Remark**
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 exists 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 fault displacements are 1×10^{-4} and 1×10^{-5}; structures, systems, and components important to safety will be designed to withstand design-basis fault displacements (Frequency Categories 1 and 2), as appropriate • Not bounded by another analysis	DOE has not yet provided information on faulting hazard or design.
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	Potential degradation of the emplacement drifts during preclosure period should be assessed. DOE has committed to address this concern in a separate 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	Staff has reviewed information provided by DOE and has no further questions at this time.

Table 4.1.3-2. List of Natural Hazards with DOE Assessment*† (continued)				
No.	**Hazard**	**Hazard Definition**	**DOE Assessment**	**Staff Remark**
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 ≥ 10^{-6} • Screened out because subsurface fault displacement will be the 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	Staff has reviewed information provided by DOE and has no further questions at this time.
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; assumed significant • Annual event frequency ≥ 10^{-6} • Not bounded by another analysis	Staff has reviewed information provided by DOE in the report *Extreme Wind/Tornado/Tornado Missiles*‡ and has no further questions at this time.
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	Staff has reviewed information provided by DOE and has no further questions at this time.

Table 4.1.3-2. List of Natural Hazards with DOE Assessment*† (continued)				
No.	**Hazard**	**Hazard Definition**	**DOE Assessment**	**Staff Remark**
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 because site characterization provided sufficient assurance these types of activities would have been detected	Staff has reviewed information provided by DOE and has no further questions at this time.
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 because site characterization provided sufficient assurance these types of activities would have been detected	Staff has reviewed information provided by DOE and has no further questions at this time.
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	Staff has reviewed information provided by DOE and has no further questions at this time. DOE should provide technical basis to exclude hazards from lava flows in the potential repository surface operations area from the Category 2 event sequence list.
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 (indeterminant; assumed true) • Consequence is indeterminant; assumed significant • Annual event frequency $\leq 10^{-6}$	Staff has reviewed information provided by DOE and has no further questions at this time.

Table 4.1.3-2. List of Natural Hazards with DOE Assessment*† (continued)				
No.	**Hazard**	**Hazard Definition**	**DOE Assessment**	**Staff Remark**
43	Volcanism (Ash Flow and Extrusive Magmatic Activity)	Highly heated mixture of volcanic gases, magma, mobile rock material, and ash traveling down flank of volcano or along ground surface	Not applicable to the hazards list • No potential exists at Yucca Mountain for silicic volcanism	Staff has reviewed information provided by DOE and has no further questions at this time.
44	Volcanism (Ash Fall)	Airborne volcanic ash falling from eruption cloud	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is indeterminant; assumed 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	Staff disagrees with the DOE assessment because potential effects of basaltic tephra fall have not been bounded by existing DOE analyses, and realistic effects may exceed current design bases for roof loads and air circulation system performance. Staff review is provided in this section of the Integrated Issue Resolution Status Report.

Table 4.1.3-2. List of Natural Hazards with DOE Assessment*† (continued)

No.	Hazard	Hazard Definition	DOE Assessment	Staff Remark
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	Staff has reviewed information provided by DOE and has no further questions at this time.

*CRWMS M&O. "MGR External Events Analysis." ANL–MGR–SE–000004. Rev. 00. Las Vegas, Nevada: CRWMS M&O. 1999.
†DOE. "Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation." TDR–MGR–SE–000009. Rev. 00 ICN 03. Las Vegas, Nevada: DOE. 2001.
‡CRWMS M&O. "Extreme Wind/Tornado/Tornado Missile Hazard Analysis." CAL–WHS–MD–000002. Rev. 00B. Las Vegas, Nevada: CRWMS M&O. 2003.
§Structural Engineering Institute/American Society of Civil Engineers. "Minimum Design Loads for Buildings and Other Structures." SEI/ASCE 7-02. Rev. of ASCE 7-98. Reston, Virginia: American Society of Civil Engineers. 2003.
‖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.
¶Bechtel SAIC Company, LLC. "Bounding Characteristics of Credible Rockfalls of Preclosure Period." 800–00C–MGR0–00200–000–00A. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2004. (Current staff understanding)
#International Conference of Building Officials. "Uniform Building Code." Whittier, California: International Conference of Building Officials. 1997.

Table 4.1.3-3. List of Human-Induced Events with DOE Assessment*†				
No.	Hazard	Hazard Definition	DOE Assessment	Staff Remark
1	Aircraft Crash	Accidental impact of aircraft on 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 ≤ 10^{-6} per year	Staff reviewed two DOE documents submitted as responses to agreement PRE.03.01 and had a technical exchange on September 30, 2003.‡ DOE committed to collect additional information on aircraft-related activities to assess the potential hazard. Additionally, the reports will be revised as new information is obtained. Staff review of this hazard is provided in this section.
2	Inadvertent Future Intrusions (Human-Induced)	Human-induced inadvertent future intrusions with regard to 100-year preclosure period involve undetected surface access into potential repository facilities	Applicable to the hazards list* • Potential exists at Yucca Mountain • Rate of process is sufficient to affect 100-year preclosure period • Consequence is indeterminant; assumed significant • Annual event frequency is indeterminant; assumed significant • Will be considered in future safeguards and security analyses—a to-be-verified item	Staff has reviewed information provided by DOE and has no further questions at this time.
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 potential 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; assumed significant • Annual event frequency is indeterminant, assumed significant • Will be considered in future safeguards and security analyses—a to-be-verified item	Staff has reviewed DOE assessment and has no further questions at this time.

Table 4.1.3-3. List of Human-Induced Events with DOE Assessment*† (continued)				
No.	Hazard	Hazard Definition	DOE Assessment	Staff Remark
4	Industrial Activity-Induced Accidents	Accidents resulting from industrial or transportation activities unrelated to the potential 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; assumed significant • Annual event frequency is indeterminant at this time; assumed significant • Not bounded by another analysis	DOE has not submitted the updated report about Nearby Facilities.
5	Loss of Offsite/ Onsite Power	Loss of electric power either generated or controlled by persons outside repository system or loss of power within the potential repository	Applicable to the hazards list • Potential exists at Yucca Mountain • Rate of the process is indeterminant at this time; assumed significant • Consequence is indeterminant; assumed significant • Annual event frequency is indeterminant at this time; assumed significant • Not bounded by another analysis	Staff has reviewed DOE assessment and has no further questions at this time.
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; assumed significant • Consequence of the process is indeterminant at this time; assumed significant • Annual event frequency is indeterminant at this time; assumed significant • Not bounded by another analysis	DOE has not submitted the updated report about Nearby Facilities.
7	Pipeline Accidents	Industrial pipeline transporting hazardous materials	Not applicable to the hazards list • No potential exists at Yucca Mountain because no industrial activities requiring pipelines containing hazardous materials exist or are planned to be located near the site	Staff has reviewed DOE assessment and has no further questions at this time.

Table 4.1.3-3. List of Human-Induced Events with DOE Assessment*† (continued)

No.	Hazard	Hazard Definition	DOE Assessment	Staff Remark
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 because site characterization provided sufficient assurance these types of activities would have been detected	Staff has reviewed DOE assessment and has no further questions at this time.

*CRWMS M&O. "MGR External Events Analysis." ANL–MGR–SE–000004. Rev. 00. Las Vegas, Nevada: CRWMS M&O. 1999.

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

‡Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange on Aircraft Hazards Analysis, September 30, 2003." Letter (October 7) to J. Ziegler, DOE. Washington, DC: NRC. 2003. <www.nrc.gov/waste/hlw-disposal/public-involvement/mtg-archive.html#KTI>

§Bechtel SAIC Company, LLC. "Preclosure Safety Analysis Guide." TDR–MGR–RL–000002. Rev. 00. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2002

Note: CRWMS M&O* states this hazard is not applicable to the hazards list requiring additional analysis; however, Bechtel SAIC Company, LLC§ includes this hazard requiring additional analysis.

Some potential hazards are bounded by the analysis carried out for another hazard. For example, potential effects of a 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. Using the screening process and bounding analyses, DOE reduced the list of possible natural hazards to the potential repository during the 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 (2001b) 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.

DOE has committed to address both range fires and fires within the potential facility (DOE, 2001b) and provide information appropriate to prevention and mitigation controls in the design of the facilities. DOE proposed to install a lightning protection system at the Waste Handling Building to protect the building from any direct lightning strikes. 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, 2001b). Staff has not yet reviewed the analysis. The current DOE plan envisions several facilities where radioactive materials would be handled (e.g., Dry Transfer Facilities, Canister Transfer Facility, etc.). Staff will review the lightning protection system of each of these structures to assess the adequacy of the lightning protection system(s) to be installed.

The staff review of the DOE identification of hazards and initiating events is ongoing. The following is a summary of the staff reviews of information concerning potential aircraft crash, tornado wind (includes both straight wind from an extreme event and tornado wind), tornado missiles, volcanic hazards (includes both volcanic ash fall and volcanic eruption), and operational hazards. Discussions given in this report include only those hazards and initiating events for which DOE provided additional information and analysis based on prior interactions with the staff. DOE provided two reports (Bechtel SAIC Company, LLC, 2002b; CRWMS M&O, 2003a) in response to the key technical issue Agreement PRE.03.01 dealing with aircraft crash hazards. CRWMS M&O (2003b) was provided in response to Agreement PRE.03.02 involving tornado and tornado missile hazards. DOE also included additional information on straight wind in that report. Several structural deformation and seismicity and repository design thermal mechanical effects agreements deal with information and analyses on seismic-related hazards. Discussions on staff review of seismic-related areas is summarized in Section 4.1.1, Site Description As It Pertains to Preclosure Safety Analysis; Section 5.1.2.2, Identification of Events with Probabilities Greater Than 10^{-8} Per Year; and Section 7.4, Expert Elicitation, and is not repeated here.

4.1.3.2.1 Aircraft Crash Hazard

4.1.3.2.1.1 Technical Basis and Assumptions for Methods Used for Identification of Hazards and Initiating Events

DOE conducted an analysis to estimate hazards to the potential repository at Yucca Mountain from potential aircraft crashes (CRWMS M&O, 1999c). DOE (CRWMS M&O, 1999c) used the suggested methodology of NRC (1981a) to estimate the probability of crash of an aircraft onto the potential high-level waste repository. Additionally, CRWMS M&O (1999c) 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). These guidance documents are commonly used for estimating the aircraft crash hazard to a facility and are sufficient for use in developing a potential license application.

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 km [5 and 10 statute mi] 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 km [10 statute mi] and the projected annual number of operations is less than $1{,}000 \times D^2$.

(b) The facility is at least 8 km [5 statute mi] 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 km [2 statute mi] 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 (1999c) concluded that proximity criteria (a) and (c) are satisfied for commercial aircraft, private aircraft, DOE aircraft, and aircraft chartered by DOE. Proximity criterion (b) is not applicable for these types of aircraft. Proximity criteria (a) and (b) are satisfied for military aircraft. DOE concluded that only criterion (c) is not satisfied for military aviation in the vicinity of the potential site; therefore, an analysis estimating the annual crash frequency of only military aviation was provided in CRWMS M&O (1999c).

The NRC staff concluded that criterion (b) of NRC (1981a) has not been met for the potential repository site (Reamer, 2001). As considered in CRWMS M&O (1999c), the number of flights per year exceeds 1,000 by a significant margin (at least 12 to 15 times), and these flights create unusual stress situations as they fly in the restricted airspaces. Importantly, the previous screening criteria are for nuclear power plants, none of which are located under a restricted military airspace. Because criterion (b) has not been satisfied, a detailed analysis is necessary for every type of aircraft flying in the vicinity of the potential site (NRC, 1991a). The annual aircraft crash probability at the potential facility will be the summation of probabilities from all types of aircraft engaged in different operations. DOE agreed to develop a detailed analysis of the aircraft crash hazard using all types of aircraft flying in the vicinity of the potential site.

DOE also is considering the option of a lower-temperature operational mode for the potential repository (DOE, 2001b, Appendix A), which would require extended surface aging of the commercial spent nuclear fuel on pads located near the Dry Transfer Facilities (Harrington, 2004). These aging pads will increase the effective area of the surface facilities that need to be considered for aircraft crash hazard analysis.

DOE (CRWMS M&O, 2003a, Section 2.1.1) developed a methodology to estimate the annual frequency of aircraft crashing onto a particular surface facility. This facility is assumed to be located near or underneath an airspace. Flights are oriented randomly in this airspace. DOE assumed that crash-initiation events are uniformly distributed throughout the flight area (CRWMS M&O, 2003a, Assumption 3.1). Additionally, crash-impact points are also uniformly distributed throughout the circular area where a mishap aircraft may impact the ground (CRWMS M&O, 2003a, Assumption 3.2). The estimated annual crash frequency, F [CRWMS M&O, 2003a, Eq. (2)], is

$$F = \frac{T\beta}{A_f} A_{\mathrm{eff}} \qquad (4.1.3\text{-}1)$$

where

T	—	expected total annual flight time (hour/year) of all flights in the flight area A_f
A_f	—	airspace where the aircraft crash could originate
β	—	mean crash rate per flight hour
$T\beta$	—	expected annual frequency of crashes initiated in flight area A_f
A_{eff}	—	effective area of the facility

DOE (CRWMS M&O, 2003a) did not define "crash-initiation events" for military aircraft flying in the Nevada Test and Training Range or in the airspace above the Nevada Test Site. DOE should provide details of how it is defining "crash-initiation events" for military aircraft. DOE should not assume a uniform distribution for all events that may be included in this term.

DOE (CRWMS M&O, 2003a, Section 2.1.2) developed a methodology to estimate the annual crash frequency of aircraft for situations where the flight time of aircraft in a flight area is unknown but frequency of flights through the airspace is known. Flight paths are assumed to be represented by straight lines in this airspace (CRWMS M&O, 2003a, Assumption 3.5). Based on the assumption of uniformly distributed crash-initiation events (CRWMS M&O, 2003a, Assumption 3.1) and uniformly distributed crashes in the circular crash rage (CRWMS M&O, 2003a, Assumption 3.2), the annual crash frequency, F, onto a particular facility [CRWMS M&O, 2003a, Eq. (5)] would be

$$F = \frac{N\lambda\pi}{L_f} A_{\mathrm{eff}} \qquad (4.1.3\text{-}2)$$

where

N	—	annual frequency of flights transiting the flight area (yr^{-1})
λ	—	crash frequency per flight mile (mi^{-1})

L_f — perimeter of the flight area (mi)
A_{eff} — effective area of the facility (mi^2)

CRWMS M&O (2003a, Section 2.2) also proposed an extension of the methodology given in NRC (1981a) to estimate annual crash frequency of aircraft transiting through an airway. CRWMS M&O (2003a) opined that the formula given in NRC (1981a) cannot be applied to the surface facilities at the potential repository because of the way the edge effects of an airway are considered. NRC (1981a) formulas assign the same crash density, defined as the number of crashes per unit width of the airway, to the entire width of an airway. Therefore, a facility at the center of an airway has the same crash density as that of a facility near the edge. On this basis, CRWMS M&O (2003a) believes that the NRC (1981a) methodology may produce too conservative results for the proposed surface facilities as these facilities are several miles outside the edge of the airways.

In the methodology proposed by DOE, the probability of a crash onto a facility sufficiently outside the established boundaries of an airway (i.e., beyond the crash range, r_c) is zero (CRWMS M&O, 2003a). This proposed methodology does not, however, address the potential for flights straying beyond the established boundaries of an airway. Flight paths outside the established boundaries of an airway are not uncommon. Flight path records, given in Figure IV–1 in Appendix IV of CRWMS M&O (2003a), show that even within 1 week, aircraft violate established boundaries of an airway. The NRC staff informed DOE about this in Schlueter (2003a,b). Additionally, the methodology presented in CRWMS M&O (2003a) points to a scenario where the airway width is significantly larger than the crash range of an aircraft. Generally, the width of federal flight corridors is smaller than the crash range used in this report {40 km [25 mi] for air carriers and 48 km [30 mi] for military aircraft in Assumption 3.17}. Therefore, applicability of the proposed methodology is limited.

The NRC (1981a) methodology has a provision to consider cases where a facility is located outside the established airway. The crash probability, P_{FA}, of aircraft flying federal airways or aviation corridors is (NRC, 1981a)

$$P_{FA} = N \times C \times \frac{A_{eff}}{W} \tag{4.1.3-3}$$

where

C — inflight crash rate per mile for a given aircraft
N — number of flights per year along the airway
A_{eff} — effective area of the facility in square miles
W — width of the airway (plus twice the distance from the airway edge to the site when the site is outside the airway) in miles

NRC (1981a) states this methodology "...gives a conservative upper bound on aircraft impact probability if care is taken in using values for the individual factors that are meaningful and conservative." Therefore, in cases where the facility is outside the established boundaries of an airway, the parameter, W, is the actual width of the airway plus twice the distance from the airway edge to the site. Consequently, the probability of crash onto a facility outside the established boundaries of an airway would be smaller than that if the facility was located inside the airway, but not necessarily zero.

CRWMS M&O (2003a, Section 2.3) also provides a methodology to estimate the annual frequency of crash of helicopters flying over the potential facility following the DOE standard DOE–STD–3014–96 (DOE, 1996). The crash frequency of helicopters, *F*, flying over a facility is [CRWMS M&O, 2003a, Eq. (9)]

$$F = \frac{NC}{2Dd} \cdot A_{eff} \tag{4.1.3-4}$$

where

N	—	number of flights per year
C	—	probability of crash per flight
D	—	average length of a flight in miles
d	—	distance in miles on either side of the flight path over which crashes are assumed to be uniform
A_{eff}	—	effective area of the facility in square miles

DOE standard DOE–STD–3014–96 (DOE, 1996) prescribes a crash range, *d*, of 0.4 km [0.25 mi] on either side of the flight path.

Additionally, CRWMS M&O (2003a, Section 2.4) provides a methodology to estimate the annual frequency of objects unintentionally dropped from a military aircraft flying over a given facility. The annual frequency of objects, *F*, unintentionally dropped from aircraft flying over a facility, striking a facility, is [CRWMS M&O, 2003a, Eq. (10)]

$$F = \frac{N\pi\alpha}{LD} \cdot A_{eff} \tag{4.1.3-5}$$

where

N	—	number of annual overflights
α	—	average rate of objects unintentionally dropped in a sortie
D	—	average flight distance in a sortie
A_{eff}	—	effective area of the facility for these dropped objects
L	—	perimeter of the area of interest

The methodology assumes that the rate at which these objects are dropped is uniform along the entire flight path. Conceptually, stressful activities, such as combat maneuvering training, flying at high Gs, may increase the rate. DOE should clarify the basis of this assumption.

4.1.3.2.1.2 Use of Relevant Data for Identification of Site-Specific Hazards and Initiating Events

Bechtel SAIC Company, LLC (2002b) provides information about the flight environment within a radius of 160 km [100 statute miles] of the North Portal of the potential repository at Yucca Mountain. This region includes

- Nevada Test and Training Range

- Nevada Test Site (which includes the potential repository facility at Yucca Mountain)
- R–2508 Range Complex including China Lake Naval Weapons Center
- Airspace supporting Nevada Test and Training Range including Low-Altitude Training Navigation areas, military training routes, and air refueling tracks
- Civilian, DOE, and military airports and airfields
- Federal airways

Figure 4.1.3-1 shows only the region within approximately 48 km [30 mi] from the North Portal. In addition, ground-to-ground missiles are tested at the Nevada Test Site Area 26. Kistler Corporation has been granted a license to operate and test space reentry vehicles at Area 18.

Bechtel SAIC Company, LLC (2002b, Appendix B), based on U.S. Air Force (1999), provided details of ordnance carried onboard an aircraft in the Nevada Test and Training Range on different types of Missions. Air-to-ground ordnance are deployed in the 60 Series and 70 Series ranges as part of training activities. In addition, the Nevada Test and Training Range uses air-to-air missiles as part of training although actual launching of these missiles is prohibited due to safety concerns (Bechtel SAIC Company, LLC, 2002b, Section 5.1.4.2).

4.1.3.2.1.2.1 Nevada Test Site

The Nevada Test Site is operated by DOE and lies underneath the restricted areas R–4808N and R–4808S. R–4808N is exclusively and continuously controlled by DOE and is divided into restricted airspaces: R–4808A, R–4808B, R–4808C, R–4808D, and R–4808E. The surface facilities of the potential repository would be located beneath restricted airspace R–4808E. Southwestern and western portions of R–4808 are used by military aircraft for transiting to and from R–4807A and R–4807B. DOE permits military aircraft to transit R–4808 across the Nevada Test Site for entering or exiting the ranges in the north. Consequently, direct overflights of the potential location of the surface facilities are possible by some aircraft. There is a Memorandum of Understanding between the U.S. Air Force and DOE regarding military flights through R–4808N (Kimura, et al., 1998). Under the Memorandum of Understanding, military aircraft are permitted to transit the airspace over the Nevada Test Site, unless specifically notified by DOE, in normal flight mode. R–4808A is not used for any flight training activities. Any overflight through this space is by emergency aircraft or other aircraft on approved missions subjected to restrictions.

Bechtel SAIC Company, LLC (2002b, Section 5.1.2.2) states that aircraft flying through the airspace above the Nevada Test Site are not restricted to any specific corridor; however, this is only an assumption in the analysis conducted by Kimura, et al. (1998). DOE should provide the basis that supports this statement.

R–4808S airspace is designated as joint use by the Federal Aviation Administration and is jointly used by the Nevada Test Site, Nellis Air Traffic Control Facility, and Federal Aviation Administration–Los Angeles Air Route Traffic Control Center for overflights by civilian aircraft. Federal Aviation Administration uses this airspace at or above Flight Level 280 {8,400-m [28,000-ft]} altitude.

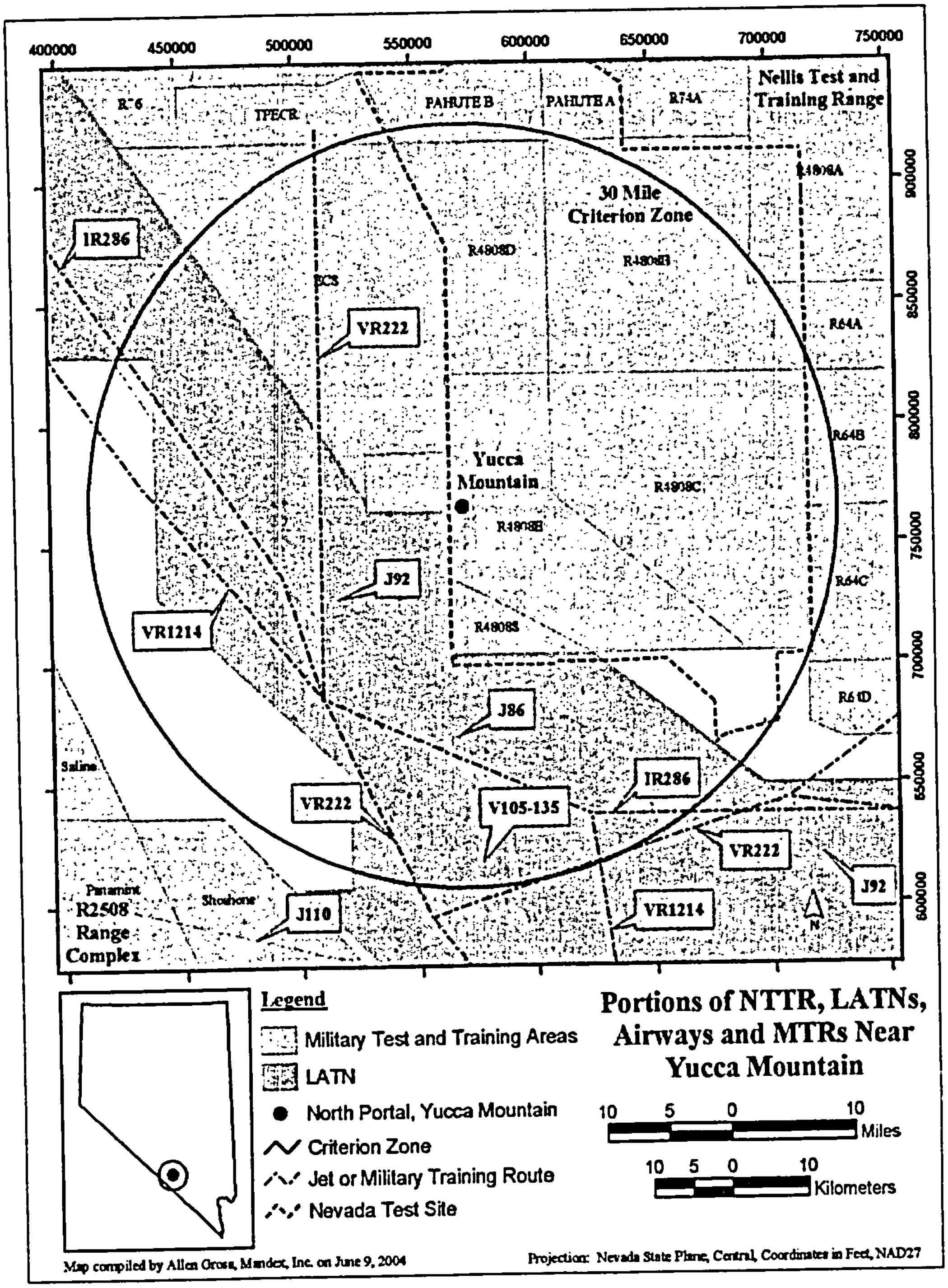

Figure 4.1.3-1. Portions of Nevada Test and Training Range (NTTR), Federal Airways, Military Training Routes (MTRs), and Low-Altitude Training Navigation Area (LATNs) (Adopted from Bechtel SAIC Company, LLC, 2002b, Figure 16)

4.1.3.2.1.2.2 Nevada Test and Training Range

Nevada Test and Training Range consists of airspace, land, and infrastructure for use by the military. The airspace and land are divided into restricted areas and military operating areas. The restricted areas (airspaces) are R–4806E, R–4806W, R–4807A, R–4807B, and R–4809; however, restricted area R–4809A is controlled by DOE and is not a part of the Nevada Test and Training Range. There are two military operating areas called Reveille and Desert.

The restricted areas are divided into North Range and South Range separated by the Nevada Test Site. Restricted airspaces R–4807A, R–4807B, and R–4809 belong to the North Range. The North Range contains three electronic combat ranges (Tonopah, Tolicha Peak, and Electronic Combat South), four unmanned weapons delivery subranges, Tonopah Test Range, and Pahute Mesa area, which is operated by DOE.

Restricted area R–4807A includes the 70 Series ranges, Tolicha Peak Electronic Combat, and Electronic Combat South Ranges. The 70 Series ranges are divided into several additional subranges, the closest ones with tactical targets (Ranges 74B and 74C) are approximately 58 km [36 mi] from the potential site for the surface facilities (Bechtel SAIC Company, LLC, 2002b). The closest boundary of the Electronic Combat South Range is approximately 8 km [5 mi] from the site for the North Portal. It is a manned electronic combat threat simulator range and does not involve bombing ground targets or dropping of any ordnance (Bechtel SAIC Company, LLC, 2002b). Caesar corridor, 4,267 m [14,000 ft] above mean sea level, overlies the Electronic Combat South Range and is used for recovery from the northern ranges to Nellis Air Force Base. Tolicha Peak Electronic Combat range is located at the southwest corner of R–4807A. It is a manned combat threat simulator range. No ordnance dropping is permitted there (Bechtel SAIC Company, LLC, 2002b).

Restricted airspace R–4807B (Pahute Mesa) is used as an annex to the Nevada Test Site by DOE. The U.S. Air Force is allowed to use this airspace for overflight. The closest boundary of R–4807B is approximately 48 km [30 mi] from the North Portal area (Bechtel SAIC Company, LLC, 2002b).

R–4809 contains the Tonopah Electronic Combat range. The Tonopah Electronic Combat Range is also a manned electronic combat threat simulator range located approximately 79 km [49 mi] from the North Portal area (Bechtel SAIC Company, LLC, 2002b). No ordnance dropping is permitted within this range. The Tonopah Test Range Airfield is located within this range and can be used for diverting aircraft experiencing in-flight emergencies. DOE controls the flight activities in this restricted airspace (Bechtel SAIC Company, LLC, 2002b).

The South Range is subdivided into restricted areas R–4806E and R–4806W. R–4806E is used primarily for air-to-air training, and the closest boundary is approximately 100 km [62 mi] from the North Portal (Bechtel SAIC Company, LLC, 2002b). R–4806W contains the 60 Series ranges used for conventional bombing and for gunnery testing and training. Additionally, the U.S. Air Force Thunderbirds Demonstration Squadron frequently practices in one of those ranges. The closest boundary of these ranges to the North Portal is approximately 43 km [27 mi] (Bechtel SAIC Company, LLC, 2002b).

4.1.3.2.1.2.3 R–2508 Range Complex

The airspace of R–2508 Complex Including China Lake Naval Weapons Center is located west and southwest of the potential repository site. The airspace and associated land are currently used and managed by Edwards Air Force Base, National Training Center, Fort Irwin, and Naval Air Warfare Center Weapons Division, China Lake. The closest boundary of this complex is approximately 58 km [36 mi] from the North Portal (Bechtel SAIC Company, LLC, 2002b).

4.1.3.2.1.2.4 Airspace Supporting Nevada Test and Training Range

There are several airspace support activities at Nevada Test and Training Range. These activities include (i) Low-Altitude Training Navigation areas, (ii) military training routes (IR–286, VR–222, VR–1214, IR–279, and IR–282), and (iii) air refueling tracks.

Low-Altitude Training Navigation areas are located east and southwest of the Nevada Test and Training Range for use by A–10s and helicopters to practice random selection of navigational points and low-altitude tactical formation flying between 33 and 457 m [100 and 1,500 ft] above ground level. The Low-Altitude Training Navigation area southwest of the Nevada Test and Training Range is approximately 1.6 km [1 mi] from the North Portal. The U.S. Air Force uses Low-Altitude Training Navigation areas when airspace within the Nevada Test and Training Range is not available for this type of training. Approximately 30 to 35 A–10 sorties are conducted weekly in the southwest Low-Altitude Training Navigation area (Bechtel SAIC Company, LLC, 2002b).

Military training routes IR–286, VR–222, and VR–1214 are close to the North Portal area. IR–286 is 30 km {16 nautical mi [18.4 statute mi]} wide (Bechtel SAIC Company, LLC, 2002b). The closest edge of this route is approximately 8 km [5 mi] from the North Portal area. Approximately 21 annual sorties use this route (Bechtel SAIC Company, LLC, 2002b; U.S. Air Force, 1999). VR–222 is 19 km {10 nautical mi [11.6 statute mi]} wide (Bechtel SAIC Company, LLC, 2002b). The closest edge is approximately 6.4 km [4 mi] from the North Portal area. Approximately 550 annual sorties are estimated to use this route (Bechtel SAIC Company, LLC, 2002b; U.S. Air Force, 1999). VR–1214 is 19 km {10 nautical mi [11.6 statute mi]} wide (Bechtel SAIC Company, LLC, 2002b). The North Portal area is approximately 21 km [13 mi] from the closest edge of this route. The last segment of IR–279 enters restricted airspace R–4809. Approximately 155 sorties use this route annually (Bechtel SAIC Company, LLC, 2002b; U.S. Air Force, 1999). Approximately 12 sorties annually use route IR–282 (Bechtel SAIC Company, LLC, 2002b; U.S. Air Force, 1999). The last segment of this route enters restricted airspace R–4807A. Bechtel SAIC Company, LLC (2002b) did not provide information on the distances of these two military training routes from the North Portal area.

Bechtel SAIC Company, LLC (2002b) identified three air refueling tracks within the 160-km [100-mi] region that are used to support activities in Nevada Test and Training Range. The closest edge of any of these refueling tracks is 126 km [78 mi] from the North Portal area.

4.1.3.2.1.2.5 Airports and Airfields

Bechtel SAIC Company, LLC (2002b) listed all the airports within 160 km [100 mi] of the North Portal of the potential repository at Yucca Mountain. Airports and airfields with a high volume of traffic and within reasonable proximity to the potential repository site have been discussed with more details about flight operations. Discussions of flight operations are given for Indian Springs Air Force Auxiliary Field, Tonopah Test Range Airfield, Nellis Air Force Base, Desert Rock Airport, Pahute Mesa Airstrip, Yucca Airstrip, Beatty Airport, Jackass Aeropark, Furnace Creek Airport, Imvite Airfield, McCarran International Airport, and North Las Vegas Airport.

Nellis Air Force Base is approximately 145 km [90 mi] from the North Portal area. Operations (takeoffs and landings) totaling 62,421 took place at Nellis Air Force Base in 2001. Indian Springs Air Force Auxiliary Field is approximately 72 km [45 mi] from the North Portal area and is located on the southern boundary of R–4806. It provides basing for operations for unmanned aerial vehicles and support for aircraft staging. It is also used as an emergency/divert base for Nevada Test and Training Range operations and is the primary training base for the Thunderbirds Air Demonstration Squadron. Bechtel SAIC Company, LLC (2002b) states, "... *the flight activity at this airfield can change as new test and development programs are introduced.*" Two hundred operations took place at Tonopah Test Range Airfield in 2001. This airfield is approximately 106 km [66 mi] from the North Portal area.

McCarran International Airport is approximately 143 km [89 mi] east-southeast of the North Portal, having 476,511 total annual operations that include 281,214 air carriers; 71,998 air taxis; 15,777 local aircraft; 89,038 itinerant private aircraft; and 18,484 military aircraft operations. North Las Vegas Airport is approximately 132 km [82 mi] east-southeast of the North Portal. Annual operations include 77,559 air taxis; 116,264 local aircraft; 81,479 itinerant private aircraft; and 84 military aircraft operations totaling 275,386. Beatty Airport is approximately 34 km [21 mi] west of the North Portal and has 1,005 annual operations. The Jackass Aeropark, located approximately 24 km [15 mi] from the North Portal, has 604 operations annually. The Furnace Creek Airport is located approximately 60 km [37 mi] from the North Portal with annual operations totaling 10,200. Imvite Airfield, owned by a division of Floridin Company, is approximately 45 km [28 mi] south of the North Portal. Currently it is inactive and had zero reported operations (Bechtel SAIC Company, LLC, 2002b).

Desert Rock Airport is approximately 43 km [27 mi] from the North Portal. The runway is oriented in such a way that landings and takeoffs are toward the northeast/southwest. Based on information from the DOE Airspace office, 330 operations have taken place each year since 1995. Pahute Mesa Airstrip is approximately 29 km [18 mi] from the North Portal with an estimated 80 operations annually. The Yucca Airstrip has not been used since 1995 (Bechtel SAIC Company, LLC, 2002b).

4.1.3.2.1.2.6 Federal Airways

Bechtel SAIC Company, LLC (2002b) listed all the airways within 160 km [100 mi] of the North Portal of the potential repository at Yucca Mountain. Only two Victor routes, V105 and V135, and two jet routes, J86 and J92, are within 32 km [20 mi] of the North Portal area.

Victor routes V105–V135 begin south of the Nevada Test Site and head northwest, paralleling the Nevada Test and Training Range and then split. V105 continues to Reno, Nevada. V135 terminates at Tonopah Airport. These airways are used by commercial air traffic between Las Vegas and Reno and other airports in the southwestern and northwestern United States. The nearest point of these airways to the North Portal is approximately 11 km [7 mi] (Bechtel SAIC Company, LLC, 2002b). The airway V105–V135 is for air traffic below 5,400 m [18,000 ft] mean sea level.

There is a discrepancy about the reported width of airway V105–V135 among CRWMS M&O (1999c), Bechtel SAIC Company, LLC (2002b), and CRWMS M&O (2003a). According to CRWMS M&O (1999c), airway V105–V135 is 16 km [10 mi] wide. The nearest edge of this airway is 17.6 km [11 statute mi] away from the potential repository surface facilities. However, Table 2 of Bechtel SAIC Company, LLC (2002b) states that the centerline of V105–V135 is 25.7 km [16 mi] from the North Portal. Additionally, the closest point on V105–135 is approximately 11.3 km [7 mi] from the North Portal. Therefore, the width of this airway is 28.8 km [18 mi]. However, CRWMS M&O (2003a, Section 5.5.2 and Assumption 3.16) assumed that the width of V105–V135 airway is 38.6 km [24 mi]. 14 CFR 71.75(b)(1) states the width of each federal airway is 12.8 km [8 mi] unless specified otherwise. The 1996 Federal Radionavigational Plan (U.S. Department of Transportation and U.S. Department of Defense, 1997) also supports this 12.8-km [8-mi] width of Federal Victor airways. DOE should clarify the discrepancies and provide the bases for the assumption of the width of V105–V135 airway used in the analysis.

Jet route J86 departs from McCarran International Airport and continues toward the Beatty Very High-Frequency Omnidirectional Range Station and/or the Tactical Air Navigation where it joins with Jet route J92. 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 (2000a) states the commercial air traffic generally is jetliners that fly above 5,400 m [18,000 ft] mean sea level through J92. The centerline of airway J86 is 28.9 km [18 mi] from the North Portal. According to Bechtel SAIC Company, LLC (2002b), the Federal Aviation Administration allows flights to use the entire width of the airspace between R–2508 and R–4808/R–4807. Therefore, the closest distance between the North Portal and the boundary is 11 km [7 mi]. Jet route J92 goes to Reno, Nevada. The centerline of the route is approximately 24.2 km [15 mi] from the North Portal. Because the Federal Aviation Administration allows flights to use the entire width of the airspace between R–2508 and R–4808/R–4807 (Bechtel SAIC Company, LLC, 2002b), aircraft flying in this route can be as close as 11 km [7 mi] to the North Portal. Jet route J92 overlies Victor Route V105 and is used by air traffic above 5,400 m [18,000 ft] mean sea level (CRWMS M&O, 2000a).

Again, there is a discrepancy about the reported width of airways J86 and J92 between Bechtel SAIC Company, LLC (2002b) and CRWMS M&O (2003a). CRWMS M&O (2003a, Section 5.5.2 and Assumption 3.16) assumed that the width of J86 and J92 airways is 38.6 km [24 mi]. DOE should clarify the discrepancy and provide the bases for the assumption in CRWMS M&O (2003a).

4.1.3.2.1.2.7 Other Activities

No launches of ground-to-ground missiles have been conducted in Area 26 of the Nevada Test Site since June 2000. Area 26 is approximately 23 km [14 mi] from the North Portal. Bechtel SAIC Company, LLC (2002b) stated no launches are anticipated in the near future.

The Kistler Aerospace Corporation is developing a reusable space launch vehicle, called K–1, and has plans to use part of Area 18 of the Nevada Test Site for operations. Once the facility is fully operational, a fleet of five K–1 vehicles will have a maximum 52 annual flights (U.S. Department of Transportation, 2002). DOE should update Bechtel SAIC Company, LLC (2002b) and estimate the potential hazard onto the proposed facilities from these flights.

Helicopters routinely fly in most areas within the restricted airspace of the Nevada Test Site. Assumption 3.22 of CRWMS M&O (2003a) states that the helicopter routes maintain a separation distance of at least 0.4 km [0.25 mi] from the surface facilities of the potential repository.

4.1.3.2.1.2.8 Summary

The NRC staff has reviewed Bechtel SAIC Company, LLC (2002b) and CRWMS M&O (2003a) as the DOE response to Preclosure Agreement PRE.03.01. During the review, the NRC staff identified topics that may need to be addressed in the potential license application. The NRC staff informed DOE regarding concerns related to aircraft hazards to the potential repository facilities (NRC, 2003). Additionally, a technical exchange took place between DOE and NRC on aircraft hazards (Schlueter, 2003c). Some information from the military regarding potential activities near the repository site may be sensitive and should be handled accordingly. The NRC staff concerns are as follows:

- A significant portion of the information regarding the Nevada Test and Training Range and associated activities, presented in Bechtel SAIC Company, LLC (2002b), has been acquired from the U.S. Air Force (1999) and is at least 5 years old. The number and type of aircraft flown, mode of flight, and other data change over time, so it is important to use the latest data available. Projected estimates also are needed in cases where there is evidence of data trending, because current conditions may not be applicable throughout the operating period. DOE should consider updating the available information used in aircraft crash hazard analysis in a potential license application.

- Section 5.1.4, Ordnance Used at the Nevada Test and Training Range, of Bechtel SAIC Company, LLC (2002b) states, "the range operating agency must ensure that weapon safety footprints exist for all aircraft, weapons, and tactics authorized for a given target and event on the range." Additionally, Section 6.3.1.1.5, Ordnance, concludes that instructions from operating and controlling agencies of the Nevada Test and Training Range provide assurance that weapon training activities would not pose a credible hazard to the potential repository operations. Also, Section 6.2.1.3, Ordnance Fired from Aircraft, indicates there are procedures for dealing with safety footprints that may extend beyond the boundaries of the range to be employed. In the event that an off-range hazard cannot be eliminated, the procedure allows the range operating agency to assess the hazard and make an informed decision on its acceptability. DOE should provide information regarding the safety instructions that would prohibit ordnance used in training activities from impacting any safety-related structures, systems, and components at the potential repository. DOE should determine how this information translates into the probability of ordnance impacting the surface facilities. DOE should demonstrate that any structures, systems, and components important to safety would not be affected by an ordnance accidently delivered outside the intended region. The information should include the safety footprint information superimposed on these locations of the target sites. An alternate approach may be to map historical data of actual off-range ordnance deliveries and use the data to estimate the probability of an ordnance impacting the proposed surface facilities.

- Bechtel SAIC Company, LLC (2002b) does not provide any information regarding the number of each type of weapon used annually and safety precautions taken to ensure that weapons do not fly or impact outside the intended region(s) of discharge and impact. In addition, Section 6.2.1.3, Ordnance Fired from Aircraft, does not provide any information on testing cruise missiles, including the tests performed at Tonopah Test Range. DOE should provide the number of each type of weapon used annually, as well as the flight paths for air-to-ground ordnance (rockets and cruise missiles) with respect to the potential repository location.

- Section 6.2.1.1 of Bechtel SAIC Company, LLC (2002b), Training More Than 30 Miles from the North Portal at Yucca Mountain, states, "... range safety practices will preclude the activities from having an adverse impact on Yucca Mountain Project operations." However, no information has been provided to substantiate the claim. DOE should provide information about the range safety practices that will preclude the activities from having an adverse effect on Yucca Mountain Project operations.

- No information has been provided on the flight paths of aircraft for recovery to Nellis Air Force Base or Indian Springs Air Force Auxiliary Field with hung ordnance in Bechtel SAIC Company, LLC (2002b). Additionally, DOE has not clarified what is meant by "*critical inflight emergencies*" that would allow an aircraft with hung ordnance to transit through restricted airspace/area R–4808N. DOE should provide necessary information on the flight paths of aircraft with hung ordnance and clarify what constitutes a critical inflight emergency. Additionally, DOE should specify the safety precautions and actions to be taken for hung ordnance and for an aircraft carrying hung ordnance in the vicinity of the potential repository location.

- Section 6.2.2.2, Military Training Routes, of Bechtel SAIC Company, LLC (2002b) concludes *aircraft flying on military training routes located more than 32 km [20 mi] from* the North Portal at Yucca Mountain do not pose a hazard to that facility. The argument is based on comparison with the proximity criterion (b) NRC (1981a), however, the proximity criterion only says that the annual aircraft crash hazard from the military training routes will be less than 10^{-7}. This estimated annual frequency will be a component of the cumulative crash hazard of the proposed facilities after taking into account all potential sources. DOE should include the contribution of the aircraft flying in the military training routes in estimating the cumulative crash hazard.

- Numerous statements in Appendix G of Bechtel SAIC Company, LLC (2002b) are presented without any basis or data. For example,

  "... it is expected that in a controllable situation at high altitudes, the pilot would eject between 3,048 and 4,572 m [10,000 and 15,000 ft] above mean sea level {approximately 1,524 and 3,048 m [5,000 and 10,000 ft] above ground level assuming a ground elevation of 1,524 m [5,000 ft] after unsuccessful restart." No basis for such an expectation has been presented.

  "... if the aircraft is at a high altitude and not in vertical descent, the pilot will regain control and a crash is averted." No basis for such an expectation has been presented.

"... a disabling event at high altitudes would result in either immediate descent of the aircraft with pilot ejection or a controlled descent, providing time for pilot action prior to ejection." No basis has been provided.

"... [a]n engine fire could result in an immediate pilot ejection. It is expected that this would result in an in-flight explosion of the aircraft or a nearby crash of the aircraft depending on its altitude, speed, and direction." No actuarial information or rationale has been presented to justify such expectations.

Appendix G states pilot errors resulting in crashes are caused by midair collisions with other aircraft or collisions with the ground. This conclusion implies crashes caused by pilots losing situational and/or positional awarenesses might not have been included.

DOE should provide the supporting technical basis for the previous statements in Appendix G of the report. Further, the technical bases should consider, as appropriate, potential deviations from the expected standards or norms that can place people, equipment, and systems at risk from aircraft hazards at a potential repository at Yucca Mountain. Deviations such as those caused by unwanted actions or inactions that arise from problems in sequencing, timing, knowledge, interfaces, and procedures need to be evaluated.

- Several sections of Bechtel SAIC Company, LLC (2002b) (e.g., Appendix G; Section 6.3.1.1.2, Desert Military Operating Areas; Section 6.3.1.1.3, 70 Series Ranges; Section 6.3.1.1.4, Electronic Combat Ranges; and Section 6.3.1.1.6, 60 Series Ranges), state that a pilot experiencing problems would direct the aircraft away from the Yucca Mountain site. For example, Section 6.3.1.1.2, Desert Military Operating Areas, states, "... if the aircraft has glide capability and, depending on the altitude, the pilot will direct the aircraft away from the range boundaries to a suitable ejection area within one of the valleys located in the Coyote Military Operating Areas; the pilot would eject and the aircraft most likely would crash into the surrounding mountains of the Coyote Military Operating Areas." Similarly, Section 6.3.1.1.4, Electronic Combat Ranges, states, "... pilots preparing to eject would avoid the mountainous western and southern areas resulting in the aircraft moving away from Yucca Mountain." Section 6.3.1.1.3, 70 Series Ranges, states, "... range 75E/W has a mountain range that borders the eastern boundary and several radioactive contaminated areas adjacent to the southern border (Pahute Mesa) that make those areas unattractive for pilot ejection." Section 6.3.1.1.6, 60 Series Ranges, states, "... if the aircraft has glide capability and depending on the altitude, the pilot will direct the aircraft away from mountainous terrain." It also states "...a suitable ejection area is within the flatter terrain found in Indian Springs Valley." Pilot actions in ejection site selection and aircraft direction prior to ejection are achievable if there is sufficient time and control of the aircraft. Emergency procedures require pilots to perform numerous actions that may encroach on the pilot's ability to exercise the appropriate ejection options. Even with sufficient time and control, other factors (e.g., weather, visibility, or knowledge and recognition of ground features) may limit the ejection options available to the pilot. DOE should determine the likelihood of unwanted actions or inactions on the part of the pilot that arise from problems in sequencing, timing, knowledge, interfaces, and/or procedures that may result in deviations from what is expected of the pilot during inflight emergencies that may place people, equipment, and systems at risks from aircraft hazards at the potential repository at Yucca Mountain.

- It is not clear for which year the flight information given in Table 1 of Bechtel SAIC Company, LLC (2002b) was compiled. DOE should clarify the year and source of information from which the number of flights in each military training route was estimated. Similarly, other information should be identified by year. DOE should ensure that it is using the most current available information.

- Assumption 3.12 of CRWMS M&O (2003a) states aircraft missions in Electronic Combat Range South and in the Caesar Corridor are "an extension in space of the missions" over the Nevada Test Site. DOE should provide a basis for the rationale that aircraft crossing the Nevada Test Site would also pass through Electronic Combat Range South. For example, confirmatory information from the U.S. Air Force could be used to support the assumption that missions in Electronic Combat Range South and in the Caesar Corridor are extensions of the airspace of the missions over the Nevada Test Site.

- Assumption 3.5 of CRWMS M&O (2003a) states, "... [f]lying the shortest distance between two points is the most efficient way to cross the Nevada Test Site." It is not clear whether actual operational planning of the U.S. Air Force has been checked to arrive at the conclusion. The path taken by an aircraft while flying in a restricted area depends on the mission with associated planning of the flight path(s). DOE should provide the basis for this assumption.

- The potential repository lies underneath restricted airspace R–4808E. Additionally, the potential repository is close to other restricted airspaces, such as the Electronic Combat Range South. Aircraft are known to engage in different maneuvers inside a restricted airspace. Bechtel SAIC Company, LLC (2002b) did not provide sufficient information to establish the possible flight paths and mode of flight in the airspaces near the potential repository. Flying characteristics (mode and paths of flights) in an area would depend on flight planners who develop the flight plans and pilots who fly through that area. Specific information (e.g., from U.S. Air Force records) should be provided to justify this assumption.

- Items included in the "dropped objects" category in Section 2.4 of CRWMS M&O (2003a) are never defined. If the definition includes any objects that can explode (e.g., a bomb) or ignite (e.g., an external fuel tank), contribution of the overpressure generated due to explosion and/or the thermal energy may need to be considered by appropriately enlarging the effective area of a ground structure. DOE should clarify what is meant by dropped objects. Additionally, DOE should clarify whether stressful activities such as maneuvers during combat training have been considered while making the assumption that the drop rate would be uniform along the flight path.

- It is not clear what is meant by "preferred altitude of ejection" {below approximately 3,048 m [10,000 ft] above ground level} in Assumption 3.11 of CRWMS M&O (2003a). DOE should provide documented evidence to establish whether this preferred altitude is recommended by the aircraft manufacturers or U.S. Air Force for ejection, or only preferred by pilots for ejection.

- Basis for Assumption 3.15 in CRWMS M&O (2003a) that aircraft on the Nevada Test and Training Range flying near the proposed surface facility would be represented by "small attack, fighter, trainer aircraft" is not provided. DOE should clarify whether trainer aircraft fly routinely near the potential repository and identify their missions. DOE should clarify also why the category of small military aircraft (all small attack, fighter, and trainer aircraft) would be more conservative when crash rates for F–16s, all single-engine, and all attack and fighter aircraft are higher. DOE should clarify whether uncertainties associated with the determination of the aircraft type flying in the vicinity of the proposed surface facilities have been appropriately considered in estimating the effective area of the buildings and in selecting the appropriate crash rate for the aircraft in the analysis.

- Basis for using a 1-week interval of flight data given in CRWMS M&O (2003a, Table 9) to establish the annual number of flights and concluding that the restricted airspace R–4808S is not heavily used by civilian air traffic (Assumption 3.16) is not provided. DOE should provide appropriate bases and should justify how the average of 1-week flight data would be representative of flights through this corridor (CRWMS M&O, 2003a, Assumption 3.19). Additionally, DOE should clarify whether uncertainties in flight information through this corridor have been appropriately considered in the analysis.

- DOE should explain the rationale for assuming the width of the aviation corridor to the southwest of Yucca Mountain to be equal to 38.4 km [24 mi] in CRWMS M&O (2003a). DOE should clarify whether this assumed width belongs to Federal airway V105–135, J86, J92, VR1214, or IR286 (Bechtel SAIC Company, LLC, 2002b, Figure 16). DOE should clarify whether the assumed width of the airway is the same as used by the Federal Aviation Administration, as discussed previously.

- Assumption 3.16 of CRWMS M&O (2003a) states, "... air traffic near and with R–4808S tends toward the very high-frequency omnidirectional range and tactical air navigation station south of Beatty." DOE should provide the source and rational for this assumption.

- Assumption 3.20 of CRWMS M&O (2003a) states military aircraft flying on the military training routes and low-altitude tactical navigation areas pose a negligible hazard to proposed surface facilities. DOE should provide the bases for this assumption. Additionally, DOE should clarify whether the zooming maneuvers conducted by the military pilots facing inflight emergencies were considered in developing this assumption. A typical zooming maneuver takes the aircraft to a higher altitude before beginning the glide and results in a potentially larger crash range.

- Assumptions 3.20 and 3.21 of CRWMS M&O (2003a) state that civilian aircraft flying at 360 m [1,200 ft] above ground level and below 3,000 m [10,000 ft] above mean sea level, irrespective of distance from the proposed surface facilities, will not pose a credible hazard to the proposed surface facilities. DOE should provide the rationale for these assumptions. Additionally, DOE should provide the conversion from mean sea level to above ground level for flights near the potential repository.

- CRWMS M&O (2003a, Section 5.5.1) has assumed that the average number of flights in years 1999 through 2002 would be representative for estimating the annual crash frequency onto the proposed surface facilities. DOE should provide the rationale for this assumption. Additionally, DOE should clarify whether uncertainties in the number of annual flights would be appropriately considered in the analysis.

- Section 5.8, Commercial Rocket Launch and Retrieval, of Bechtel SAIC Company, LLC (2002b) should be revised because Kistler Aerospace Corporation has received approval from the Federal Aviation Administration for operations in Area 18 of the Nevada Test Site. DOE should demonstrate that operations by Kistler Aerospace Corporation in Area 18 would not pose any undue hazard to the potential repository.

- Bechtel SAIC Company, LLC (2002b) did not provide the distance of the North Portal from the nearest points on the military training routes IR–279 and IR–282. DOE should provide this information.

- DOE should clarify the discrepancy about the width of V105–V135 airway among CRWMS M&O (1999c), Bechtel SAIC Company, LLC (2002b), and CRWMS M&O (2003a). Additionally, DOE should provide the bases for the assumption of the width of V105–V135 airway used in the analysis presented in CRWMS M&O (2003a).

- DOE should clarify the discrepancy about the width of airways J86 and J92, reported in Bechtel SAIC Company, LLC (2002b) and CRWMS M&O (2003a). Additionally, DOE should provide the bases for the assumption made in CRWMS M&O (2003a).

- Assumption 3.22 of CRWMS M&O (2003a) states that a separation distance of at least 0.4 km [0.25 mi] will be maintained by all helicopter flights from the surface facilities of the potential repository. This distance should be verified.

4.1.3.2.1.3 Determination of Frequency or Probability of Occurrence of Hazards and Initiating Events

Commercial aircraft use both McCarran International and North Las Vegas Airports. Limited chartered aircraft use Tonopah Airport (CRWMS M&O, 1999c). All three airports are more than 48 km [30 mi] from the potential 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 potential 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). DOE estimated that the crash probability at the potential site from aircraft takeoff and landing at these three airports would be negligible.

The NRC staff reviewed Bechtel SAIC Company, LLC (2002b) and CRWMS M&O (2003a) as the DOE response to preclosure agreement PRE.03.01. DOE recommended in the cover letter of CRWMS M&O (2003a) that the analysis presented in CRWMS M&O (2003a) will be updated as the design of the facilities are evolving and NRC should review the analysis only on the methodology (Ziegler, 2003). Consequently, the staff reviewed the information and analysis presented in CRWMS M&O (2003a) principally on the methodology. During the review, NRC staff identified potential issues that may need to be addressed in the license application. The NRC staff has informed DOE regarding these concerns related to aircraft hazards to the potential repository facilities (Schlueter, 2003a,b). Additionally, a technical exchange took place between DOE and NRC on aircraft hazards (Schlueter, 2003c). Some information from the military regarding potential activities near the potential repository site may be sensitive and should be handled accordingly. The NRC staff concerns are as follows.

- In Section 5.3 of CRWMS M&O (2003a), crash rates of aircraft considered in the analysis are limited only to normal inflight mode. DOE should provide the rationale, taking into account information on flight characteristics of the aircraft flying in the vicinity of the proposed surface facilities, for considering crash rates limited to aircraft flying only in normal in-flight mode.

- Assumption 3.12 of CRWMS M&O (2003a) states, "... because EC South is at least several miles from the North Portal, the aircraft crash hazard is insensitive to flight activity in EC South." DOE should provide the basis for this assumption. In addition, DOE should provide detailed information on the flight activities, flight mode, and aircraft type(s) flying in Electronic Combat Range South that have been considered to arrive at the assumption. DOE should identify whether crash range of each type of aircraft, type(s) of aircraft that fly in Electronic Combat Range South, and missions conducted have been taken into account.

- Assumption 3.13 in CRWMS M&O (2003a) assumed a general aviation pilot would at all times steer away from the potential Yucca Mountain facilities. DOE should provide the basis for this assumption.

- CRWMS M&O (2003a) assumed in Assumption 3.14 that "an impact into a support area would not jeopardize the integrity of the process zone" and, therefore, the support areas of the buildings need not be considered in estimating the effective areas of the buildings. DOE should provide the basis for the assumption. Information should include whether skid of the aircraft involving "ploughing" the support facilities was considered.

- Assumption 3.14 of CRWMS M&O (2003a) neglects the effective areas represented by the "... transportation casks inside the Transporter Receipt Building or in transit between buildings, and waste packages in shielded transporters heading underground." DOE should provide the basis of this assumption. Information should include whether frequency of shipment of waste packages for emplacement has been considered along with the skid of the aircraft. Additionally, information should clarify why the transportation casks inside the Transporter Receipt Building would provide insignificant effective area for estimation of the annual crash frequency when the Transporter Receipt Building itself was not considered. As the surface facility layout is still evolving, DOE should use the final surface facility layout in the analysis to be submitted in the potential license application.

- It is not clear whether the rail yard or the area used for casks waiting to be processed have been considered in estimating the annual crash frequency. DOE should clarify whether the rail yard and the cask waiting area have been considered in estimating the annual crash frequency.

- Section 6.1, Qualitative Approach to Hazard Screening, of Bechtel SAIC Company, LLC (2002b) states DOE "screened out event sequences considered not credible ..." using "... criteria based on qualitative and quantitative bases that include distance, flight characteristics and pilot actions." It is not clear to the staff what quantitative information has been used to characterize flight activities and pilot actions. No information has been presented on the mode of flight, which is an essential element of flight characteristics, used to determine the appropriate crash rate for a particular aircraft (DOE, 1996;

Kimura, et al., 1996). Additionally, no initiating events and event sequences have been identified in the report. Therefore, it is not clear how some event sequences were eliminated without information on the frequency of occurrence or estimated dose consequences. DOE should identify the initiating events and event sequences and provide an analysis using Probabilistic Safety Assessment methodology, including the estimated frequency of occurrences and associated uncertainties, that have been used to eliminate potential event sequences. In addition, DOE should identify the qualitative (description and characteristics of the facilities and equipment, distance of the activity from the North Portal, identification of initiating events that could occur during the activity, identification of probable event sequences following the initiating event, and determination of the credibility of these event sequences impacting the repository facilities and operations) and quantitative (distance, flight characteristics, and pilot action) parameters used in assessing potential hazards for each case.

4.1.3.2.1.4 Technical Basis for Inclusion or Exclusion of Specific Hazards and Initiating Events

DOE is in the process of collecting additional information regarding flight activities in the vicinity of the potential repository facilities. Based on this information and taking into account the final design of the facilities, DOE has stated that it will update the analysis.

4.1.3.2.2 Tornado Wind

4.1.3.2.2.1 Technical Basis and Assumptions for Methods Used for Identification of Hazards and Initiating Events

DOE submitted a report CRWMS M&O (2003b) as a response to Key Technical Agreement PRE.03.02. This report replaces the CRWMS M&O (1999d) report. CRWMS M&O (2003b) provided an analysis that establishes the design basis wind speeds for the straight wind and tornadoes. Information contained in this report on development of design basis straight and tornado wind speeds is classified as Official Use Only.

Staff reviewed CRWMS M&O (2003b), documents referenced therein, and other technical documents to assess the information on the methodology, technical bases, and assumptions used in developing the design basis straight and tornado wind speed (NRC, 2004). Staff review of the information finds that the design basis straight wind speed is based on limited site-specific data available for only 4 years. The region is identified in SEI/ASCE 7–02 (Structural Engineering Institute/American Society of Civil Engineers, 2003) to be a special region requiring site-specific data to account for local topographical conditions. Therefore, DOE should use site-specific data for additional years to better quantify the design basis wind speed for structures, systems, and components important to safety.

4.1.3.2.2.2 Use of Relevant Data for Identification of Site-Specific Hazards and Initiating Events

CRWMS M&O (2003b) provided an analysis to develop the design basis straight and tornado wind speed. The design basis straight wind was developed based on limited site-specific data available for only 4 years (CRWMS M&O, 1997a) and compared with the DOE Standard DOE–STD–1020–2002 (DOE, 2002). Information contained in this report on development of design basis straight and tornado wind speeds is classified as Official Use Only. The region is identified in SEI/ASCE 7–02 (Structural Engineering Institute/American Society of Civil Engineers, 2003) to be a special region requiring site-specific data to account for local topographical conditions. Therefore, DOE should use site-specific data, qualified in accordance with 10 CFR Part 63, Subpart G, for additional years to better quantify the design basis wind speed for structures, systems, and components important to safety.

4.1.3.2.2.3 Determination of Frequency or Probability of Occurrence of Hazards and Initiating Events

CRWMS M&O (2003b) provided an analysis to develop the design basis straight and tornado wind speed. Information contained in this report on development of design basis straight and tornado wind speeds is classified as Official Use Only. The design basis straight wind speed for structures, systems, and components important to safety was developed for a 50-year return period (CRWMS M&O, 2003b). This return period contradicts the information presented in CRWMS M&O (1999a,b), DOE (2001b), and Bechtel SAIC Company, LLC (2002a) for extreme wind (Hazard number 10 in Table 4.1.3-2) and also the screening criteria used. DOE should provide a rationale for a 50-year return period design basis wind speed for structures, systems, and components important to safety at the proposed surface facilities or use a return period that is commensurate with the safety functions of the proposed facilities.

4.1.3.2.2.4 Technical Basis for Inclusion or Exclusion of Specific Hazards and Initiating Events

Based on the discussion given in previous sections, DOE should provide a rationale to justify why use of a 50-year return period design basis wind speed for structures, systems, and components important to safety at the proposed surface facilities would be acceptable. Alternatively, DOE should use a return period that is commensurate with the safety functions of the proposed facilities. Additionally, the design basis straight wind speed is based on limited site-specific data. The region is identified by SEI/ASCE 7–02 (Structural Engineering Institute/American Society of Civil Engineers, 2003) to be a special region that requires site-specific data to account for local topographical conditions. Therefore, DOE should use site-specific data for additional years to better quantify the design basis wind speed for structures, systems, and components important to safety.

4.1.3.2.3 Tornado Missiles Hazard

4.1.3.2.3.1 Technical Basis and Assumptions for Methods Used for Identification of Hazards and Initiating Events

DOE submitted CRWMS M&O (2003b) as a response to Key Technical Agreement PRE.03.02. This report replaces CRWMS M&O (1999d). CRWMS M&O (2003b) provided an analysis that selected the design basis tornado missile spectrum and established a methodology to estimate the annual tornado missile impact probability on structures. Information contained in this report on development of tornado missile spectrum and tornado missile impact probability is classified as Official Use Only. Staff reviewed CRWMS M&O (2003b) and has no further questions on the information concerning the design basis tornado missiles (NRC, 2004).

4.1.3.2.3.2 Use of Relevant Data for Identification of Site-Specific Hazards and Initiating Events

CRWMS M&O (2003b) submitted as a response to Key Technical Agreement PRE.03.02, provided an analysis that selected the design basis tornado missile spectrum. Selection of the tornado missile spectrum is based on Section 3.5.1.4, Missiles Generated by Natural Phenomena (NRC, 1981b). Information contained in this report on development of tornado missile spectrum and tornado missile impact probability is classified as Official Use Only. Staff concluded that CRWMS M&O (2003b) provided a methodology and technical bases to select credible tornado missile characteristics for structures, systems, and components important to safety that are sufficient for use in developing a potential license application (NRC, 2004).

4.1.3.2.3.3 Determination of Frequency or Probability of Occurrence of Hazards and Initiating Events

CRWMS M&O (2003b), provided an analysis to estimate the annual tornado missile impact probability on structures, systems, and components important to safety. Estimation of impact probability is based on Cramond, et al. (1987). Information contained in this report on development of tornado missile impact probability is classified as Official Use Only. Staff has reviewed the information given in the analysis and have no further questions at this time (NRC, 2004).

4.1.3.2.3.4 Technical Basis for Inclusion or Exclusion of Specific Hazards and Initiating Events

DOE has provided an analysis that is sufficient for use in developing a potential license application, 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.

4.1.3.2.4 Volcanic Hazards

4.1.3.2.4.1 Technical Basis and Assumptions for Methods Used for Identification of Hazards and Initiating Events

DOE used information from volcanoes in the western United States to conclude that any potential volcanic eruption in the Yucca Mountain region would deposit less than 3 cm [1.2 in] of tephra on surface facilities during the preclosure period (CRWMS M&O, 1999b). Thus, DOE 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 specified by the Uniform Building Code (International Conference of Building Officials, 1997). The NRC staff does not agree with DOE that 3 cm [1.2 in] is an upper bound of potential tephra-fall deposit thickness at the surface facilities site. Information in Section 4.1.3.3.2 of this report indicates that thicker deposits could occur from a future volcanic eruption located within approximately 10 km [6.2 mi] of the site. These deposits have the potential to exceed the minimum design load requirements specified by the Uniform Building Code (International Conference of Building Officials, 1997).

DOE asserts that the effects of a potential tephra-fall deposit on air circulation systems is bounded by the effects of a sandstorm (CRWMS M&O, 1999b). Effects of potential sandstorm hazards would be mitigated by an orderly facility shutdown during sandstorm events, to avoid the potential for air filter clogging (CRWMS M&O, 1999b). This analysis, however, does not address potentially important differences between windblown sand and volcanic eruption deposits. For example, airborne particle concentrations {i.e., particles finer than 0.1 mm [0.004 in]} can be elevated by factors of 10 in the years following a volcanic eruption (e.g., NRC, 1999). In contrast, sandstorms primarily consist of relatively larger diameter particles that do not remain suspended above the soil once a sandstorm has ended (e.g., Wiggs, 1997). Additionally, repository surface facility designs have not been finalized by DOE. Passive air circulation systems commonly associated with dry-cask storage facilities may be sensitive to blockages by tephra-fall deposits or to sustained, elevated airborne particle concentrations. Thus, the NRC staff does not agree with DOE that the effects of potential tephra-fall deposits on air circulation systems are simply bounded by the effects of a sandstorm hazard.

DOE concluded that the probability of a new volcano forming directly at the potential repository site is less than 1×10^{-6} per year (CRWMS M&O, 1999b). In this analysis, DOE evaluated only the probability of new volcano formation within the potential repository subsurface facility footprint, but did not consider the potential for a new volcano to form at the location of surface facilities. In addition, DOE did not consider the probability of a new volcano forming at a location outside the potential repository operations area, but sufficiently close so that lava flows could reach and potentially affect surface facilities. While DOE has not provided a specific analysis of lava flow hazards from a new volcano forming within the potential repository surface operations area, or in a location where lava flows may pose a hazard, analyses done for postclosure volcanic hazards (CRWMS M&O, 1999b; Connor, et al., 2000) suggest that this probability is also less than 1×10^{-6} per year. DOE should provide the technical basis to support this probability if it intends to exclude this hazard from the Category 2 event sequence list.

4.1.3.2.4.2 Use of Relevant Data for Identification of Site-Specific Hazards and Initiating Events

DOE analyzed possible hazards of volcanic ash (i.e., tephra) to the potential repository and concluded that future eruptions could form a maximum 3-cm [1.2-in]-thick deposit at the potential repository site. The 3-cm [1.2-in] thickness is based on the assumption that a possible volcanic eruption during the preclosure period would be located at least 150 km [94 mi] from the potential repository site [i.e., Perry and Crowe (1987)]. Although large volcanoes located this distance away from the potential repository site could create deposits less than 3 cm [1.2 in] thick (e.g., Hoblitt, et al., 1987), new volcanoes characteristic of the area within 20 km [12.5 mi] of the site could produce appreciably thicker deposits. Based on analyses presented in Section 5.1.2.2.4.1, the probability of a new volcano forming within approximately 10 km [6.3 mi] of the potential repository site exceeds 1×10^{-6} per year. 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 [1–39 in] thick (e.g., Sagar, 1997). These analog deposits appear reasonably comparable to poorly preserved deposits within several kilometers of Lathrop Wells volcano at the southern terminus of Yucca Mountain (e.g., Sagar, 1997). This topic was not discussed at the first technical exchange and management meeting for preclosure safety (Reamer, 2001). DOE has not presented information on the characteristics of tephra-fall deposits sufficient to support the analogy to sandstorms or to evaluate the potential effects on air circulation systems important to safety.

4.1.3.2.4.3 Determination of Frequency or Probability of Occurrence of Hazards and Initiating Events

For the analysis of potential natural hazards to the potential repository, DOE concluded 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 analyses 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 potential repository site (i.e., Perry and Crowe, 1987). DOE has not quantified the probability for this type of eruption to occur, but state that this probability is 1×10^{-6} per year (CRWMS M&O, 1999b).

DOE has not evaluated the probability for a basaltic volcanic eruption to occur at a location close enough to the potential repository site to possibly affect safe operation of surface facilities. Staff has performed an independent analysis of this probability for potential tephra-fall hazards. 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). These deposits increase in thickness to approximately 400 cm [157 in] within 1 km [0.6 mi] of the volcanic vent. Perry and Crowe (1987) also conclude that a 1-m [3.3-ft]-thick tephra deposit could occur approximately 3 km [1.9 mi] from a basaltic volcanic vent. As a first approximation, a new basaltic volcano located within 10 km [6.3 mi] of the potential repository site appears capable of creating a tephra deposit 1–100 cm [0.4–39 in] thick at the surface facility site.

Using the probability models in Connor, et al. (2002), the probability of a new volcano forming in a 10-km [6.2-mi]-long buffer zone around the potential repository site exceeds 1×10^{-6} per year for a range of credible recurrence rates and conceptual probability models. Thus, additional analyses appear warranted to evaluate the likelihood of a tephra-fall deposit exceeding the minimum design load requirements of 98 kg/m^2 [20 lb/ft^2] specified in the Uniform Building Code (International Conference of Building Officials, 1997).

Noncompacted, dry basaltic volcanic tephra has a bulk deposit density that can range 1,000–1,700 kg/m^3 [62–106 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 (e.g., Sarna-Wojcicki, et al., 1981). Thus, a 1–100-cm [0.4–39-in]-thick tephra deposit could result in a load of 10–1,700 kg/m^2 [2–348 lb/ft^2] when dry, which could increase to 20–3,400 kg/m^2 [4–697 lb/ft^2] if the deposit was wet. Most of these potential roof loads are significantly larger than assumed to occur during the DOE screening of volcanic ash-fall as an applicable natural hazard for the potential repository site (CRWMS M&O, 1999b). This topic was outside the scope of the first technical exchange and management meeting for preclosure safety (Reamer, 2001).

DOE has not evaluated the probability for a basaltic volcanic eruption to occur at a location close enough to the potential repository site for lava flows to possibly affect safe operation of surface facilities (CRWMS M&O, 1999b). The probability of potential lava-flow hazards depends on important assumptions regarding location and extent of surface facilities, area upslope of these facilities capable of potential volcanic activity, and distribution of potential lava-flow lengths (e.g., Volcanism Working Group, 1990). Current uncertainties on the number, age, and location of possible volcanic events in the Yucca Mountain area (e.g., Hill and Stamatakos, 2002) likely affect the probability estimates for new volcano formation in the area upslope from potential repository surface facilities. In addition, the physical characteristics of possible lava flows appear to exceed the design basis characteristics for other hazardous events (e.g., CRWMS M&O, 1999b). The current information has not been presented in a way to clearly support screening of lava flow hazards based on either low probability or negligible consequence. Thus, DOE should present a specific, transparent technical basis to justify exclusion of this event from the Category 2 event sequences list, or include this potential hazard in preclosure safety analyses.

4.1.3.2.4.4 Technical Basis for Inclusion or Exclusion of Specific Hazards and Initiating Events

DOE eliminated the potentially adverse effects of volcanic eruptions characteristic of the Yucca Mountain region from the list of Category 2 event sequences without adequate justification. DOE has not provided a traceable methodology to support its conclusion that the probability of a new volcano forming at the potential repository site (including surface and subsurface facilities) is less than 1×10^{-6} per year. DOE should provide a quantitative analysis for staff review, which includes the technical basis used to select probability models, definition of model parameters relevant to the geographic area of concern, and delineation of surface and subsurface facilities relevant to safety. This analysis also should address concerns with the DOE postclosure igneous event probability estimate discussed in Section 5.1.2.2 of this report.

A future volcano that formed some distance away from the potential repository facilities also could create hazardous conditions from lava-flow or tephra-fall deposits. DOE eliminated the potential hazards from tephra-fall deposits by concluding the thickness of these potential deposits would not exceed minimum design load requirements specified by the Uniform Building Code (International Conference of Building Officials, 1997). DOE has not provided sufficient information to support this conclusion and eliminate these hazards. Available information indicates a new volcano forming within approximately 10 km [6.3 mi] of the potential repository site could form a tephra deposit that, in many instances, might exceed the minimum design load by up to factors of 35. DOE also assumed that the effects of volcanic tephra on high-efficiency particulate air filters and heating, ventilation, and air conditioning systems are bounded by sandstorms (CRWMS M&O, 1999b). However, tephra-fall deposits contain a greater range of particle sizes than wind-blown sands and, thus, may have different effects on high-efficiency particulate air filters and heating, ventilation, and air conditioning systems. Independent probability calculations show that, for many different models and parameter ranges, the likelihood of a new volcano forming within approximately 10 km [6.3 mi] of the potential repository site exceeds 1×10^{-6} per year. The cumulative effect of these concerns is that DOE should provide additional rationale for excluding tephra-fall hazards from the Category 2 event sequences list. DOE also should provide a technical basis to exclude hazards from lava flows in the potential repository surface operations area from the Category 2 event sequence list. These issues were outside the scope of the first technical exchange and management meeting for preclosure safety (Reamer, 2001).

4.1.3.2.5 Operational Hazards

4.1.3.2.5.1 Technical Basis and Assumptions for Methods Used for Identification of Hazards and Initiating Events

The DOE operational hazard analysis methodology is documented in CRWMS M&O (1999a) and Bechtel SAIC Company, LLC (2002a). DOE used a combination of three hazard evaluation techniques, namely, Energy Analysis, Energy Trace and Barrier Analysis, and Energy Trace Checklist (System Safety Society, 1997), to develop the generic checklist of hazards applicable to preclosure operations. The operational hazards have been categorized as (i) Collision/ Crushing, (ii) Chemical/Contamination/Flooding, (iii) Explosion/Implosion, (iv) Fire, (v) Radiation/Magnetic/Electrical/Fissile Materials, and (vi) Thermal. DOE divided the surface and subsurface facilities into several functional areas and applied the checklist to identify hazards in each functional areas.

The main objective of hazard analysis is to identify initiating events that potentially may result in radioactive consequences to public and workers. Although the DOE methodology to identify hazards and potential initiating events is based on standard hazard analysis techniques, preliminary review of Bechtel SAIC Company, LLC (2002a) shows a potential weakness. The methodology does not provide a clear process to identify initiating events from the identified hazards. For example, probabilistic risk assessment studies have shown that human errors can be important contributors to the risk associated with nuclear facility operations (Swain and Guttman, 1983). Human error also is expected to contribute significantly to risk in potential repository operations (Eisenberg, 2001a). 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 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 (DOE, 2001c). Software reliability may be a significant factor in the safe operation of the potential Yucca Mountain repository (Eisenberg, 2001b). DOE should identify hazards and initiating events associated with reliability of hardware and software including reliability of remote systems used in the operations in preclosure safety analysis.

A hazard by itself would not result in event sequences and radiological consequences unless it was initiated by an event. Each hazard could be initiated by one or more events. Initiating events depend on facility operations and procedures, and functions of structures, systems, and components. DOE analysis of event sequences from initiating events in preliminary preclosure safety analysis (DOE, 2001a) did not clearly indicate how initiating events were identified from the hazard analysis. Additionally, the DOE preclosure safety analysis methodology (Bechtel SAIC Company, LLC, 2002a) also does not indicate how the initiating events are identified from the hazard analysis.

Natural and human-induced hazards may become potential initiators during facility operations, resulting in radiological consequence to the public and workers. DOE stated it plans to design the facility to withstand the natural and human-induced hazards; therefore it eliminated the effects of those hazards on facility operations from further consideration in the preclosure safety analysis (CRWMS M&O, 1999b). DOE should identify the design bases for the structures, systems, or components relied on to withstand any natural and human-induced hazards.

The NRC staff has reviewed (Lesher, et al., 2003) the DOE evaluation of the hazard potential of an underground transporter under runaway condition (CRWMS M&O, 2000c) in the North Ramp. Waste packages would be transported from the surface through the North Ramp to the emplacement drifts. The North Ramp is more than 2 km [1.24 m] long at a slope greater than −2 percent (CRWMS M&O, 2000c). The long downward slope of the North Ramp is the primary concern for loss of braking and subsequent runaway of the transporter, leading to a derailment. CRWMS M&O (2000c) presented the results of a preliminary evaluation of the potential hazards of a tip-over and derailment of a transporter train while descending the North Ramp. CRWMS M&O (2000c) analyzed different scenarios in which transporter speed for runaway conditions initiated at different locations along the ramp was evaluated at the curve and before the transporter reaches the access main tunnel. This analysis confirmed that the maximum velocity of the transporter in the event of a runaway would exceed the speed at which the transporter would tip over and, therefore, runaway of the transporter is a potential hazard. This analysis also confirmed that the runaway transporter would derail only under worn track conditions. In addition, the staff review of the methodology used by CRWMS M&O (2000c) for calculating the transporter tipover speed indicates that the methodology is consistent with accepted industry and engineering practices.

4.1.3.2.5.2 Use of Relevant Data for Identification of Site-Specific Hazards and Initiating Events

Identification of operational hazards and initiating events should encompass all relevant aspects of radiological systems and modes of operation in the geologic repository operations area. Appropriate information on structures, systems, components, and operational process activities described in Section 4.1.2 of this report should be used to evaluate hazards in the facility. The DOE facility design is being modified from that presented in the site suitability report (DOE, 2001a) with significant changes in layout, design, and functionality (McDaniel, 2004; Board, 2004). Information about the DOE facility and preclosure safety analysis for the revised facility design is not available to the staff for review.

4.1.3.2.5.3 Determination of Frequency or Probability of Occurrence of Hazards and Initiating Events

The DOE demonstration of compliance with performance objectives for Categories 1 and 2 events sequences would depend on identification of relevant initiating events and estimation of their frequencies. In addition, DOE would need to take into account uncertainties in its approach to evaluate probabilities or frequencies for identification of initiating events and analysis of event sequences. The following discussion is based on review of limited hazards and initiating events presented by DOE.

For hazards initiated as crane system failure, DOE estimated crane drop frequency for heavy lifts, such as shipping casks, disposal containers, and canisters, using actuarial data on crane operations available from Newport News Shipbuilding Facility (CRWMS M&O, 1998, Attachment X). The bridge crane failure rate of 1.4×10^{-5} drops per lift is based on the total number of dropped loads and total number of lifts of nonmagnetic cranes during 1996 and 1997. The estimated drop rates for normal operation drop events and two-block drop events are based on a relatively short period of 2 years and should be justified. In addition, data from a 2-year period do not reflect the commonly observed initial high failure rates of mechanical and electrical components (NRC, 1994) immediately after a crane is commissioned. Further, the type and complexity of operations at the shipbuilding facility are likely to be substantially different from the cask and waste package lifting operations at the potential repository, which would be performed remotely in a hot cell environment. In addition, the DOE evaluation of the initiating event frequency for the assembly drop in the assembly transfer area is based on the drop rate experience in fuel handling operations at commercial nuclear reactor facilities (CRWMS

M&O, 2000b). DOE estimated the failure rate of the assembly transfer machine as 1.8×10^{-5} based on identified assembly drop events and analysis of handling operations data obtained from 110 nuclear power plants between 1970 to 1991 (CRWMS M&O, 1997b). DOE should reassess failure rates of assembly transfer machines using updated information from nuclear crane operating experience beyond 1991.

The preliminary evaluation of the runaway probability of underground transporter at North Ramp as given in CRWMS M&O (2000c) was reviewed by staff (Lesher, et al., 2003). The consequence of a runaway transporter train or an uncontrolled descent along the North Ramp would be derailment or partial tipover (with wall impact). Both accident scenarios potentially can damage the waste package. CRWMS M&O (2000c) used the fault tree analysis technique to determine the probability of a transporter runaway. CRWMS M&O (2000c) revised previous fault tree studies (CRWMS M&O, 1997c,d) to determine transporter runaway probability after incorporating several safety features. One of the goals of the analysis presented in CRWMS M&O (2000c) is to assess safety features to reduce the annual frequency of occurrence to less that 10^{-6} events per year (i.e., Category 2 frequency limit assuming 100 years as the preclosure period) so that a runaway event can be eliminated from further consideration.

CRWMS M&O (1997d) used actuarial data from accidents involving commercial railway and mine locomotives to estimate a median transporter runaway probability of 6.04×10^{-4} events per year. CRWMS M&O (1997d) modeled Runaway Occurs on North Ramp as the top event in fault tree analysis deriving from probability of occurrence of Runaway Initiated from failure of components and systems and Failure to Apply Brakes After Runaway Initiation caused by a human failure. A failure probability of 5.88×10^{-4}, which is greater than the 10^{-6} per year Category 2 frequency limit, was derived from the fault tree analysis in CRWMS M&O (2000a).

CRWMS M&O (2000c) investigated safety features that could reduce the likelihood of the operator error, which is the dominant contributor to the runaway probability. CRWMS M&O (2000c) provided fault tree analyses to demonstrate the extent to which the transporter runaway probability could be reduced by adding supplemental design and safety features to the basic transporter design. The safety features analyzed include an electronic interlock to ensure dynamic brakes are engaged before the operator can start the train down the North Ramp, an alarm to alert the operator when the train speed exceeds the normal range during descent, a control system to automatically actuate the service brakes during normal descent (speed controls) with human operators providing backup actuation. Additional safety features consisting of a device to actuate the emergency brakes automatically during excessive speed and a redundant and diverse brake system were analyzed to study the effect on the runaway probability. CRWMS M&O (2000c) studied the effects of incorporating these safety features individually and in combinations in the fault tree analysis and determined that combining safety features was most effective in reducing the probability of runaway events. For instance, combining a speed alarm and automatic emergency brake that addresses the limitations in human response reduces the runaway event probability to 3.69×10^{-9}. Staff review of the DOE analysis indicates that it generally contains information sufficient for use in developing a potential license application. However, staff is concerned about zero probability assigned to a communication link failure event assuming that a communication link would remain operable throughout descent in Failure to Apply Brakes After Runaway Initiation fault tree. There is a possibility, however, that safety-critical information may be transmitted through the link, either to the operator or directly to one of the safety systems during descent through the North Ramp. Therefore, the possibility for communications failure during descent should be considered. In addition, DOE should investigate whether failure of the speed controller would initiate a runaway event. The analysis presented in CRWMS M&O (2000c, 1997c,d) considered only point estimates for data used in the analysis. Future analysis should consider uncertainties in input data to assess sensitivity of the top event to these uncertainties.

CRWMS M&O (2000d, 1997b) analyzed operational hazards and initiating events involving waste packages during surface handling, and transportation and emplacement in subsurface facilities. For many of these identified events, DOE did not evaluate the initiating event frequency but instead assumed the annual frequency for these hazards to be greater than 10^{-6}. For example, a waste package tipping over and slapping down on a flat surface, waste package falling on sharp objects, the emplacement gantry dropping waste packages, transporter door closing on waste packages were assumed to have an annual frequency larger than 10^{-6}. DOE eliminated these hazards from further analysis on the basis that radiological release would be prevented by the design of waste packages. DOE should provide appropriate design bases and criteria for the waste packages to show that these hazards are mitigated.

4.1.3.2.5.4 Technical Basis for Inclusion or Exclusion of Specific Hazards and Initiating Events

Based on the site suitability design (DOE, 2001b), DOE presented a list of hazards in DOE (2001b) and CRWMS M&O (1999a). The event sequence analysis presented in preliminary preclosure safety analysis (DOE, 2001b) considered a limited number of initiating events that have potential to cause radiological consequences. CRWMS M&O (1999a) also presented a list of events that are excluded from further analysis because the designs of structures, systems and components are credited to prevent radiological release. In addition, CRWMS M&O (1999a) presented a list of events excluded from further consideration because frequency of occurrence is below the regulatory limits. As stated before, DOE facility design is being significantly modified (McDaniel, 2004; Board, 2004) from that presented in the site suitability report (DOE, 2001a).

4.1.3.2.5.5 List of Hazards and Initiating Events To Be Considered in the Preclosure Safety Analysis

The staff is reviewing the DOE list of hazards and initiating events for appropriateness and completeness. Based on the review to date, the staff has questions on the aircraft hazard assessment by DOE. In a separate technical exchange (Schlueter, 2003c), staff has discussed the concerns with DOE. Preclosure Agreement PRE.03.02 related to tornado missile has been closed based on information provided by DOE. Staff has no further questions at this time.

No information has been provided by DOE on how it will ensure that concurrent construction activities do not compromise public and worker safety during preclosure operations. For example, Dry Transfer Facility 1 will be constructed first and, once it is in operation, Dry Transfer Facility 2 will be constructed. Similarly, new emplacement drifts will be constructed simultaneously with the emplacement operations in already constructed drifts. DOE has not addressed potential hazards resulting from simultaneous construction and operation of both surface and subsurface facilities. DOE should analyze these potential hazards for inclusion or exclusion from the hazard list.

4.1.3.3 Summary and Status

DOE has not provided sufficient information for the staff to completely assess the identification of hazards and initiating events during the preclosure period. DOE should provide further information at the time of the potential license application.

Table 4.1.3-4 provides the status of the preclosure identification of hazards and initiating events.

Table 4.1.3-4. Summary of Resolution Status Hazard and Initiating Events Identification Preclosure Topic

Preclosure Item	Status	Related Agreement*
Hazards and Initiating Events Consideration	Open Closed	PRE.03.01 PRE.03.02
Site Data	Open Closed	PRE.03.01 PRE.03.02
Estimation of Frequency	Open Closed	PRE.03.01 PRE.03.02
Exclusion or Inclusion of Hazards and Initiating Events	Open Closed	PRE.03.01 PRE.03.02
List of Hazards and Initiating Events	Staff review incomplete	None at this time

*The first Technical Exchange and Management Meeting for Preclosure Safety focused only on Aircraft Crash and Tornado Missiles Hazards [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. ML021340719. Washington, DC: NRC. 2001. <www.nrc.gov/ waste/hlw-disposal/public-involvement/mtg-archive.html#KTI>]

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4.1.4 Identification of Event Sequences

4.1.4.1 Areas of Review

This section provides review of the identification of event sequences during the preclosure period. The applicable requirements are

- 10 CFR 63.21(c)(5) requires a preclosure safety analysis of the geologic repository operations area, for the period before permanent closure, to ensure compliance with 10 CFR 63.111(a), as required by 10 CFR 63.111(c).

- 10 CFR 63.112(b) requires the preclosure safety analysis must include an identification and systematic analysis of naturally occurring and human-induced hazards at the geologic repository operations area, including a comprehensive identification of potential event sequences.

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 for normal operations, Category 1 event sequences, and Category 2 event sequences meet the preclosure performance objectives stated in 10 CFR 63.111.

Event sequence analyses are based on development of 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 or inactions. The scenarios are analyzed using event trees that results in a series of event sequences. DOE should ensure in its preclosure safety analysis that all possible event scenarios are considered and all event tree analysis account for uncertainty and variability in the estimated frequency and probability data. The probability of events that appear in event tree are quantified using actuarial data, fault trees, Bayesian analyses, expert judgement or other methods of estimation.

4.1.4.2 Staff Review of Available Information

4.1.4.2.1 Technical Basis and Assumptions for Methods Used for Identification of Event Sequences

DOE provided the preclosure safety analysis guide (Bechtel SAIC Company LLC, 2002) in response to Preclosure Agreement PRE.6.02. Bechtel SAIC Company LLC (2002) describes the overall DOE approach to conduct a preclosure safety analysis; identify structures, systems, and components important to safety; and develop design bases. The guide presents the DOE preclosure safety strategy; overview of the preclosure safety analysis; external and internal hazard analysis methodologies; event sequence analysis, including human reliability, common-cause and dependent failures, and technical information related to failure rates of components; consequence analysis; uncertainty analysis; and categorization of structures, systems, and components important to safety. Based on a review of this guidance document, NRC considered Preclosure Agreement PRE.6.02 (Schlueter, 2002) completed.

Consistent with the guidance developed in Bechtel SAIC Company, LLC (2002), DOE performed event sequence analyses using the event tree technique starting in 1998 (CRWMS M&O, 1998). The event tree methodology is widely used in probabilistic risk analysis for nuclear power plants Hickman, et al.(1983). The success of the technique is based on three basic presumptions (Hickman, et al., 1983; System Safety Society, 1997): (i) all system events have been anticipated, (ii) all end states of these events have been explored, and (iii) the probabilities of all the events have been correctly assumed.

The DOE identification of operational event sequences which are based on design of the facilities for site recommendation (DOE, 2001a), is reported in CRWMS M&O (2000a). DOE event tree analysis includes an initiating event and subsequent events associated with the failure of structures, systems, or components intended as safety features for prevention or mitigation of a given event. In the DOE analysis, the failure probability of a component or system is based on either information available from industry and literature or fault tree analysis in which individual components were modeled to ascertain the failure probability of the system. Staff agree with the DOE overall methodology for identifying potential event sequences at the repository. However, staff identified the following concerns with the DOE implementation of the methodology:

- DOE presents event sequence analyses with only point estimates of probability of failure of different components (CRWMS M&O, 2000a). It is unclear whether the probability estimate DOE uses in its analyses represents mean, median, or some other point estimate. Frequency of component failure can be, however, highly uncertain. By ignoring the uncertainty and variability associated with each frequency or probability estimate, there is a distinct possibility of incorrectly categorizing an event sequence with associated consequences. DOE should assign distributions to component failures and propagate uncertainty to estimate event sequence frequency. NRC stated this position at the Technical Exchange on Preclosure Safety (Reamer, 2001a).

- The DOE approach to categorize event sequences in low-temperature facility design is inconsistent and unclear. DOE states that if the preclosure period is extended beyond 100 years for a low-temperature operating mode, the preclosure period could be divided into two phases (Bechtel SAIC Company LLC, 2002). The Phase 1 period would consider waste-handling operations at the surface facility and waste-emplacement operations in the subsurface facility. Phase 2 would be the period after the emplacement and before final closure of the repository. The guide does not elaborate on how the two-phase preclosure period will affect the probabilities of occurrence or the associated event sequence categorization in the preclosure safety analysis. 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 4.1.4.1.

4.1.4.2.2 Category 1 and 2 Event Sequences

Based on the preliminary design of the potential repository, DOE identified some event sequences reportrd in DOE (2001a) and associated reports (CRWMS M&O; 2000a, 1998, 1997a). As discussed in Section 4.1.2, the DOE facility design is being modified from the design presented in the site suitability report (DOE, 2001b), with significant changes in layout, design, and functionality (DOE 2004a,b). Information is not available about the DOE analysis of event sequences from surface and subsurface operations for the revised facility design.

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–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). Adequacy of DOE identification natural and human-induced hazards are discussed in Section 4.1.3. DOE did not develop scenarios from these initiating events because DOE proposed to design, construct, and operate the potential repository to withstand these events (DOE, 2001b). In the future, when DOE submits the design, staff will review and evaluate adequacy of the DOE design, construction, and operations to prevent or mitigate natural and human-induced initiating events.

DOE has developed lists of potential event sequences from the events generated only from the facility operations. These potential event sequences are classified into three groups: internal event sequences with potential release, internal event sequences with no release, and beyond design basis events (DOE, 2001a). Staff comments in this version of the Integrated Issue Resolution Status Report are limited to only the operational hazards for the design submitted for the site suitability evaluation (DOE, 2001b).

The event sequences resulting from the potential facility operations of a geologic repository operations area that could potentially release radioactive material are 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 is further reduced to nine (DOE, 2001a–c).

DOE identifies 35 operational event sequences not expected to result in radiological release (DOE, 2001a, Table 5-7). The event sequences in this group are determined to be credible (i.e., expected to occur during the geologic repository operations area operational period), however, DOE excludes these event sequences from the 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 mitigate a release should the event occur. Event sequences identified in this group are waste package drops during surface and subsurface operations (CRWMS M&O, 2000b, 1997b).

DOE also generates 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 the preclosure period and is based on specific facility design features, physical barriers, and administrative controls or a combination of these factors. DOE excludes 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 these event sequences may become credible if the prevention and mitigation features are altered because of changes in the facility design (DOE, 2001a).

Staff identified the following concerns with the DOE identification of event sequences:

- DOE has not demonstrated consistency and traceability in its preclosure safety analysis from the identification of hazards to development of event sequences. Potential initiating events are analyzed for the frequency of occurrences in several CRWMS M&O reports (2000a,b, 1999a, 1998, 1997a,b) and credible initiating events are used in the event scenario development and event tree analysis (CRWMS M&O, 2000a, 1998). DOE does not, however, show how the initiating event list is generated from hazard analysis (CRWMS M&O, 1999b). This information should be more transparent in the analyses.

- DOE identifies event sequences for the geologic repository operations area that are not expected to result in radiological release (DOE, 2001a). These event sequences, listed in Table 5-7, could be classified as Category 1 or Category 2. DOE, however, has not classified them as Category 1 or Category 2 and instead 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 are excluded from Category 1 or Category 2 event sequences and are not considered in the safety assessment. Structures, systems, and components credited to prevent radiological consequences from the set of event sequences in Table 5-7 are waste package, shipping cask, canisters, bridge crane and lifting fixtures, waste package lifting systems, and so on. DOE has not provided adequate technical justification that screening of event sequences on the basis of design only is consistent with the 10 CFR Part 63 requirements (Reamer, 2001b) NRC stated that DOE should take into account the staff views and comments on this issue as quoted here (Lee, 2001):

  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.]

DOE stated it would screen preclosure design basis events based on features that reduce either frequency or consequences consistent with the overall risk-informed, performance-based philosophy in 10 CFR Part 63 (Reamer, 2001a). DOE further stated 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. DOE described the methods for identifying sources of uncertainties and consideration of uncertainties in probabilities in event sequence modeling in Bechtel SAIC Company, LLC (2002).

4.1.4.3 Summary and Status

DOE has not provided sufficient information for the staff to completely assess the identification of event sequences during the preclosure period. The DOE facility design is being modified with significant changes in layout, design, and functionality. Information on the DOE analysis of event sequences from surface and subsurface operations for the revised facility design is not available. DOE should provide further information at the time of the license application for staff to evaluate this area.

Table 4.1.4-1 provides status of the preclosure identification of hazards and initiating events.

Table 4.1.4-1. Summary of Resolution Status of Identification of Event Sequences Preclosure Items

Preclosure Item	Status	Related Agreement
Justification for Methodology and Assumptions	Staff review in progress	None*
Identification of Category 1 and 2 Event Sequences	Staff review in progress	†

*Limited general concerns were discussed in the DOE and NRC Technical Exchange and Management Meeting on Preclosure Safety [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. ML021340719. Washington, DC: NRC. 2001. <www.nrc.gov/waste/hlw-disposal/public-involvement/mtg-archive.html#KTI>]. No agreements were reached.
†Not discussed at the DOE and NRC Technical Exchange on Preclosure Safety [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. ML021340719. Washington, DC: NRC. 2001. <www.nrc.gov/waste/hlw-disposal/public-involvement/mtg-archive.html#KTI>].

4.1.4.4 References

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4.1.5 Consequence Analyses

4.1.5.1 Areas of Review

This section provides the review of the preclosure consequence analyses. The applicable requirements are

- 10 CFR 63.21(c)(5) requires a preclosure safety analysis of the geologic repository operations area, for the period before permanent closure, to ensure compliance with 10 CFR 63.111(a), as required by 10 CFR 63.111(c).

- 10 CFR 63.111(c)(1) and (2) requires a preclosure safety analysis of the geologic repository operations area that meets the requirements of 10 CFR 63.112 must be performed. This analysis must demonstrate the requirements of 10 CFR 63.111(a) will be met and the design meets the requirements of 10 CFR 63.111(b).

- 10 CFR 63.111(a) requires protection against radiation exposures and releases of radioactive material. (1) The geologic repository operations area must meet the requirements of Part 20 of this chapter. (2) During normal operations, and for Category 1 event sequences, the annual TEDE {total effective dose equivalent} (hereafter referred to as "dose") to any real member of the public located beyond the boundary of the site may not exceed the preclosure standard specified at 10 CFR 63.204.

- 10 CFR 63.111(b) requires numerical guides for design objectives. (1) The geologic repository operations area must be designed so that, taking into consideration Category 1 event sequences and until permanent closure has been completed, the aggregate radiation exposures and the aggregate radiation levels in both restricted and unrestricted areas and the aggregate releases of radioactive materials to unrestricted areas will be maintained within the limits specified in paragraph (a) of this section. (2) The geologic repository operations area must be designed so that, taking into consideration any single Category 2 event sequence and until permanent closure has been completed, no individual located on or beyond any point on the boundary of the site will receive, as a result of the single Category 2 event sequence, the more limiting of a TEDE of 0.05 Sv [5 rem] or the sum of the deep dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 0.5 Sv [50 rem]. The lens dose equivalent may not exceed 0.15 Sv [15 rem], and the shallow dose equivalent to skin may not exceed 0.5 Sv [50 rem].

Consequence analyses assess the potential radiological doses to members of the public and onsite workers during the preclosure period from operations in the surface and subsurface facilities of the geologic repository operations area. In general, the preclosure safety analysis considers potential radiological consequences resulting from normal operations, Category 1 event sequences, and Category 2 event sequences. Consequences are not required to be analyzed for event sequences with probabilities of occurrence less than the minimum probability specified in 10 CFR 63.2 for Category 2 event sequences (see Section 4.1.4 for more details).

4.1.5.2 Staff Review of Available Information

The DOE general description of the preclosure consequence analyses, including the dose calculation methodology, and summary results are documented in DOE (2001a). CRWMS M&O (2000) provides detailed documentation of the preclosure dose calculation. Portions of additional available documentation were reviewed to the extent they contain data or analyses that support the preclosure consequence analyses. The review documented in the following sections is focused on if DOE has (i) an acceptable methodology and (ii) sufficient data to demonstrate compliance. This review does not include a determination of compliance with the preclosure performance objectives. This review is based on the publicly available information which lags behind the information on the latest design.

4.1.5.2.1 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

4.1.5.2.1.1 Assessment of the Consequence Analyses Conducted for Normal Operations and Category 1 Event Sequences

The publicly available consequence analyses presented by DOE consider doses to the public offsite, but not to onsite workers. 10 CFR 63.111(a)(1) requires repository operations to meet the requirements of 10 CFR Part 20. 10 CFR Part 20 stipulates the dose limits for workers (Subpart C) and for members of the public (Subpart D), including the as low as is reasonably achievable requirements of 10 CFR 20.1101. These requirements with respect to worker safety were discussed with DOE at a technical exchange on a preclosure safety analysis guide (Schlueter, 2002). NRC noted the general need for the Preclosure Safety Analysis Guide to better address and integrate consequence analyses for onsite workers. DOE agreed future updates to the Preclosure Safety Analysis Guide would address the interfaces with the onsite worker dose analyses.

DOE asserted, because of the robust nature of the cladding of Naval spent nuclear fuel, credible impacts will not breach this cladding. The validity of this assumption has not yet been assessed. DOE conducted offsite consequence analyses for the release of activated corrosion products from Naval spent nuclear fuel (CRWMS M&O, 1999a). Without taking credit for high-efficiency particulate air filters in the ventilation system, DOE estimated offsite doses from the breach of a disposable canister containing Naval spent nuclear fuel to be below the regulatory limits in 10 CFR 63.111. Based on these offsite dose results (i.e., the consequences from a hypothetical canister breach did not exceed the limits), DOE stated Naval spent nuclear fuel canisters would not be certified to withstand all credible handling events. The onsite consequences to workers also should be determined from a breach of Naval spent nuclear fuel canisters. In addition, consequence analyses to members of the public offsite and to workers onsite should be presented for credible breaches of other canisters. This topic was discussed with DOE at a technical exchange on a preclosure safety analysis guide (Schlueter, 2002). DOE stated work on the spent nuclear fuel canisters was not complete. If the consequence analyses of a hypothetical canister breach result in doses that exceed the preclosure performance objectives, DOE stated the canister would be certified not to breach.

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 emergency planning after a Category 1 event sequence has not been addressed, there is a possibility 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 and members of the public are present within the 11-km [6.8-mi] withdrawal area boundary, Category 1 consequence analyses would be required to consider these individuals {i.e., dose calculations for members of the public within 11 km [6.8 mi]}. DOE should justify whether an emergency plan for members of the public is needed for 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 topic was discussed with DOE at a technical exchange on a preclosure safety analysis guide (Schlueter, 2002). DOE stated any members of the public located within the withdrawal area boundary would be evacuated to outside the site boundary. Staff has no additional comments at this time.

Bechtel SAIC Company, LLC (2002, Section 8.2.4) states DOE is not assessing the impacts from plutonium disposition wasteforms because the program is said to be on hold. It is not clear whether Section 8.2.4 is applied to vitrified plutonium, mixed-oxide fuel, or both. This topic was discussed with DOE at a technical exchange on a preclosure safety analysis guide (Schlueter, 2002). DOE clarified that mixed-oxide fuel would be included in the impact assessments. DOE would evaluate vitrified plutonium and any other types of high-level waste sent to the potential repository.

4.1.5.2.1.2 Assessment of Calculations of Consequence to Workers and Members of the Public from Normal Operations and Category 1 Event Sequences

In analyzing radiation doses from Category 1 event sequences, DOE proposed to use input parameters based on long-term average data, such as annual average atmospheric dispersion factors and average waste characteristics for the source term (Bechtel SAIC Company, LLC, 2002). 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 the 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 of an annual dose to any real member of the public from Category 1 event sequences from normal operations that must not be exceeded in any year, parameters based on appropriate short-term data may have to be used to enable a demonstration with reasonable assurance the parameters used in the calculations are appropriate for the scenario used. DOE may have to use short-term data for atmospheric dispersion and other parameters and provide a technical justification for the appropriateness of all data used for the dose calculations. This topic was discussed with DOE at a technical exchange on a preclosure safety analysis guide (Schlueter, 2002). DOE stated that acute 2-hour dispersion factors will be used to demonstrate that no single Category 1 event sequence will exceed the preclosure standard in 10 CFR 63.204. Annual average dispersion factors, however, would be used when evaluating the aggregate sum of Category 1 event sequences and normal operational events. DOE has indicated that it intends to model Category 1 events as chronic releases.

CRWMS M&O (2000, Attachment IV, Section 2.2) states the dose coefficients for external exposure are based on soil contaminated to a depth of 15 cm [5.9 in]. Using this contamination depth 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 will slowly migrate deeper into the soil with time. Attachment IV presents the dose calculation methodology for Category 1 event sequences, for which an exposure of 1 year is assumed. Studies of the depth distribution of radionuclides in soil for depositions less than 1 year shows 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. Selection of a normalized dose conversion (Sv yr^{-1} per Bq m^{-3}) based on a 15-cm [5.9-in] contaminated layer in U.S. Environmental Protection Agency (EPA) (1993) could be reasonable and considered to be conservative because a thicker contaminated layer added to the source term increases the normalized dose conversion factor. The uniform distribution assumption, however, would reduce the activity concentration (Bq m^{-3}) and result in lower estimates of the external dose. It is unclear if the expected activity of radionuclides deposited on the soil is distributed uniformly to a depth of 15 cm [5.9 in]. This topic was discussed with DOE at a technical exchange on a preclosure safety analysis guide (Schlueter, 2002). DOE stated that, for consequence calculations, radionuclides deposited on the soil are assumed to be distributed uniformly to a depth of 15 cm [5.9 in]. DOE should provide a technical basis to support its assumption.

NRC (2003, Section 2.1.1.5.1.2, Review Method 2, Page 2.1-30) includes guidance on calculations for onsite and offsite direct exposures during normal operations and Category 1 event sequences. For completeness, direct exposure calculations are required for external radiation sources, whether or not these are related to the release of radioactive material. DOE calculates direct exposure doses resulting from the release of radioactive material. DOE consequence analyses do not include direct radiation exposure dose estimates from radioactive material; this information on direct exposure should be addressed.

CRWMS M&O (2000, Page 11) describes the local deposition factor as the fraction of the airborne release fraction that is deposited locally within the Waste Handling Building. It appears 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, implying zero release from the Waste Handling Building. The local deposition factor is set at a value equal to 1 to maximize releases from the Waste Handling Building as part of Assumption 3.20, which was inconsistent with this definition. Furthermore, Eq. (11) of CRWMS M&O (2000) calculates the total release fraction to the environment and uses the local deposition factor directly to calculate the release fraction instead of using one minus the local deposition factor. Staff would prefer (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.

4.1.5.2.1.3 Assessment of the Methodology for Compliance with Regulatory Requirements

Although the DOE approach for demonstrating compliance applies a frequency weighting to the doses from Category 1 event sequences, the approach does not consider the potential for 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, credible combinations of multiple Category 1 event sequences occurring in a single year may have to be considered. Only those combinations expected to occur at least once before permanent closure (consistent with the Category 1 event sequence definition in 10 CFR 63.2) should be considered. DOE may have to present a table of the doses for each Category 1 event sequence and credible combination to ensure the limits specified in 10 CFR 63.111(a) are not exceeded. Staff communicated this comment to DOE at the Technical Exchange and Management Meeting for Preclosure Safety (Reamer, 2001), and DOE agreed to demonstrate the dose from any single Category 1 event sequence will not exceed the regulatory limit. Bechtel SAIC Company, LLC (2002, Section 8.3) states the total effective dose equivalent caused by combinations of Category 1 event sequences that can occur in a single year will be compared with the preclosure standard in 10 CFR 63.204 to show compliance with 10 CFR 63.111. DOE, however, has not presented its method for determining which combinations of Category 1 event sequences will be compared with the preclosure standard.

The DOE consequence analyses for workers from Category 1 event sequences are incomplete, based on available information. Occupational doses are calculated only for a noninvolved worker at an outside distance of 100 m [328 ft] (CRWMS M&O, 2000). Although DOE has only considered noninvolved workers at 100 m [328 ft], the floor plan (DOE, 2001b) clearly indicates worker activities inside the building, in the operating galleries by the side of the canister transfer, and in the assembly transfer areas. DOE (2001a, Section 5.3.6.2) asserts "... the potential radiological exposure during an accident for workers located less than 100 meters from a radiological release (e.g., inside the Waste Handling Building) is expected to be minimal." The radionuclide air concentrations and 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 different worker doses. Doses to involved workers inside the Waste Handling Building also should be addressed for Category 1 event sequences to reasonably assure the occupational limits of 10 CFR Part 20 are met. CRWMS M&O (2000) presents doses for a worker at 100 m [328 ft] from the routine releases (CRWMS M&O, 2000, Attachment V). DOE should also discuss how subsurface ventilation reduces the radionuclide concentrations of airborne activation products expected within the drifts, when assessing the performance requirements of 10 CFR Parts 20 and 63 for workers inside the emplacement drifts. These topics were discussed with DOE at a technical exchange on a preclosure safety analysis guide (Schlueter, 2002). DOE explained the guide does not address worker safety. NRC noted there is a general need for the guide to better address and integrate analyses for onsite workers. DOE agreed future updates to the guide would address the interfaces with the onsite worker dose analyses.

DOE (2001a, Section 5.3.5.3) states staff located on the Nevada Test Site and the 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 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 the 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 repository-related 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 should be complied with. Consequently, staff located on the Nevada Test Site and the 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. This topic was discussed with DOE at a technical exchange on a preclosure safety analysis guide (Schlueter, 2002). Based on information available at that time, DOE stated the staff located on the Nevada Test Site and the Nellis Air Force Range would be monitored and trained in accordance with an established radiation protection program.

Bechtel SAIC Company, LLC (2002, Page 8-11) states probabilistic uncertainty analyses will be included as part of the consequence analyses for Category 2 event sequences but not for Category 1 event sequences. This topic was discussed with DOE at a technical exchange on a preclosure safety analysis guide (Schlueter, 2002). DOE pointed out probabilistic uncertainty analyses techniques were not necessary for Category 1 event sequences because of the proposed three-fold approach for demonstrating compliance: (i) aggregating frequency-weighted doses from Category 1 event sequences with the annual dose from normal operations will not exceed the preclosure performance objectives, (ii) no single Category 1 event sequence will exceed the preclosure performance objectives, and (iii) no credible Category 1 combination of event sequences will exceed the preclosure performance objectives. Staff has no comments at this time.

Bechtel SAIC Company, LLC (2002, Page 8-9) stated the doses received via the direct exposure pathway from a passing airborne radioactive material would be compared with the dose constraint of 0.02 mSv/hr [2 mrem/hr] from external sources. Direct radiation from contaminated ground surfaces and the surface facilities represent additional sources of external exposure pathways that should also be accounted for in the direct exposure pathway assessment. This topic was discussed with DOE at a technical exchange on a preclosure safety analysis guide (Schlueter, 2002). DOE agreed and stated that future updates to the documentation would include additional discussion on the pathways for external sources.

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

4.1.5.2.2.1 Assessment of the Consequence Analyses Conducted for Category 2 Event Sequences

The staff evaluation of the identification of the Category 2 event sequences is contained in Sections 4.1.3 and 4.1.4 of this report. Consequence analyses would be required for additional Category 2 event sequences identified in those sections. Based on the available documentation, staff has not identified any other information needs regarding data or methodology.

4.1.5.2.2.2 Assumptions of Calculations of Consequences to Members of the Public from Category 2 Event Sequences

Bechtel SAIC Company, LLC (2002, Page 8-17) states, "Potential doses from the ingestion pathway are not included in the comparison to the regulatory limits because during the preclosure operations period there would be interdiction programs in place (to be established in a DOE emergency response program) to prevent the ingestion of contaminated food and water in the event of a Category 2 event sequence." NRC noted that DOE should demonstrate the facility design is in compliance with the performance objectives for all pathways and not automatically eliminate the ingestion pathway because of interdictions. DOE should calculate ingestion doses for an assumed exposure time or provide additional bases to justify exclusion of the ingestion pathway, including assumptions, from the consequence analyses for Category 2 event sequences. This topic was discussed with DOE at a technical exchange on a preclosure safety analysis guide (Schlueter, 2002). DOE stated it will consider the NRC position on ingestion pathways for the Category 2 event sequence consequence analyses.

In DOE (2001a), by assuming a 2-hour occupancy time, credit was taken for removing offsite members of the public after a Category 2 event sequence, yet an emergency plan was not described. DOE should justify how its emergency plan and assumed exposure time is appropriate for Category 2 consequence analysis.

Failed fuel (e.g., with cladding damage, debris, or pieces of fuel present) is to be placed in single-element disposable canisters. The consequences from failed fuel is 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, 1999b), which may have different particulate release fractions and source terms. The potentially different particulate release fractions from failed fuel should be addressed to support the argument that failed fuel was bounded by commercial spent nuclear fuel. For significant releases of radioactive material, credit can be taken for the mitigation of doses from pathways associated with radionuclides deposited on the ground and potential long-term exposure (NRC, 2003, Section 2.1.5.2.3, Review Method 2, page 2.1-38). This topic was discussed with DOE at a technical exchange on a preclosure safety analysis guide (Schlueter, 2002). DOE stated an update to the calculation is in progress which will address the release fractions for failed commercial spent nuclear fuel. In this update, DOE should evaluate the release fractions for wasteforms other than commercial spent nuclear fuel and include the potential differences in release fractions. In addition, DOE could provide a comparison of these alternative source terms with those used for commercial spent nuclear fuel.

Bechtel SAIC Company, LLC (2002, Section 8.2.1) and CRWMS M&O (2000, Section 5.2.7) estimate the bounding Co-60 crud source term based on commercial spent nuclear fuel with a burnup of 33 GWd/MTU [3.0×10^{12} Btu/ton uranium] and an enrichment of 3.2 percent. The maximum pressurized water reactor and boiling water reactor fuel characteristics are estimated to be 75-GWd/MTU [6.8×10^{12}-Btu/ton uranium] burnup, 5-percent enrichment, and 5-year decay time (Bechtel SAIC Company, LLC, 2002, Section 8.2.1). For pressurized water reactor and boiling water reactor spent nuclear fuels, the technical basis for the Co-60 crud activities per fuel assembly surface area should consider the range of potential fuel characteristics.

DOE intends to model the consequence analyses for Category 2 event sequences as acute releases (Bechtel SAIC Company, LLC, 2002, Section 8.4.3). Although the Preclosure Safety

Analysis Guide states the 50th percentile (median) acute dispersion factors are used for calculating mean doses, DOE clarified that mean acute dispersion factors were intended, and DOE will correct this inconsistency in a revision to the guide.

4.1.5.2.2.3 Assessment of the Methodology for Compliance with Regulatory Requirements

Bechtel SAIC Company, LLC (2002, Page 8-2) states doses to the skin and extremities are approximated using only the air submersion pathway. The report does not address direct exposure from contaminated ground surface. This topic was discussed with DOE at a technical exchange on a preclosure safety analysis guide (Schlueter, 2002). DOE stated that skin dose calculation would include the air submersion pathway and the contaminated ground surface, and this information will be clarified in a revision to the guide.

4.1.5.3 Summary and Status

4.1.5.3.1 Normal Operations and Category 1 Event Sequences

At the first Technical Exchange and Management Meeting for Preclosure Safety (Reamer, 2001), 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. No specific agreements were reached on the consequence analyses.

At a DOE technical exchange on a preclosure safety analysis guide (Schlueter, 2002), DOE and NRC engaged in more detailed discussions of the DOE consequence analyses approach that included both general and specific comments. The NRC comments and DOE responses are discussed in Section 4.1.5.2, Staff Review of Available Information. No specific agreements on the consequence analyses were reached at that meeting. Table 4.1.5-1 provides status of the preclosure consequence analyses for normal operations and Category 1 event sequences.

The preceding review indicates additional information may be needed from DOE for the preclosure consequence analyses for normal operations and Category 1 event sequences. Through the prelicensing issue resolution process, DOE provided responses to the NRC comments that may be sufficient for use in developing a license application, or involve the DOE commitment on future documentation. In its future documentation, DOE agreed to (i) address worker safety, (ii) provide consequence analyses for waste package breaches, (iii) demonstrate the doses for each Category 1 event sequence and credible combinations of Category 1 event

Table 4.1.5-1. Summary of Resolution Status of Consequence Analyses for Normal Operations and Category 1 Event Sequences Preclosure Topic

Preclosure Item	Status	Related Agreement
Assessment of the Consequence Analyses	Staff review in progress	None reached
Assessment of Calculations of Consequences to Workers and Members of the Public	Staff review in progress	None reached
Assessment of the Methodology for Compliance with Regulatory Requirements	Staff review in progress	None reached

sequences will not exceed the regulatory limits, and (iv) provide additional discussion on the pathways for external exposures.

Additional information regarding the consequences of Category 1 event sequences may be needed from DOE to support conclusions reached in the potential license application:

- A technical basis to support the assumption that any airborne radionuclides deposited on the soil are uniformly distributed to a depth of 15 cm [5.9 in]

- Direct exposure dose calculations from external sources not related to the release of radioactive material

- The method for determining which combinations of Category 1 event sequences will be compared with the preclosure standard or regulatory limits

4.1.5.3.2 Category 2 Event Sequences

At the first Technical Exchange and Management Meeting for Preclosure Safety (Reamer 2001), 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. No specific agreements were reached on the consequence analyses.

At a DOE technical exchange on a preclosure safety analysis guide (Schlueter, 2002), DOE and NRC engaged in more detailed discussions of the DOE consequence analyses approach that included both general and specific comments. No specific agreements on the consequence analyses were reached at that meeting. Table 4.1.5-2 provides status of the preclosure consequence analyses for Category 2 event sequences.

The preceding review indicates additional information may be needed from DOE for the preclosure consequence analyses for normal operations and Category 2 event sequences. Through the preliminary issue resolution process, DOE provided responses to the NRC comments that may be sufficient for use in developing a license application or involve the DOE commitment on future documenation. In its future documentation, DOE agreed to (i) consider the inclusion of all pathways in the consequence analyses for Category 2 event sequences, (ii) perform updated calculations for the release fractions of failed commercial spent nuclear fuel, and (iii) revise documentation on the doses to the skin and extremities. Additional

Table 4.1.5-2. Summary of Resolution Status of Consequence Analyses for Category 2 Event Sequences Preclosure Topic

Preclosure Item	Status	Related Agreement
Assessment of Consequence Analyses	Staff review in progress	None reached
Assessment of Calculations of Consequences to Members of the Public	Staff review in progress	None reached
Assessment of the Methodology for Compliance with Regulatory Requirements	Staff review in progress	None reached

information regarding the consequences of Category 2 event sequences may be needed from DOE to support conclusions reached in the potential license application.

4.1.5.4 References

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

4.1.6.1 Areas of Review

This section provides review of the identification of structures, systems, and components important to safety, safety controls, and measures to ensure availability of the safety systems. The applicable requirements are

- 10 CFR 63.21(c)(5) requires a preclosure safety analysis of the geologic repository operations area, for the period before permanent closure, to ensure compliance with 10 CFR 63.111(a), as required by 10 CFR 63.111(c).

- 10 CFR 63.112(e) requires the preclosure safety analysis must include an analysis of the structures, systems, and components to identify those that are important to safety. This analysis identifies and describes controls that are relied on to limit or prevent potential event sequences or mitigate their consequences. This analysis also identifies measures taken to ensure the availability of safety systems.

- 10 CFR 63.142(c)(1) requires that DOE shall identify structures, systems, and components to be covered by the quality assurance program.

According to 10 CFR 63.2, structures, systems, and components important to safety are 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. In addition, structures, systems and components, must be identified as important to safety if they are required to comply with 10 CFR Part 20 dose limits for Category 1 event sequences. Structures, systems, and components required to maintain compliance with 10 CFR Part 20 and 10 CFR Part 63 limits during normal operations, but not during Category 1 event sequences, are not considered important to safety. For determining those structures, systems, and components important to safety, Table 4.1.6-1 presents the dose limits required by 10 CFR 63.111 for Categories 1 and 2 event sequences for both public and worker safety.

The DOE identification of structures, systems, and components important to safety is the end product of its preclosure safety analysis. To properly identify structures, systems, and components important to safety, DOE must adequately identify hazards and initiating events, identify event sequences, evaluate frequencies, and evaluate the consequences of the preclosure operations at the geologic repository operations area. Staff will verify that analysis and identification of structures, systems, and components for the geologic repository operations area use results of the preclosure safety analysis and are consistent with the definitions specified in 10 CFR 63.2. Staff will review design bases and criteria, design methodology and analysis, and design of structures, systems, and components, using review methods in Section 2.1.1.7 (NRC, 2003), with emphasis and focus on those important to safety.

Table 4.1.6-1. Dose Limits Used for Determining Those Structures, Systems, and Components Important to Safety			
Event Sequence Frequency	**Applicability**	**Dose Limits**	**Regulations**
Category 1	Member of the public	0.15 mSv/yr [15 mrem/yr] total effective dose equivalent to any real member of the public located beyond the site boundary	10 CFR 63.111(b)(1) 10 CFR 63.111(a)(2) 10 CFR 63.204
		1.0 mSv/yr [100 mrem/yr] total effective dose equivalent	10 CFR 63.111(b)(1) 10 CFR 63.111(a)(1) 10 CFR 20.1301(a)(1)
		0.02 mSv/hr [2 mrem/hr] and 0.5 mSv [0.05 rem] in a year effective dose equivalent in any unrestricted area from external source	10 CFR 63.111(b)(1) 10 CFR 63.111(a)(1) 10 CFR 20.1301(a)(2)
	Radiation worker receiving occupational dose as defined in 10 CFR 20.1003	0.05 Sv/yr [5 rem/yr] total effective dose equivalent	10 CFR 63.111(b)(1) 10 CFR 63.111(a)(1) 10 CFR 20.1201(a)(1)
		0.5 Sv/yr [50 rem/yr] individual organ dose equivalent to any organ or tissue (other than the lens of the eye)	10 CFR 63.111(b)(1) 10 CFR 63.111(a)(1) 10 CFR 20.1201(a)(1)
		0.15 Sv/yr [15 rem/yr] dose equivalent to the lens of the eye	10 CFR 63.111(b)(1) 10 CFR 63.111(a)(1) 10 CFR 20.1201(a)(2)
		0.5 Sv/yr [50 rem/yr] shallow dose equivalent to the skin or any extremity	10 CFR 63.111(b)(1) 10 CFR 63.111(a)(1) 10 CFR 20.1201(a)(2)
Category 2	Member of the public located on or beyond site boundary	0.05 Sv [5 rem] total effective dose equivalent per event	10 CFR 63.111(b)(2)
		0.5 Sv [50 rem] organ dose equivalent to any individual organ or tissue (other than the lens of the eye) per event	
		0.15 Sv [15 rem] dose equivalent per event to the lens of the eye	
		0.5 Sv [50 rem] shallow dose equivalent per event to the skin	

4.1.6.2 Staff Review of Available Information

4.1.6.2.1 List of Structures, Systems, and Components Important to Safety, Technical Bases for Identification of Structures, Systems, and Components and Safety Controls, and List and Analyses of Measures to Ensure Availability and Reliability of Safety Systems

DOE provided procedure AP–2.22Q (DOE, 2002) in response to Preclosure Agreement PRE.6.01. DOE planned to use the procedure for controlling the Q-list, to reflect items important to safety and their quality level categorizations. The objective of the Preclosure Agreement PRE.6.01 was to ensure the DOE approach to the categorization process is based on an acceptable technical basis and is consistent with the regulatory requirements of 10 CFR Part 63. Staff reviewed the response and needed additional information to close the agreement (Schlueter, 2002a). At a DOE and NRC quality assurance meeting (Schlueter, 2003), DOE indicated the structures, systems, and components identified as important to safety would not be further categorized commensurate with its safety function and will not implement a graded quality assurance approach. On May 25, 2004, DOE provided a revised procedure AP–2.22Q (DOE, 2003), in response to the staff need for additional information on Preclosure Agreement PRE.6.01. Staff did not identify any concerns with this revision of AP–2.22Q.

DOE provided Bechtel SAIC Company, LLC (2002) in response to Preclosure Agreement PRE.6.02. This guide describes the overall DOE approach to conduct a preclosure safety analysis; identify structures, systems, and components important to safety; and develop design bases. The preclosure safety analysis methodologies presented in the guide are used in safety assessments for nuclear power and other industries and present an approach to address the requirements in 10 CFR Part 63. The guide is an internal DOE document intended to assist with preparing the preclosure safety analysis and to serve as a training tool for the DOE technical staff. Staff reviewed this guide and did not identify any significant concerns (Schlueter, 2002b).

DOE presented a preliminary list of structures, systems, and components determined to be important to safety (DOE, 2001a, 2000) for a facility design in a site suitability report (DOE, 2001b). This preliminary list is categorized according to relative importance to safety. DOE revised the list of structures, systems, and components determined to be important to safety (Bechtel SAIC Company, LLC, 2003); however, the supporting information for this list was unavailable at the writing of this report.

The preclosure safety analysis process, as shown in Figure 4.1.6-1, is described in DOE (2003). The block diagram in Figure 4.1.6-1 illustrates the process of implementing the preclosure safety analysis. Staff did not identify any concerns with the DOE schematic representation of the preclosure safety analysis methodology.

The preclosure safety analysis required by 10 CFR 63.112 is the basis for identifying the structures, systems, and components important to safety. As a part of preclosure safety analysis, each of the structures, systems, and components will be analyzed to identify individual structures, systems, and components important to safety (Cereghino, 2004). Thus, an iterative design-classification process will be used, as indicated in Bechtel SAIC Company, LLC (2002). Finally, this iterative preclosure safety analysis process will be completed by adding the

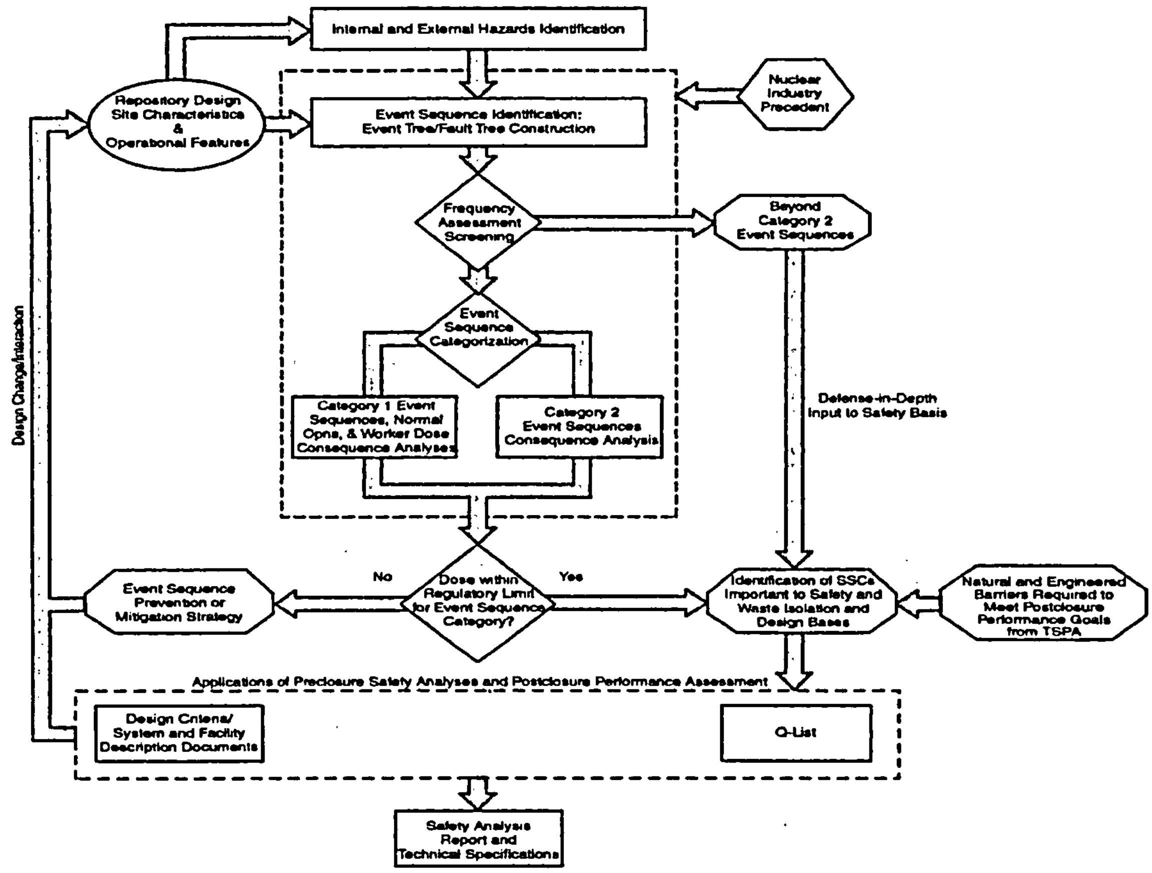

Figure 4.1.6-1. Overview of the DOE Preclosure Safety Analysis Process (DOE, 2003)

appropriate structures, systems, and components to the Q-list and the associated design criteria to the systems and facility description documents to support design.

Based on 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. An annual dose limit of 0.15 mSv [15 mrem] is specified in 10 CFR Part 63 for members of the public and the aggregated radiation exposure limits in the restricted and unrestricted areas must meet the dose limits in 10 CFR 63.111(a).

DOE proposes to classify individual structures, systems, and components for Category 1 event sequences based on the summation of three terms (Bechtel SAIC Company, LLC, 2002): (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 dose from a single Category 1 event sequence involving the failure of that particular structure, system, or component.

The current Q-List (Bechtel SAIC Company, LLC, 2003) includes structures and systems used in the geologic repository operations area. The systems are divided further to consider systems or operations. The current Q-List of structures, systems, and components important to safety is based on a design that is evolving. As discussed in Section 4.1.2, the DOE facility design is being modified from that presented in the site suitability report (DOE, 2001b) with significant changes in layout, design, and functionality (McDaniel, 2004; Board, 2004). Consequently, this Q-List may change as the design and the preclosure safety analysis mature. Lacking the final design interaction, no attempt has been made to review the adequacy and completeness of the Q-List until DOE submits the potential license application.

At the May 12, 2004, technical exchange meeting (Reamer, 2004), DOE described its process to identify structures, systems, and components important to safety and discussed the relationship of normal operations to the identification process. DOE stated a structure, system, or component is important to safety if a function is credited to prevent or mitigate Categories 1 and 2 event sequences. DOE noted an event sequence is a series of actions and occurrences within the natural and engineered components of a geological repository operation area that could potentially lead to radiological exposure to individuals. Structures, systems, and components required only for the facility to function within preclosure compliance requirements during normal operations are not considered important to safety. Not every element of a structure, system, or component is important to safety. Only those elements required to provide the credited function(s) are subjected to quality assurance requirements. DOE discussed the radiation dose limits that will be applicable for preclosure safety analyses and provided several examples for safety classification of structures, systems, and components (important and not important to safety) involved in the potential surface and subsurface operations. DOE defined normal operations as repository conditions where structures, systems, and components operate in the one designed configuration for handling and emplacing spent nuclear fuel or high-level waste without unplanned worker or public doses. The method used by DOE to identify structures, systems, and components important to safety appears to be sufficient for use in developing a potential license application.

4.1.6.2.2 Administrative and Procedural Safety Controls to Prevent Event Sequences or Mitigate Their Effects

To comply with 10 CFR Part 63, DOE is required to include in the list of structures, systems, and components important to safety any administrative or procedural safety controls needed to prevent event sequences or mitigate their effects. However, DOE does not include in its list of structures, systems, and components important to safety those administrative or procedural safety controls (DOE, 2001a). Further, management systems and procedures have not been provided to ensure administrative or procedural controls fulfill their intended purpose.

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

As stated in 10 CFR 63.142(c)(1), the quality assurance program must control activities affecting the quality of the identified structures, systems, and components to an extent consistent with importance to safety. Section 2.1.1.6 (NRC, 2003) provides review methods to evaluate any methodology of risk significance categorization of structures, systems, and components important to safety and to verify this methodology is consistent with applicable policy and guidance and is conducted using preclosure safety analysis. DOE and NRC discussed earlier quality level categorizations at two technical exchange meetings (Reamer, 2001; Schlueter, 2002c). DOE decided, however, not to implement graded quality assurance controls for structures, systems, and components important to safety (Schlueter, 2003). The DOE quality assurance program will be applied to those elements of important to safety structures, systems, and components required to provide the credited function(s) (Reamer, 2004).

4.1.6.3 Summary and Status

The information concerning the DOE methodology for identifying structures, systems, and components important to safety appears to be sufficient for use in developing a potential license application. However, the DOE facility design is being modified with significant changes in layout, design, and functionality, therefore, further information should be provided at the time of the potential license application for the staff to evaluate this area.

Table 4.1.6-2 provides the status of the preclosure identification of hazards and initiating events.

Table 4.1.6-2. Summary of Resolution Status of the Preclosure Topic: Identification of Structures, Systems, and Components Important to Safety, Safety Controls; and Measures to Ensure Availability of the Safety Systems

Preclosure Item	Status	Related Agreement
List of Structures, Systems, and Components Identified As Important to Safety*	Complete Complete	PRE.06.01 PRE.06.02
Administrative or Procedural Safety Controls	None	†
Risk Significance Categorization of Structures, Systems, and Components Important to Safety	Not applicable. DOE no longer proposes a graded quality assurance approach.	None

*Limited general concerns were discussed in the DOE and NRC Technical Exchange and Management Meeting on Preclosure Safety in Las Vegas, Nevada. [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. ML021340719. Washington, DC: NRC. 2001. <www.nrc.gov/waste/hlw-disposal/public-involvement/mtg-archive.html#KTI>], and Important to Safety Technical Exchange [Reamer C.W., "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Important to Safety Technical Exchange, (May 12, 2004)." Letter (June 18) to J.D. Ziegler, DOE. ML041700192. Washington, DC: NRC. 2004. <www.nrc.gov/reading-rm/adams.html>], in Las Vegas, Nevada. No agreements were reached. The DOE facility design is not finalized; therefore, the Q-List was not discussed.
†Not discussed at the DOE and NRC Technical Exchange on Preclosure Safety [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. ML021340719. Washington, DC: NRC. 2001. <www.nrc.gov/waste/hlw-disposal/public-involvement/mtg-archive.html#KTI>].

4.1.6.4 References

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

4.1.7.1 Areas of Review

This section provides review of the design, specifications, component assessment, and fabrication methods (as applicable) for the important to safety surface facilities, subsurface facilities, aging facilities, and the waste package and engineered barrier system. The applicable requirements are

- 10 CFR 63.21(c)(3) requires that the safety analyses report, filed with the license application, include a description and discussion of the design of the various components of the geologic repository operations area and the engineered barrier system including (i) dimensions, material properties, specifications, and analytical and design methods used, along with any applicable codes and standards, and (ii) the design criteria used and their relationships to the preclosure and postclosure performance objectives.

- 10 CFR 63.112(e) requires an analysis of the performance of the structures, systems, and components to identify those that are important to safety. This analysis identifies and describes the controls relied on to limit or prevent potential event sequences or mitigate their consequences. This analysis also identifies measures taken to ensure the availability of safety systems. The analysis must include, but not necessarily be limited to, consideration of

  (1) Means to limit concentration of radioactive material in air

  (2) Means to limit the time required to perform work in the vicinity of radioactive materials

  (3) Suitable shielding

  (4) Means to monitor and control the dispersal of radioactive contamination

  (5) Means to control access to high radiation areas or airborne radioactivity areas

  (6) Means to prevent and control criticality

  (7) Radiation alarm system to warn of significant increases of radiation levels, concentrations of radioactive material in the air, and increased radioactivity in effluents

  (8) Ability of structures, systems, and components to perform their intended safety functions, assuming the occurrence of event sequences

  (9) Explosion and fire detection systems and appropriate suppression systems

  (10) Means to control radioactive waste and radioactive effluents, and permit prompt termination of operations and evacuation of personnel during an emergency

(11) 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

(12) Means to provide redundant systems necessary to maintain, with adequate capacity, the ability of utility services important to safety

(13) Means to inspect, test, and maintain structures, systems, and components important to safety, as necessary, to ensure their continued functioning and readiness

- 10 CFR 63.112(f) requires that the preclosure safety analysis include a description and discussion of the design, both surface and subsurface, of the geologic repository area. This discussion should include the design bases and their relation to the design criteria, and the relationship between design criteria and the preclosure performance objectives specified in 10 CFR 63.111(a) and (b).

The level of detail for design information in the license application must provide sufficient information to support evaluations that demonstrate compliance with performance objectives for the repository system and demonstrate compliance with other NRC requirements.

4.1.7.2 Staff Review of Available Information

Staff reviewed the available information on the following surface facilities including aging facilities, subsurface facilities, and waste package and other engineered barriers. This review is to assess if, to date, DOE has provided information in the areas noted that will be sufficient to support a potential license application.

4.1.7.2.1 Surface Facilities

The surface facilities will be used to receive spent nuclear fuel and defense high-level waste shipments, provide capability to age waste as necessary, and prepare and package the wastes for underground emplacement (McDaniel, 2004; Board, 2004). In addition, the surface facilities also will house radiological protection, utilities, and ventilation for the underground facilities and provide other supporting functions. The surface facilities consist of four primary functional areas: (i) the waste receiving and inspection area, where incoming trucks and rail cars are inspected, received, and temporarily staged; (ii) the aging areas, where the received wastes are placed for cooling and radiological decay until ready for disposal; (iii) the surface portion of the waste handling operations area, which includes all buildings where radioactive material is handled for packaging; and (iv) the general support facilities, consisting of administrative buildings, security stations, and warehouses (McDaniel, 2004). Discussion about items (ii) and (iii) will be the focus of this section. The specific areas of review are

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

- Design Methodologies

- Geologic Repository Operations Area Design and Design Analyses

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

DOE provided NRC with limited information regarding the relationship between the design criteria and design bases and the regulatory requirements for the surface facilities. In addition, DOE provided limited information on the design and design analyses for these same facilities. DOE discussed the conceptual design and operation of the surface facilities during technical exchanges on February 3–4, 2004 (Schlueter, 2004) and May 12, 2004 (Reamer, 2004). During those technical exchanges, DOE provided overviews of the conceptual designs and operations and discussed its methodology for identifying structures, systems, and components that will be designated important to safety.

4.1.7.2.1.2 Design Methodologies

DOE is revising its seismic design methodology (DOE, 1997). The new design methodology will be risk-informed and consider the evolution of regulations related to seismic design for other nuclear facilities. An outline of the new seismic design methodology was submitted to NRC (DOE, 2004). In this document, DOE informed NRC it intends to revise and reissue Topical Report No. 2 (DOE, 1997) related to the preclosure seismic design methodology. This report is the second in a series of three topical reports originally planned by DOE and agreed to by NRC. The outline also indicates DOE no longer intends to issue the third seismic topical report, which was expected to include details of the implementation of the design methodology presented in the second topical report and a summary of seismic inputs used in the repository design and performance assessment. Instead, DOE will provide this information in Technical Basis Document No. 14, Low Probability Seismic Events.

DOE has indicated that the revised Topical Report No. 2 will provide the technical basis for its new seismic design approach. The DOE annotated outline for this report proposes the following:

- To use two design basis ground motion levels (1 and 2) as having mean annual exceedance probabilities of 1×10^{-3} and 5×10^{-4}. DOE states the design basis ground motion hazard levels adopted in the revised topical report are comparable to those given in the final rule at 10 CFR Part 72 for independent spent nuclear fuel storage installations and monitored retrievable storage facilities.

- To use preclosure safety analysis to identify structures, systems, and components important to safety and to associate the structures, systems, and components with design basis ground motion levels 1 or 2, based on the significance of the structures, systems, and components.

- To conduct two additional analyses of the structures, systems, and components to ensure adequate conservatism:

    — In the first additional analysis, "beyond design basis ground motions," the structures, systems, and components will be evaluated at larger ground motion levels (2,000 and 10,000-year return period ground motions for design basis ground motion levels 1 and 2 structures, systems, and components). The beyond design basis analyses will compare the resulting linear/nonlinear elastic seismic demands with high-confidence strength capacities. If seismic demands exceed the strength capacities, the structures, systems, and components will be redesigned.

— In the second additional analysis, DOE will conduct a high-confidence of low probability of failure analysis to ensure the structures, systems, and components have adequate seismic margins such that seismically initiated event sequences will meet the preclosure performance objectives.

A brief review of the annotated outline indicates the structures designed for certain design ground motions will be evaluated for beyond design basis earthquake ground motions. The information provided, however, is insufficient to assess feasibility of the methodology for designing and evaluating the structures, systems, and components of the potential geologic repository operations area at Yucca Mountain. The NRC staff will, therefore, look to the information that is available at the time of a potential license application.

4.1.7.2.1.3 Geologic Repository Operations Area Design and Design Analyses

4.1.7.2.1.3.1 Dry Transfer Facilities, Fuel Handling Facilities, and Canister Handling Facility

DOE has submitted limited design and design analysis information to NRC. The meeting minutes of the February and May 2004 technical exchanges (Schlueter, 2004; Reamer, 2004) contain conceptual drawings and work process diagrams of these facilities. This design information is not sufficient for staff to review the design and design analyses of these facilities.

4.1.7.2.1.3.2 Aging Facilities

DOE has submitted limited design and design analysis information to NRC. The meeting minutes of the February and May 2004 technical exchanges (Schlueter, 2004; Reamer, 2004) contain conceptual drawings of these facilities. This design information is not sufficient for staff to review the design and design analyses of these facilities.

4.1.7.2.2 Subsurface Facilities

Subsurface facilities consist of (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 (CRWMS M&O, 2000a). 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 potential repository and would connect the surface and subsurface facilities through the access mains (CRWMS M&O, 2000a). The meeting minutes of the February and May 2004 technical exchanges (Schlueter, 2004; Reamer, 2004) contain conceptual drawings of these facilities.

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 composed 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 are connected 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. Parts of the access mains also may be used for subsurface ventilation. The waste packages will be transported from the surface facility to the emplacement drift using a specially designed transporter. A self-propelled, remotely operated emplacement gantry will be used for emplacement operation.

To review the DOE design of the subsurface facilities, the repository design and operations are examined to identify the structures, systems, and components relied on to perform functions important to safety or waste isolation, or needed for normal operation. The staff review of DOE's design will consider the following repository operations: waste emplacement, ventilation for heat removal, monitoring and performance confirmation, potential retrieval, and closure operations such as drip shield emplacement. In addition, design conditions that bear on postclosure performance assessment, such as heat removal through ventilation or stability of the invert, are evaluated to determine the structures, systems, and components needed to support the conditions. The following subsurface facility structures, systems, and components are evaluated: emplacement drifts, turnout tunnels, ventilation shafts, access mains, ground-support system, invert structures, and the subsurface rail system.

The three specific areas of staff review are

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

- Design Methodologies

- Design and Design Analyses for Structures, Systems, and Components Equipment, and Safety Controls

4.1.7.2.2.1 Waste Transportation and Emplacement Equipment

DOE has submitted limited design and design analyses information to NRC regarding the waste transportation and emplacement equipment. The meeting minutes of the February and May 2004 technical exchanges (Schlueter, 2004; Reamer, 2004) contain conceptual drawings of this equipment. This design information is not sufficient for staff to review the design and design analyses of this equipment.

4.1.7.2.2.2 Access Ramps and Main, Emplacement Drifts, and Performance Confirmation Drifts

The design of the subsurface facilities 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, Thermal-Mechanical Effects on Underground Facility Design and Performance. In the subsequent sections, applicable portions of these subissues are considered but are not specifically identified.

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 access mains to the emplacement drifts (which are nearly perpendicular to the mains) consists of curved tunnels referred to as turnouts (CRWMS M&O, 2000a, Figure 1). The other openings of the underground facility may include ventilation shafts and other drifts within the emplacement block that may be used for purposes other than waste emplacement.

Harrington (2003) provides a description of the ground-support system for the walls and roof of the openings. For the emplacement drift ground support, DOE intends to use 3-m [9.8-ft]-long friction rock bolts spaced at 1.25 m [4.1 ft] with thin-wall {3-mm [0.12-in]-thick} Bernold-style perforated sheets. The bolts and sheets will be made of stainless steel. For the access and exhaust mains and the ramps, DOE intends to use fully grouted rock bolts and welded wire fabric, both of carbon steel. The turnouts and intersections between openings would be supported using fully grouted rock bolts, wire mesh, shotcrete, and, where necessary, lattice girders. Shafts would be supported using rock bolts and shotcrete or concrete.

The DOE information (Harrington, 2003) also indicates the emplacement drift invert would consist of a carbon-steel structure with crushed-tuff ballast. The carbon-steel structure would support the gantry rail system, waste packages, and drip shields during the preclosure period. The gantry rail system should be operational through permanent closure to support waste emplacement and the installation of drip shields. In addition, DOE points out the carbon-steel structure would not be needed thereafter because the crushed-tuff ballast will be designed to provide sufficient support to the waste packages and drip shields during the postclosure period.

DOE has not provided an analysis to demonstrate adequate performance of the current ground-support systems. For the ground-support system proposed by DOE to support its site recommendation (CRWMS M&O, 2000c), the analysis of emplacement-drift stability did not consider any degradation of the ground-support system during the preclosure period. DOE asserts in CRWMS M&O (2000d) that the carbon steel ground-support system (consisting of steel sets and occasional rock bolts) proposed as part of the site-recommendation design would not experience significant corrosion for 300 years. Therefore, DOE proposed as part of its site-recommendation analysis that the emplacement drift ground support would not need planned maintenance during a preclosure period of up to 175 years, and that planned maintenance would be needed only if the preclosure period were to be extended to 300 years (CRWMS M&O, 2000e). DOE asserted the ground-support system would not significantly corrode during the preclosure period because the corrosion rates would be negligible at the anticipated relative humidity in the range 1–40 percent, which is below the critical relative humidity for humid-air corrosion. The NRC staff requested information from DOE through DOE and NRC Agreement RDTME.3.01 to determine if there is an acceptable technical basis for excluding corrosion effects from consideration in the design of a maintenance-free ground-support system. Staff reviewed information provided by DOE to complete this agreement (Schlueter, 2003a) along with other DOE information regarding ground support design changes (Schlueter, 2003b) and design strategy (Bechtel SAIC Company, LLC, 2003a). The DOE ground-support design strategy consists of four steps: (i) develop an initial design using industry practice based on empirical relationships; (ii) evaluate the design through numerical modeling, considering an appropriate range of rock mass properties, loading combinations, environmental conditions, and repository operational requirements; (iii) estimate the corrosion potential and life expectancy of the ground support; and (iv) develop monitoring, inspection, and maintenance programs for the emplacement drifts as the design progresses from the conceptual to the detailed phases. This design strategy can be expected to result in

DOE providing sufficient information to permit an NRC assessment of the effectiveness of the ground-support system for the emplacement drifts.

During the preclosure period, forced ventilation will remove heat from the emplacement drifts. Ventilation also will remove water vapor and lower the relative humidity within the emplacement drifts. The external environmental conditions may alter the relative humidity within the emplacement drifts; however, heat generated by the waste should limit the effects of external conditions. Increases in the relative humidity above a threshold value may lead to the initiation of corrosion of the ground support materials. For most metals, including steel, the critical relative humidity for humid air corrosion is approximately 60 percent. Information provided by DOE shows the relative humidity inside the emplacement drifts should be maintained below the critical relative humidity for humid air corrosion, and the external environmental conditions should not significantly alter the relative humidity inside the emplacement drifts (Ziegler, 2003, 2002).

The rock bolts will be in complete or partial contact with the rock matrix. Thus, the water content and relative humidity of the rock mass are relevant to determining the environmental conditions surrounding the rock bolts. DOE infers the relative humidity of the air mass in the drifts should not be used to estimate the relative humidity adjacent to engineered materials in direct contact with the wallrock. Water potential measurements infer water contents in the rock matrix. There are no measurements of water potential, water content, or relative humidity in fractures at Yucca Mountain. DOE uses water potential data to infer a dryout thickness. Data from Bechtel SAIC Company, LLC (2001a) indicate the "driest" *in-situ* field testing of processes measured water potential in the wallrock is approximately −3 MPa [−30 bars]. This water potential can be shown to correspond to a relative humidity of approximately 98 percent in pore spaces using the standard Kelvin equation for porous media, which is the basis for psychrometers used at Yucca Mountain to measure water potential. Conceptually, the first few centimeters of the matrix probably have a fairly low (large negative value) water potential, such that the relative humidity in the pore space is significantly lower. Beyond the first few centimeters of depth into the wallrock, the pore space relative humidity is likely high. In the rock matrix near large aperture fractures, the water potential also is likely low. For most of the fractures, the water potential is probably slightly lower than the adjacent matrix, but not low enough to reduce the relative humidity significantly.

DOE did not evaluate the effects of mixed salts on the degradation of the drift support materials because credit for the performance of ground-support systems is limited to the preclosure period. DOE will provide an assessment of the effects of mixed salts on the deliquescence point will be provided in the responses to Agreements ENFE.2.13 and 2.15. The DOE analysis of ground support design and the strategy for monitoring, inspection, and maintenance of the ground support materials should consider the effects of any mixed salts that might form in the vicinity of the rock bolts during the preclosure period.

Based on the information provided by DOE, it is not clear ventilation would lead to a low relative humidity environment surrounding rock bolts or prevent the possibility that water may reside in the crevices between the rock bolts and the surrounding rock. DOE indicates, however, the potential effects of localized liquid phase water on the various ground support materials will need to be assessed. The environment in contact with the rock bolts should be included in the DOE strategy for monitoring, inspecting and maintaining the ground support materials.

It is expected DOE will consider these concerns in executing its ground support design strategy, which includes estimating the corrosion potential and life expectancy of the ground support and developing monitoring, inspecting, and maintaining programs for the emplacement drifts. The DOE information also should include the technical basis for the service life of the invert structural materials, to ensure the invert will be capable of supporting the waste packages and the emplacement gantry rail system during the preclosure period (Harrington, 2003).

DOE has not provided NRC with information regarding

- Load combinations for subsurface facility design
- Models and rock properties for subsurface facility design
- Subsurface ventilation system design
- Subsurface power and power distribution systems design
- Maintenance plan for subsurface facility
- Subsurface ground-support systems design

4.1.7.2.2.3 Ventilation System

DOE submitted a design and design analysis for the ventilation system at the time of site recommendation (DOE, 2002). The design of the ventilation system, however, is being modified by DOE. The staff does not have sufficient information on the new design to review the ventilation system.

4.1.7.2.3 Waste Package and Other Engineered Barriers

In addition to the waste package, other components of the engineered barrier system that may be used during preclosure operations at the potential geologic repository include a drip shield, drift invert, waste package pallet, and backfill. However, it is not clear whether these barriers will be considered as important to safety during the preclosure period. Designs of the waste package and engineered barrier system components incorporate 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 Lifetime of the Containers; Subissue 2, Effects of Phase Instability of Materials and Initial Defects on the Mechanical Failure and Lifetime of the Containers; and Subissue 6, Effects of Alternate Engineered Barrier Subsystem Design Features on Container Lifetime and Radionuclide Release from the Engineered Barrier Subsystem. The Design of Structures, Systems, and Components and Safety Controls that are safety related for the waste package and engineered barrier system 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 DOE site recommendation reference design (CRWMS M&O, 1999a) indicates 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 corrosion-resistant nickel alloy referred to as Alloy 22, surrounding an inner container made of Type 316 nuclear grade stainless steel (CRWMS M&O, 2000f). Fabrication processes used to construct the waste packages (e.g., forming, welding, and stress-relieving operations) may alter 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 also will contain several 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, 2000g). Each waste package will rest on an Alloy 22 emplacement pallet made of two V-shaped supports connected by hollow tubes with square cross sections. The waste package pallets will, in turn, rest on the drift invert. An inverted U-shaped drip shield, fabricated with titanium-palladium alloys (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, 2000h). The current repository reference design does not include engineered backfill. Drift degradation, however, may produce natural backfill in the postclosure period (Ofoegbu, 2000).

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. The ductility, fracture toughness, and impact strength of Alloy 22 are unlikely to be significantly affected by the fabrication processes necessary to construct the waste package outer container (Dunn, et al., 2004). 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 subsequent sections of this report, 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 system components are necessary to evaluate the DOE preclosure safety strategy. DOE provides information for the current designs of the waste packages and engineered barrier system components in CRWMS M&O (2000 f–h). Fabrication methods that may be used to construct the waste packages and engineered barrier system components also are provided in Bechtel SAIC Company, LLC (2001b,c). DOE has indicated that the potential license application design will use a Type 316 nuclear grade stainless steel inner container constructed to the requirements of the 2001 (with 2002 addenda) ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC (ASME, 2001a). The Alloy 22 outer container will not be an ASME stamped vessel (Brown, 2003a).

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 (Reamer, 2001a). Agreements were reached on specific topics concerning waste package design, inspection methods, variations in the mechanical properties of the waste package materials, and the effects of fabrication and repair on waste package performance. The postclosure performance of the engineered barrier system is addressed in Sections 5.1.3.1 and 5.1.3.2.

Overall, the available information, along with agreements reached between DOE and NRC (Schlueter, 2000; Reamer, 2001a,b,c), is sufficient to expect that the necessary information needed to assess the design of the waste package and engineered barrier system 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.

4.1.7.2.3.1 Waste Package Design Description

The waste package design consists of two concentric cylinders (i.e., disposal containers, fabricated from plate material). The meeting minutes of the February and May 2004 technical exchanges (Schlueter, 2004; Reamer, 2004) contain conceptual drawings of the waste package. The inner disposal container will be fabricated using Type 316 nuclear grade stainless steel, a minimum 50 mm [1.97 in] thick (Bechtel SAIC Company, LLC, 2001b). The inner disposal container will fit inside the outer disposal container constructed from Alloy 22. A radial gap of 0–4 mm [0–0.16 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, 2000f). The technical basis for the temperature used to establish these gaps, however, is not provided. Type 316 nuclear grade stainless steel is selected for the inner disposal container to provide mechanical integrity to the waste package during both the preclosure and postclosure periods of the potential repository. The selection of Alloy 22 as the outer disposal container material is 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 (CRWMS M&O, 2000g; DOE, 2002).

Several waste package configurations are needed to encapsulate the various commercial spent nuclear fuel wasteforms (CRWMS M&O, 2000g). 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. Moreover, additional waste package configurations are 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 also will 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 also are 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. 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, 2000g; DOE, 2002). 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 loading, the disposal containers will be sealed. The single closure lid for the Type 316 nuclear grade stainless steel inner container is held using a spread ring with a seal weld. A dual-closure lid design is used for the Alloy 22 outer disposal container (Brown, 2003b).

In summary, the waste package design description appears to be sufficient for use in developing a potential license application. The design of the waste package is still being developed, so DOE should provide additional design information in future documents.

4.1.7.2.3.2 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, 2000g). 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 performance-based limits, control criticality, and provide structural support to the spent nuclear fuel in the waste package. In addition, the materials used in the waste packages (internals) should be compatible with the wasteform, spent nuclear fuel cladding, and the waste package disposal container materials. The materials should not be reactive or pyrophoric.

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 for structural support and to maintain the desired geometric configuration. The internal structure should maintain the desired geometric configuration when subjected to mechanical loads to ensure criticality protection during handling, emplacement, and retrieval (CRWMS M&O, 2000g). In addition, the material used to provide criticality control should be compatible with the other materials and components inside the waste package and should not degrade the wasteform. 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 should provide additional design information in future documents.

4.1.7.2.3.3 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 being developed (Bechtel SAIC Company, LLC, 2001b). 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 (CRWMS M&O, 2000h). Individual segments of the drip shield are connected 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 to protect the waste package against rockfall during the postclosure period (Bechtel SAIC Company, LLC, 2001b; DOE, 2002).

DOE provided a conceptual design description for the drip shield, including the materials of construction, and configuration and the method of emplacement. Details of the fabrication methods have yet to be provided, however. DOE should provide additional design information in future documents. A discussion of the ability of the proposed drip shield to withstand mechanically disruptive events for the postclosure period is provided in Section 5.1.3.2.

4.1.7.2.3.4 Waste Package Pallet

The waste package pallet is designed using Alloy 22 plate material (CRWMS M&O, 2000h). Each waste package pallet has two V-shaped supports connected by hollow, square cross-sectional tubing. 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 waste package static loading and its lifting during handling operations (CRWMS M&O, 2000i,j). Results of analyses supporting these structural evaluations are reported using stress intensity values as defined by ASME (2001b, Subparagraph NB–3213.1). It is not clear if the normal stress components generated at the contact interface between the waste package and pallet are considered when calculating the stress intensity results presented in the reports. Seismic loads are not addressed in lifting a loaded pallet structural evaluation. DOE should either assess the effects of seismic loads on a loaded pallet for all relevant handling operations or justify the exclusion. Similarly, DOE should assess the potential consequences of dropping a loaded emplacement pallet or provide the basis for excluding this particular event from consideration.

4.1.7.2.3.5 Disposal Container Fabrication and Closure

DOE has indicated that the Type 316 nuclear grade stainless steel inner disposal container will be fabricated according to ASME (2001a, with 2002 addenda) and will be a nuclear or N-stamped vessel. The Alloy 22 outer barrier will be fabricated to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC, however, the outer container will not be an ASME stamped vessel (Brown, 2003a).

Filler materials used in welding processes should conform to the requirements specified in ASME (2001c, Section II, Part C) or equivalent. 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 (Bechtel SAIC Company, LLC, 2001a). The weld filler material for the Alloy 22 outer container will be ERNiCrMo-14 (Brown, 2003c).

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 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 inside the outer container. The bottom lid is then fit and welded in place. The trunnion collar sleeve is 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 fabrication processes. Water quenching will be used to reduce the temperature of the Alloy 22 outer container to below 800 °C [1,472 °F] in approximately 4 minutes (Bechtel SAIC Company, LLC, 2001b). DOE has not developed design criteria for residual stress mitigation by solution annealing.

To reduce residual stresses in the Alloy 22 final closure welds, DOE indicates it will use laser peening or low-plasticity burnishing of the outer Alloy 22 closure lid weld. A description of the process and the application of laser peening to the waste package closure weld is reported by Chen (2002). Measurements on Alloy 22 welds show compressive residual stresses can be created in the near surface layers by laser peening. No residual stress mitigation methods will be used for the inner Alloy 22 closure lid weld or the inner Type 316 nuclear grade stainless steel spread ring seal welds. DOE has not developed design criteria for residual stress mitigation of the waste package closure weld using either laser peening or low-plasticity burnishing.

The combination of cold work used in forming and machining operations and elevated temperature exposures caused by welding and annealing processes of the Alloy 22 waste package outer container may precipitate 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 corrosions 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. DOE has indicated that during the preclosure period, however, corrosion may not be a significant issue.

DOE evaluated the thermal stability of nickel-chromium-molybdenum alloys using several criteria: (i) microstructural examination for the presence of secondary phase precipitates at the grain boundaries or in the interdendritic regions of the welds, (ii) intergranular corrosion susceptibility, and (iii) mechanical properties such as ductility, yield strength, and impact toughness. Heubner, et al. (1989) provides 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) report 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 is observed after 1 hour at 800 °C [1,472 °F], based on the results of standardized tests (ASTM International, 1999). Bulk precipitation of topologically close-packed phases is 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, 2000k; Rebak, et al., 2000; Summers, et al., 2002). Table 4.1.7-1 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 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

Table 4.1.7-1. Relationship Between Alloy 22 Condition, Ductility, Impact Resistance, and Corrosion Rate Using ASME 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]

*ASTM International. "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. Vol. 3.02. West Conshohocken, Pennsylvania: ASTM International. 2001.

the impact energy for material in the solution-annealed condition. The reduction in area measured on tensile test specimens decreases slightly from 75 percent following thermal aging. Complete grain boundary precipitation is required for significant decreases in ductility or impact toughness. The activation energy necessary to decrease the impact energy to 203 J [150 ft·lb] is 247 kJ/mol [59 kcal/mol].

At 760 °C [1,400 °F], an exposure of 10 hours is required to decrease the Charpy impact energy to 203 J [150 ft·lb] (Rebak, et al., 2000). 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]. Dunn, et al. (2004) reported welded Alloy 22 remains ductile and resistant to fracture even after thermal aging at 870 °C [1,598 °F] for a period of 1 hour. Solution annealed welds contained a slight volume fraction of topologically close-packed phases but retained high fracture toughness and impact strength. These results have led DOE to conclude that waste package fabrication processes will not significantly degrade the mechanical properties.

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. Systematic studies about the effects of compositional variations of Alloy 22 on thermal stability show 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. Trace elements such as sulfur, cobalt, and carbon also may alter thermal stability and mechanical properties of the base and filler metals.

Additional information 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 information may result in unanticipated variations in waste package corrosion resistance and mechanical properties. To address these concerns, DOE agreed (Reamer, 2001b) to provide justification that the ASME 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 the Alloy 22 base metal and the allowed compositional variations in the weld filler metals used in fabrication of the waste packages. In addition, DOE agreed (Reamer, 2001b) 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 indicates future work will include developing and testing of welding, heat treating, and inspecting 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. Fabrication processes can alter the microstructure decrease localized corrosion resistance. The formation of topologically close-packed phases as a result of fabrication processes is unlikely to significantly alter the mechanical properties of the waste packages, however, the effects of compositional variations in the base and filler metals should be evaluated. With the DOE agreement to provide the additional information, sufficient information should be available at the time of a potential license application.

4.1.7.2.3.6 Nondestructive Evaluation of the Disposal Container and Closure Welds

Fabrication of the outer and inner cylinders will require longitudinal and circumferential seam welds. Prior to forming and welding, the Alloy 22 base plate will be examined using ultrasonic testing. Fabrication welds for the Alloy 22 outer cylinder will be examined using liquid-penetrant, radiographic, and ultrasonic testing techniques. The Type 316 nuclear grade stainless steel inner vessel welds will be nondestructively examined using liquid-penetrant, radiographic, and ultrasonic testing. After nondestructive testing, the Type 316 nuclear grade stainless steel inner container will be pressure tested and helium leak tested (Brown, 2003a).

The waste package design for the potential license application will have a single Type 316 nuclear grade stainless steel closure lid and dual Alloy 22 closure lids (Brown, 2003b). 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. This machine will provide a redundant check of the visual inspection for the weld preparations. 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 that will identify problems with the welding process. It may be possible to perform some repairs 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 (Bechtel SAIC Company, LLC, 2001c).

The inner disposal container lid, made of Type 316 nuclear grade stainless steel, will be held using a shear ring with a seal weld. The shear ring will be assembled from three or four segments and welded in place. The 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 the closure cell control system will notify the operator immediately of any parameter anomalies. 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 (Bechtel SAIC Company, LLC, 2001c) 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 welded to the outer barrier using a fillet weld. The gas-tungsten arc welding method is presently being considered for remote welding of this lid (Bechtel SAIC Company, LLC, 2001c). 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 done by visual inspection and eddy current testing.

Prior to installation of the outer lid, remote visual inspection of the weld preparation surfaces will be used to ensure 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 by gas-tungsten arc welding. Nondestructive evaluation of the weld will be performed. The inspection will require two passes (rotations). A remote visual examination will be performed, followed by a volumetric inspection using ultrasonic testing and a couplant.

To experimentally determine the minimum detectable flaw size using ultrasonic testing, DOE fabricated two Alloy 22 mockups using 25-mm [1-in]-thick material. Examinations were performed at several scanning angles to determine the optimum scanning orientation. DOE indicates that planar-type flaws (i.e., fusion and penetration) with a minimum area of 16 mm^2 [0.025 in^2] can be detected in the tested weld joint geometry. DOE concludes that the inability to detect small volumetric porosity reflectors may be acceptable because the geometric discontinuities associated with the individual gas pores do not cause localized stress increases that appreciably affect the initiation of stress corrosion cracking or mechanical failure.

Recent information in Bechtel SAIC Company, LLC (2003b) includes a comparison of nondestructive evaluation methods for the inspection of Alloy 22 waste package closure welds. The size and geometry of the closure weld specimens are designed to duplicate the configuration of the waste package for 21 pressurized water reactor fuel assemblies. Specimens are remotely welded using the gas-tungsten arc welding process. The remote welding operation used to fabricate the test specimens is similar to closure welding operations to be performed in the closure cell facility of the Waste Handling Building at the potential repository site (Bechtel SAIC Company, LLC, 2001c). After welding is completed, the specimens are examined using four nondestructive examination methods. Volumetric examinations are conducted using ultrasonic and radiographic testing. Surface inspection is conducted with liquid penetrant and eddy current testing. DOE has indicated that inspection of the waste package closure welds most likely would be performed using ultrasonic and eddy current testing because radiographic and liquid penetrant testing will not be possible owing to the waste package design and anticipated temperature constraints. Nevertheless, radiographic and penetrant testing are included in the study to provide a comparison with the ultrasonic and eddy current methods.

Standard metallurgraphic techniques were used to characterize volumetric flaws identified in the nondestructive examinations. The characteristics included size and position and each flaw was classified as either a round or a linear flaw. Good agreement is found between the ultrasonic and radiographic test methods. Identification and characterization of surface flaws using penetrant and eddy current testing are similar. Several linear flaws were identified in the welded specimens by ultrasonic and radiographic methods, and most of these linear flaws occur because of lack of fusion between weld passes. The size of the indications varied from approximately 3 to 38 mm [0.12 to 1.5 in] in length. Considering the total length of the weld material and the cumulative length of all flaws, a total flaw of 0.16 percent is determined from ultrasonic test results. In addition to the flaws identified using the ultrasonic and radiographic test methods, porosity also is identified in the metallurgical analyses. The pores are less than 1 mm [0.04 in] in diameter, rounded, and, therefore, unlikely to promote cracking. Clustering of the pores is not observed in any welded specimens.

The recent information in Bechtel SAIC Company, LLC. (2003b) indicates ultrasonic testing can be used to detect flaws in Alloy 22 welds. Testing was conducted on the simulated closure welds (Bechtel SAIC Company LLC, 2003b), however, that are not representative of the current waste package closure weld design (Brown, 2003b). In the revised waste package design, ultrasonic examination from the outer diameter surface of the waste package would not be possible because of a trunnion welded to the Alloy 22 outer container. Change in the waste package design means linear flaws such as lack of fusion defects may be more difficult to detect.

In summary, DOE agreed (Reamer, 2001b) 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 adversely may 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 (Bechtel SAIC Company LLC, 2003b, 2001c). The effects of recent waste package design changes on the use of ultrasonic testing to detect flaws in the Alloy 22 closure welds should be evaluated.

4.1.7.2.3.7 Criticality Design Criteria

10 CFR 63.112(e)(6) requires that the preclosure safety analysis include an analysis of the performance of the structures, systems, and components that provide means to prevent and control criticality. 10 CFR 63.112(f) requires a description and discussion of the design and the relationship between the design bases, the design criteria, and the preclosure performance objectives. In its review of the preliminary preclosure safety assessment (DOE, 2001), the staff identified two general issues. The first is 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. The second includes 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, Category 1 and 2 events with respect to criticality) when designing the surface and subsurface facilities.

According to NRC Regulatory Guide 3.71 (NRC, 1998), burnup of the spent nuclear fuel assemblies should be verified through measurements before the assemblies can be loaded into the waste packages, if credit is taken for the burnup when designing the criticality control system of the waste package. During the preclosure technical exchange (Reamer, 2001b), DOE agreed to provide an approach to verify fuel assembly burnup. DOE stated that burnup credit is being sought only for commercial spent nuclear fuel, and that the best source of burnup information for the majority of these fuel assemblies is that developed and available through reactor records maintained in accord with NRC-accepted quality assurance requirements. Reactor records are a more accurate source of fuel assembly burnup data than physical measurements. Measurements may be needed to verify the burnup indicated by reactor records.

Several waste package internal component configurations are considered in determining 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, 2000g). Although the configurations with degraded baskets are more significant for postclosure performance than for preclosure performance, 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 states that established design requirements "... preclude preclosure criticality unless two unlikely independent events occur [e.g., CRWMS M&O (2000l)]. The probability of two unlikely independent events occurring will be less than 10^{-6}/yr." While 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, may not require the licensee to quantify the probability of the unlikely events, under 10 CFR Part 63 events must be identified, their probabilities quantified, and designations assigned as Category 1 or 2 events. 10 CFR 63.112(e)(6) also requires an analysis of the performance of structures, systems, and components to control and prevent nuclear criticality. Therefore, DOE has indicated that the repository preclosure structures, systems, and components will be designed to prevent criticality under normal operation and Categories 1 and 2 events (Reamer, 2001b).

4.1.7.2.3.8 Waste Package Shielding

The current site recommendation waste package design does not provide additional shielding for worker protection (CRWMS M&O, 1999b). It is intended the waste package containment barriers provide sufficient shielding to protect the waste package materials from radiation-enhanced corrosion (CRWMS M&O, 2000g). As calculated by DOE, 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, 2000f). Shielding for worker 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 provide information on the shielding to prevent radiolysis-induced corrosion that is sufficient to use in developing a potential license application. Protection for workers is provided by other structures, systems, and components. The design of the waste package is still being developed. DOE should provide additional information as the design of the waste package is further developed.

4.1.7.2.3.9 Designing for Normal Operation and Categories 1 and 2 Event Sequences

DOE identifies event sequences presently being considered in establishing the design criteria and specifications for important to safety structures, systems, and components (DOE, 2001). A detailed discussion of the DOE identification and categorization of event sequences that pertain to the preclosure period of the potential repository can be found in Subsections 4.1.3 and 4.1.4. The discussion presented in this section is limited to the postulated waste package drop event.

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, 2000f). Analyses intended to support this characterization have been performed (CRWMS M&O, 2000m). These analyses are 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

(Reamer, 2001b) 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.

The means used to demonstrate the ability of the waste package to withstand the postulated event sequences is at the discretion of DOE. DOE has chosen to use numerical simulations based on the finite-element method as the sole basis for its safety case, demonstrating the ability of the waste package to withstand handling drops without breaching. Although DOE has not precluded using actual waste package drop tests in the future to demonstrate the structural integrity of the waste package, there are no specific plans at this time.

DOE agreed (Reamer, 2001b) to (i) demonstrate 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 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 variabilities, (iii) ground motion effects caused by a seismic event (waste package drops and tipovers are more likely to occur during seismic events), and (iv) sliding and inertial effects of the spent nuclear fuel.

DOE addressed the aforementioned concerns in its response to key technical issue agreement PRE.07.02 (Bechtel SAIC Company, LLC, 2003c). Reviews of the (i) methodology proposed by DOE to assess adequacy of a given finite-element model discretization and (ii) proposed structural failure criteria are provided in Section 5.1.3.2.4.4.

The methodology employed by DOE to assess the potential effects of residual stresses created in the waste package by the various fabrication processing steps is presented in Bechtel SAIC Company, LLC (2003c, Section 3.2.1.2). Discussion is limited to the potential residual stresses arising from the solution annealing and quenching processes proposed to generate compressive stresses on the exterior surface of the waste package outer shell prior to emplacement of the spent nuclear fuel or high-level waste and installation of the closure lids. Although quenching may be performed on the exterior surface only or on both the interior and exterior surfaces simultaneously, the report only discusses the latter.

DOE acknowledges the study of residual stress effects on waste package performance is limited by the through-wall, finite-element model discretization used for the waste package outer shell in that only four, one-point-integration solid elements are used. Moreover, DOE acknowledges the one-point-integration solid elements are not formulated to represent residual stress distributions in an accurate manner. DOE points out, however, this modeling approach for assessing the effects of residual stresses was chosen because DOE wants to maintain consistency with the models that have been used to evaluate responses of the waste package to various design basis events that do not explicitly include the presence of these residual stresses.

The DOE report presents results of the study that compared the maximum stress intensity, the maximum effective plastic strain, and the size of the damaged area with and without residual stresses included in the analyses. The ASTM International standard for Alloy 22 (1998) indicates this material has a minimum elongation in 50 mm [2 in] of 45 percent. If it can be shown that a significant loss of material ductility does not result from the residual stresses created within the waste package outer shell during its fabrication, residual stresses are not likely to appreciably affect the design basis loads that could cause a breach by plastic collapse. The basis for the residual stress distribution used in the assessment is presented in Herrera, et al. (2002). Because the finite-element mesh discretization through the thickness of the waste package outer shell used to assess the potential effects of residual stresses was constructed using only four, equally sized, single-integration-point solid elements, the compressive and tensile residual stresses should be defined as having equal magnitudes for the model to be in a state of equilibrium before applying the design basis loads. Thus, the model is not capable of representing the distinct maximum compressive and tensile residual stresses at the same time. Nevertheless, this deficiency is only relevant if a significant loss of material ductility can be expected to occur because of the presence of these residual stresses. Residual stresses in the waste package outer shell may significantly effect the potential for stress corrosion cracking during the postclosure period.

A summary of the methodology employed by DOE to assess the potential effects of material strain rates on the waste package response to dynamic loads is presented in Bechtel SAIC Company, LLC (2003c, Section 3.2.2). DOE indicates strain rate data for the waste package inner and outer shell materials (i.e., Type 316 SS and Alloy 22) are not readily available. As a result, the potential effects of material strain rate variability are studied parametrically using the strain rate characteristics of Type 304 SS to establish the adjusted inner and outer shell material constitutive models.

The tangent moduli for both waste package shell materials are assumed to be unaffected by strain rate, consistent with the behavior of Type 304 SS. The range of material strain rates evaluated is reported to be 20–900 per second. The effects of strain rate on the waste package response to tipover from an elevated surface is summarized by comparing the results obtained from the finite-element analyses of this design basis event. According to Levin, et al. (1999, Figure 5), however, it would appear a potential loss of ductility for Alloy 22, at least for relatively high strain rates, does, in fact, exist. As a result, justification may be necessary for not considering the potential loss of ductility for both Type 316 SS and Alloy 22 for the full range of strain rates these materials are expected to experience during various design basis events.

Also, it was not clear in the agreement response whether the constitutive models employed within the finite-element models used to assess the potential effects of strain rates on the response of the waste package accommodated the spatial variability of the strain rate. It is expected waste package materials will experience significant strain rate spatial variations when subjected to dynamic loads. As a result, the applicable material strengths and corresponding stress-strain relationships will vary spatially. In other words, the constitutive relationships implemented within the finite-element models should define explicitly the material yield and ultimate strengths by strain rate. Lastly, the spatially varying material strengths should be considered when assessing the potential for failure. This information should be provided at the time of the potential license application.

DOE indicates in Bechtel SAIC Company, LLC (2003c, Section 3.2.3) the effects of waste package dimensional variability will be accounted for by assuming the thicknesses of the inner and outer shells are the minimum allowable, as defined by the waste package allowable tolerances, in the finite-element models. In addition, future design drawings will indicate the applicable dimensional tolerances.

DOE indicates in Bechtel SAIC Company, LLC (2003c, Section 3.2.3) the effects of waste package material variability are accounted for by assuming the minimum yield and ultimate tensile strength values available from the applicable codes and standards [e.g., the ASME Boiler & Pressure Vessel Code (2001c)] for the inner and outer shell materials in the finite-element models. It is noted, however, the analyses presented in Bechtel SAIC Company, LLC (2003d,e) uses minimum elongation values approximately 50 percent greater than the applicable ASTM International standard for Alloy 22 (1998). As a result, justification for using minimum elongation values exceeding the applicable ASTM International standards may be necessary to support the potential license application.

DOE indicates in Bechtel SAIC Company, LLC (2003c, Section 3.2.4) that it has been assumed fixtures will be provided to restrain the waste package in the surface facilities during preclosure handling operations so no damage will be incurred by the waste package during a seismic event. Therefore, evaluations of the waste package responses to seismic events during the preclosure period are limited to in-drift conditions after emplacement. The report also indicates the vibratory ground motions used for the evaluation represent seismic events that have an annual exceedance frequency of 5×10^{-4} per year (i.e., a 2,000-year return period). Use of a 5×10^{-4} annual exceedance frequency seismic event as the preclosure design basis, which has been informally discussed by DOE during DOE and NRC technical interactions, has not been formally presented by DOE nor formally accepted by the NRC staff.

Justification is not provided for the assumption that restraints will be sufficient to prevent damage to the waste package during preclosure handling operations when subjected to a seismic event.

A summary of the methodology employed by DOE to establish the potential initial tipover velocities that may be experienced by the waste package if the tipover is initiated by a seismic event is provided in Bechtel SAIC Company, LLC (2003c, Section 3.2.5). Using the conservation of energy principal, a mathematical relationship is developed to approximate the rotational velocity of the waste package at the time of impact. This relationship includes consideration of the initial tipover velocity of the waste package. A series of analyses was performed using a range of initial tip-over velocities spanning 0–1.62 rad/s. This range of initial tipover velocities is consistent with horizontal ground motion velocities varying from 0 to 4.38 m/s [0 to 14.4 ft/s]. The methodology proposed by DOE to establish the initial waste package tipover velocity that could result from a seismic event is consistent with standard engineering practice. The report notes the initial waste package tipover velocities considered in analyses performed to date are based on repository horizon vibratory ground motions. No discussion is provided, however, addressing the potential effects of the vertical motion of the floor created by the seismic event on the level of damage incurred by the waste package during a tipover event in the region of impact. This information may be necessary for the staff to make a determination on adequacy of the potential license application.

DOE indicates (Bechtel SAIC Company, LLC, 2003c, Section 3.2.6) the sliding and inertial effects of the waste package contents are evaluated in calculations where they are anticipated to affect performance of the waste package inner and outer shells. Depending on whether DOE intends to take credit for the structural integrity of the cladding for preclosure or postclosure performance, explicitly including the sliding and inertial effects of the waste package contents may be necessary to demonstrate the loads incurred by the wasteform are not sufficient to cause appreciable damage.

In summary, DOE has provided sufficient information on its methodologies for use in developing a potential license application for (i) developing adequate finite-element model mesh discretizations, (ii) establishing differential thermal expansion gaps between the waste package inner and outer shells, (iii) assessing residual stress and dimensional and material variability effects on waste package response to preclosure design basis events, (iv) evaluating the response of the waste package to preclosure seismic events, and (v) approximating the initial waste package tipover velocities as a function of the ground motion initiating the tipover. Sufficient information concerning material failure criteria has also been provided.

The waste package drop analyses DOE performed (CRWMS M&O, 2000m), however, do not indicate if the structural integrity of the spent nuclear fuel was considered when establishing allowable drop heights. At the preclosure technical exchange (Reamer, 2001b), DOE stated that, in case of a drop, an assessment would be made if the wasteform must be repackaged, but the primary consideration when establishing drop heights is the integrity of the waste package. DOE also noted the repackaging requirements have not yet been established, however, they will be based on long-term performance needs.

4.1.7.2.3.10 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, 2000g). With the exception of waste packages with 24 boiling water reactor fuel assemblies, 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 commercial spent nuclear fuel waste package configurations, the 21 pressurized water reactor fuel waste packages with absorber plates have the highest heat output, an average of 11.33 kW [38,650 BTU/hr] (CRWMS M&O, 2000g). Peak cladding temperatures are calculated by DOE to be less than 300 °C [572 °F], even with close waste package spacing (CRWMS M&O, 2000f). The heating, ventilation, and air conditioning system within the Waste Handling Building in intended to maintain fuel cladding temperatures within acceptable limits before packaging and emplacement.

The current DOE waste package design description appears to contain sufficient information concerning components to provide thermal control so the fuel cladding temperature will be maintained within acceptable limits. Since the design of the waste package is still being developed, however, staff cannot currently determine the sufficiency of information to evaluate the thermal control on fuel cladding temperature. Staff will, therefore, look to the information that is available at the time of a potential license application.

4.1.7.3 Summary and Status

4.1.7.3.1 Surface Facilities

DOE has provided only conceptual designs and operational features for the dry transfer facility, canister handling facility, and fuel handling facility. DOE has provided information only on the capacity and location of the aging facilities. DOE has not discussed the design basis and details for the types of structures, systems, and components and equipment that will be used at the aging facilities. This information is not sufficient for a staff assessment of the surface facilities.

4.1.7.3.2 Subsurface Facilities

DOE has provided information regarding the drift design, ground-support systems, location of ramps, and ventilation shafts. However, DOE has provided only minimal information on the design basis and details for the waste package transportation and emplacement equipment. Furthermore, DOE has not provided sufficient information on the design basis for the subsurface ventilation system. This information is not sufficient for a staff assessment of the subsurface facilities.

4.1.7.3.3 Waste Package and Other Engineered Barriers

Staff has reviewed the design methodology for the waste package and have not identified any major concerns for preclosure activities. The staff noted, however, DOE has not supplied sufficient information about the following:

- The final waste package design bases and their relationships to the design criteria
- The final waste package design and specifications
- Design criteria for residual stress mitigation of the waste package fabrication welds by solution annealing
- Design criteria for residual stress mitigation of the waste package closure weld using either laser peening or low-plasticity burnishing
- 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

4.1.7.3.4 Status of Key Technical Issue Agreements

Table 4.1.7-2 provides the status of agreements related to the preclosure design of structures, systems, and components important to safety and safety controls. The agreements listed in the table are associated with reviews described in this section.

Table 4.1.7-2. Summary of Resolution Status for Design of Structures, Systems, and Components Important to Safety and Safety Controls Preclosure Topic

Preclosure Item	Status	Related Agreement*	Note
Relationship between the Design Criteria and Design Basis and the Regulatory Requirements	Staff review incomplete	†	—
Geologic Repository Operations Area Design Methodologies	Staff review incomplete	†	—
Assumptions, Codes, and Standards for Surface Facilities Design	Staff review incomplete	†	—
Materials for Surface Facilities Design	Staff review incomplete	†	—
Load Combinations for Surface Facilities Design	Staff review incomplete	†	—
Surface Facilities Design Analyses and Documentation	Staff review incomplete	†	—
Assumptions, Codes, and Standards for Subsurface Facilities Design	Staff review incomplete	†	—
Subsurface Operating Systems Design	Staff review incomplete	†	—
Material and Material Properties for Subsurface Facilities Design	Complete	RDTME.3.01‡	Section 4.1.7.2.2.2
Load Combinations for Subsurface Facilities Design	Staff review incomplete	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 Facilities Design	Staff review incomplete	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

Table 4.1.7-2. Summary of Resolution Status for Design of Structures, Systems, and Components Important to Safety and Safety Controls Preclosure Topic (continued)			
Preclosure Item	**Status**	**Related Agreement***	**Note**
Subsurface Ground-Support Systems Design	Staff review incomplete	RDTME.3.06 RDTME.3.09	Drift invert stability and rock support system analyses
Subsurface Ventilation System Design	Staff review incomplete	RDTME.3.14	Ventilation modeling and validation
Subsurface Power and Power Distribution Systems Design	Staff review incomplete	†	—
Maintenance Plan for Subsurface Facilities	Staff review incomplete	†	—
Waste Package and Engineered Barrier System Design	Staff review incomplete	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 review methods.
†Not discussed at the first DOE and NRC Technical Exchange and Management Meeting on Preclosure Safety [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. ML021340719. Washington, DC: NRC. 2001. <www.nrc.gov/waste/hlw-disposal/public-involvement/mtg-archive.html#KTI>]
‡No further concerns at this time.

4.1.7.4 References

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4.1.8 Meeting the 10 CFR Part 20 As Low As Is Reasonably Achievable Requirements for Normal Operations and Category 1 Event Sequences

The plans to meet as low as is reasonably achievable requirements that are required to be submitted as part of a potential license application have not been the subject of DOE and NRC prelicensing discussions and no issues have been identified.

4.2 Plans for Retrieval and Alternate Storage of Radioactive Wastes

The plans for retrieval and alternate storage that are required to be submitted as part of a potential license application have not been the subject of DOE and NRC prelicensing discussions and no issues have been identified.

4.3 Plans for Permanent Closure and Decontamination, or Decontamination and Dismantlement of Surface Facilities

The plans for permanent closure and decontamination, or decontamination and dismantlement of surface facilities, that are required to be submitted as part of a potential license application have not been the subject of DOE and NRC prelicensing discussions and no issues have been identified.

5 REPOSITORY SAFETY AFTER PERMANENT CLOSURE

5.1 Performance Assessment

5.1.1 System Description and Demonstration of Multiple Barriers

5.1.1.1 Description of Issue

Postclosure performance objectives specified in 10 CFR Part 63 require a system of multiple barriers consisting of at least one engineered and one natural. As defined in the regulations, a barrier is any material, structure, or feature that prevents or substantially delays movement of water or radionuclides. Thus, any potential U.S. Department of Energy (DOE) license application must identify and describe the capabilities of the repository barriers. Examples of potential natural barriers at Yucca Mountain include the unsaturated and saturated volcanic and alluvial rock units that affect movement of water or radionuclides by processes such as infiltration, matrix diffusion, and sorption. Engineered barriers the 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 has the potential to provide additional assurance the postclosure performance objectives can be met. The description of each barrier capability provides an overall understanding of the contribution of the barrier to the DOE demonstration of compliance with 10 CFR Part 63 and how the different sorts of barriers enhances the resiliency of the repository system. The result of the multiple barrier review is a staff understanding of each barrier waste isolation capability, which will influence the emphasis placed on the reviews of scenario analysis and event probability and on model abstraction.

As provided in 10 CFR Part 63, the potential DOE license application is required to identify the barriers, describe the capabilities of each barrier, and provide the technical bases for the capabilities of the barriers in a manner consistent with the technical basis used to support the 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 system.

- The engineered barrier system must be designed so that, working in combination with natural barriers, release of radionuclides from the repository is within the limits specified in 10 CFR Part 63, Subpart L. Compliance must be demonstrated through a performance assessment that meets the requirements specified in 10 CFR 63.114.

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 system, 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 the technical bases for descriptions of the capabilities of the 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 performance assessments used to demonstrate compliance with 10 CFR 63.113(b) and (c).

Consistent with 10 CFR Part 63, the review of multiple barriers in U.S. Nuclear Regulatory Commission (NRC) (2003) focuses on the demonstration of multiple barriers and includes (i) identification of design features of the engineered barrier system and natural features of the geologic setting considered barriers important to waste isolation, (ii) descriptions of the capabilities of the barriers to isolate waste, and (iii) description of the technical basis for each barrier capability.

This section provides a review of the multiple barrier analysis presented in the DOE performance assessment for site recommendation (CRWMS M&O, 2000a) and agreements reached with DOE. The staff review is limited to evaluation of information supporting the DOE methodology. Compliance with the standards in 10 CFR Part 63 for individual and ground water protection and human intrusion is not considered in prelicensing issue resolution. Comments describe the staff expectation of the contents of the DOE performance assessment in the potential license application and supporting documents that will allow an independent review of the performance assessment results and methodology.

5.1.1.2 Relationship to Key Technical Issue Subissues

All key technical issue subissues contribute to (i) identification of design features of the engineered barrier system and natural features of the geologic setting, (ii) descriptions of the capabilities of the barriers, and (iii) description of the technical basis for each barrier capability.

5.1.1.3 Importance to Postclosure Performance

The concept of multiple barriers (i.e., engineered and natural barriers) is integral to geologically disposing high-level waste, developing risk insights, and understanding postclosure performance. For example, the safety of geologic disposal is enhanced if the system includes (i) a long-lived waste package that retains its integrity during the period of the highest thermal output of the waste when the wasteform behavior is most uncertain because of potentially high temperatures, (ii) slow release rates of radionuclides from the engineered barrier system once the waste packages are breached, and (iii) slow travel of released radionuclides from the engineered barrier system to the area where potential exposures might occur. Multiple barriers, as an element of a defense-in-depth approach, result in a robust repository system more tolerant of failures and external challenges (e.g., poor or highly degraded performance of more than one barrier would have to occur to have a significant effect on overall safety).

The risk insights contained in Appendix D were developed within the multiple barrier context (i.e., understanding the significance to waste isolation of the long-lived waste package, release rates of radionuclides, and transport of radionuclides in the context of the effect on risk estimates). The staff grouped the risk insights into three categories of relative significance (high, medium, and low) based on preliminary evaluations of the contribution to, or effect on, the waste isolation capabilities of the repository system. The risk insights, and their relative significances, are in general used in reviewing the DOE approach to the treatment of multiple barriers.

The description of each barrier capability provides information that helps understand the performance assessment results. Each barrier waste isolation capability (e.g., the attributes of a particular barrier and the effect these have on waste isolation) and an understanding of the features, events, and processes that could significantly degrade each barrier capability influence the reviews presented in the Scenario Analysis and Event Probability, and Model Abstraction sections of this report.

5.1.1.4 Technical Basis

NRC developed a review plan (NRC, 2003) consistent with the acceptance criteria and review methods found in the previous issue resolution status reports. A status assessment of the DOE approach for multiple barriers is provided in the following subsections. This assessment is organized according to the three review methods identified in Section 2.2.1.1.2 (NRC, 2003): (i) Identification of Barriers, (ii) Description of Barrier Capability, and (iii) Technical Basis for Barrier Capability. The information resulting from these three review methods is used to guide the staff reviews conducted in the Scenario Analysis and Event Probability and Model Abstraction sections, performance assessment, and the performance confirmation program.

5.1.1.4.1 Identification of Barriers

This section addresses solely the information available on the DOE approach to multiple barriers important to waste isolation (e.g., affect movement of water or radionuclides), with at least one engineered and one natural barrier.

DOE documents its current approach to identifying natural and engineered barriers in Civilian Radioactive Waste Management System Management and Operating Contractor (CRWMS M&O) (2000a,b). DOE identifies four natural barriers and five engineered barriers. Natural barriers consist 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 consist of (i) the titanium drip shield, (ii) the C-22 waste canister, (iii) the commercial spent nuclear fuel cladding, (iv) the wasteform (e.g., high-level waste glass), and (v) a drift invert (e.g., crushed tuff). DOE states the capabilities of these 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. A presentation, Total System Performance Assessment and Integration (TSPAI) Key Technical Issue Subissue 1—Multiple Barriers, given at the technical exchange (Reamer, 2001) provides additional understanding of the DOE multiple barriers approach and future plans to support the DOE performance assessment.

Overall, the available information, along with key technical issue agreements between DOE and NRC is sufficient to expect that the information necessary to assess the identification of barriers will be available at the time of a potential license application.

5.1.1.4.2 Description of Barrier Capability

DOE has indicated that it will provide a risk-informed description for the waste isolation capability of each barrier that includes

- Attributes or functions of each barrier and the relationship of those attributes or functions to the effectiveness of each barrier to isolate waste (e.g., sorptive properties of a rock unit and corrosion resistance of the waste package material)

- Independent and interdependent capabilities of the barriers

DOE documents its approach to describing the capability of natural and engineered barriers in CRWMS M&O (2000a,b). In CRWMS M&O and DOE (2001), DOE states barrier importance analysis is used in conjunction with sensitivity analysis to demonstrate barrier capability. Barrier importance analysis encompasses (Andrews, 2000) (i) evaluation of the significance of parameter and model uncertainty, (ii) evaluation of the 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 were performed: degraded barrier importance analysis and neutralized barrier importance analysis. The degraded barrier importance analysis fixes several parameters associated with a barrier at the 95th percentile (or at the 5th percentile, if that leads to maximizing the dose rate) values in the total system performance assessment model and reruns the probabilistic analyses. For the neutralized barrier importance analysis, the function of a barrier is eliminated by setting selected parameters in a way that corresponds to omission (i.e., neutralization) of a process-model factor, or equivalently (in most cases), a barrier. DOE points out 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.

The NRC review of the two DOE documents describing the demonstration of multiple barriers (CRWMS M&O, 2000a,b) identified several concerns regarding descriptions of the barrier capabilities. Although 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; DOE presented the capabilities of the barriers primarily in terms of dose. In the documents reviewed, DOE did not provide a discussion relating the dose curves to the specific barrier capabilities. As discussed previously, descriptions of the barrier capabilities need to discuss the attributes of the barriers that provide the waste isolation function, and discuss the uncertainties.

NRC presented the preceding concerns to DOE, and general agreements were reached at the DOE and NRC Total System Performance Assessment and Integration Issue Resolution Meeting (Reamer, 2001). For TSPAI.1.01, DOE agreed to provide an enhanced descriptive treatment for presenting barrier capabilities in its final approach for demonstrating multiple barriers. For TSPAI.1.02, DOE agreed to provide a discussion of the following when documenting barrier capabilities: (i) independent and interdependent capabilities of the barriers (e.g., including a differentiation of the capabilities of barriers performing similar functions) and (ii) barrier effectiveness with regard to individual radionuclides.

Subsequent to the agreements, DOE provided a report (Bechtel SAIC Company, LLC, 2002a) that discusses results of extensive total system performance assessment studies of the effects of changes in parameter values, including those changes outside the range used in the baseline performance assessment, either singly (e.g., neptunium and plutonium solubility, in-package pH, and such) or grouped to represent the pessimistic assumption for the entire model components or barriers. Barriers are neutralized individually or in combination. Results are given for the arithmetic mean values of doses based on the entire inventory or for the most significant radionuclide contributions to dose. Conclusions regarding the potentially significant factors agree generally with those already found in the CRWMS M&O (2000a) analysis. Additional insight into the DOE treatment of risk information and multiple barriers is derived from presentations at management meetings (Ziegler, 2003) and a technical exchange (McCartin, 2004) between DOE and NRC.

In response to TSPAI.1.01, DOE provided Bechtel SAIC Company, LLC (2002b). This document provides an overview of the approach DOE plans to use in its total system performance assessment license application. For barriers important to waste isolation, the description will focus on barrier capabilities to limit the movement of water or radionuclides. The description will include discussions of model and parameter uncertainties as well as temporal and spatial variabilities. Quantitative analyses would be incorporated into the description of multiple barriers, when appropriate. By using the quantitative results directly from the total system performance assessment license application (not from any hypothetical extreme scenario or degraded barrier simulation), DOE asserts they can account for the uncertainty in barrier characteristics and barrier interdependence. Bechtel SAIC Company, LLC (2002b) discusses two types of quantitative analyses: intermediate performance analyses and pinch-point analyses. Examples of intermediate performance measures and pinch-point metrics consider the movement of water and transport of radionuclides. The approach also includes a figure depicting barrier effectiveness for a single radionuclide at two different times.

By providing Bechtel SAIC Company, LLC (2002b), DOE has satisfied the intent of agreement TSPAI.1.01 (Schlueter, 2003). The NRC staff will evaluate the implementation of this approach as it follows the DOE progress toward satisfying agreement TSPAI.1.02.

Overall, the available information, along with key technical issue agreements reached between DOE and NRC, is sufficient to expect that the information necessary to assess the description of barrier capability will be available at the time of a potential license application.

5.1.1.4.3 Technical Basis for Barrier Capability

The level of staff review of the technical basis for each barrier is informed by the waste isolation significance of each barrier's capability, as noted in the DOE description of barrier capability. Staff expect the technical bases for barrier capability to be based on and consistent with the technical bases for the performance assessment. An important aspect of the technical basis is a discussion of the uncertainty in each barrier capability that might diminish the ability of the barrier to isolate waste. Discussion of barrier uncertainties would include, as appropriate, temporal and spatial uncertainty, and uncertainties in features, events, and processes that could significantly degrade each barrier capability. Technical basis for the models and abstractions contained within the DOE performance assessment will be provided in the potential license application; however, staff expect the technical basis for barrier capability to summarize the technical basis for the performance assessment with a focus on the uncertainties in barrier capabilities.

DOE documents its approach for natural and engineered barriers in CRWMS M&O (2000a,b). The staff review of this approach to multiple barriers results in several concerns for the technical basis for barrier capability. These concerns are the same as those identified in the description of barrier capability section (i.e., DOE treatment of barriers relies mostly on discussion of dose rather than particular attributes or capabilities of the barriers). NRC presented the previously mentioned concerns to DOE, and general agreements were reached at the DOE and NRC technical exchange (Reamer, 2001). For TSPAI.1.01, DOE agreed to provide a discussion of the capabilities of individual barriers, in light of existing parameter uncertainty (e.g., in barrier and system characteristics) and model uncertainty. For TSPAI.1.02, DOE agreed to provide a discussion of the following when documenting barrier capabilities: (i) parameter uncertainty, (ii) model uncertainty (i.e., the effect of viable alternative conceptual models), and (iii) spatial and temporal variabilities in the performance of the barriers.

In response to TSPAI.1.01, DOE provided Bechtel SAIC Company, LLC (2002b). This document provides an overview of the approach DOE plans to use in its total system performance assessment license application. DOE indicates the level of information provided to describe a barrier (i.e., the technical basis) would be commensurate with the relative importance of the barrier to demonstrating compliance with the individual protection requirement of 10 CFR 63.113(b) and ground water protection requirement of 10 CFR 63.113(c). By providing Bechtel SAIC Company, LLC (2002b), DOE has satisfied the intent of agreement TSPAI.1.01 (Schlueter, 2003). Staff will evaluate the implementation of this approach as it follows the DOE progress toward satisfying agreement TSPAI.1.02.

Overall, the available information, along with key technical issue agreements between DOE and NRC, is sufficient to expect that the information necessary to assess the technical basis for barrier capability will be available at the time of a potential license application.

5.1.1.5 Summary and Status of Key Technical Issue Subissues and Agreements

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 5.1.1-1. This subissue is considered closed-pending by the NRC staff as documented following the DOE and NRC technical exchange (Reamer, 2001). The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (e.g., specified testing or analyses), indicates that information necessary to begin a technical review will likely be available at the time of a potential license application.

It should be noted the staff review to date has been limited to the methodology portion of multiple barriers, and NRC is not addressing if DOE has adequately identified multiple barriers or if DOE has demonstrated multiple barriers are present. The status and the detailed agreements pertaining to all key technical issue subissues are provided in Table 1.2-1 and Appendix A.

Table 5.1.1-1. Status of Resolution of the System Description and Demonstration of Multiple Barriers Subissue			
Key Technical Issue	**Subissue**	**Status**	**Related Agreement***
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 review methods.			

5.1.1.6 References

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5.1.2 Scenario Analysis and Event Probability

5.1.2.1 Scenario Analysis and Event Probability

5.1.2.1.1 Description of Issue

Performance assessment is a systematic analysis that identifies features, events, and processes that might affect performance of a geologic repository, examines their effects on performance, and estimates the radiological exposures to the reasonably maximally exposed individual. Features, events, and processes considered in the performance assessment should represent a wide range of both beneficial and potentially adverse effects on performance during the regulatory period.

Scenario analysis is a systematic enumeration of features, events, and processes that can reasonably occur in the repository system, and is a starting point for the performance assessment. Scenario analysis facilitates the identification of possible ways in which a geologic repository environment can evolve so a defensible representation of the system can be implemented 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. It 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 potential 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 DOE scenario analysis methodology and implementation. Technical bases for the scenario analysis are documented in analysis and model reports, CRWMS M&O (2000a), Bechtel SAIC, LLC (2002a,b), and other technical reports (associated with the key technical issue subissues). 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 other referring to the identification of events with probabilities greater than 10^{-8} per year.

5.1.2.1.2 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.

5.1.2.1.3 Importance to Postclosure Performance

Scenario analysis identifies features, events, and processes that could influence, directly or indirectly, dose risk from the potential 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 potential 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 aspects analyzed in the evaluation of the repository safety and serves as a road map to the location of analyses and their conclusions. Therefore, the goal of scenario analysis is to ensure that no important aspect of the potential high-level waste repository is overlooked in the evaluation of its safety.

5.1.2.1.4 Technical Basis

NRC developed a review plan (NRC, 2003) based in part on acceptance criteria and review methods developed in previous issue resolution status reports. A review of the DOE approaches for development of a scenario analysis to support the total system performance assessment is provided in the following subsections. The assessment is organized according to the four review methods in NRC (2003): (i) Identification of an Initial List of Features, Events, and Processes; (ii) Screening of the Initial List of Features, Events, and Processes; (iii) Formation of Scenario Classes Using the Reduced Set of Events; and (iv) Screening of Scenario Classes.

5.1.2.1.4.1 Identification of an Initial List of Features, Events, and Processes

Scenario Analysis identifies the features, events, and processes that could influence, directly or indirectly, dose risk from the potential high-level waste repository to a reasonably maximally exposed individual and is an integral part of the performance assessment. Therefore, staff will evaluate whether the initial list of features, events, and processes is complete enough that no aspect with potential to have more than a minimal effect on repository performance is overlooked.

The process used to construct the initial list of features, events, and processes is detailed in CRWMS M&O (2000a, 2001a) and Bechtel SAIC, LLC (2002b). DOE compiled a database of features, events, and processes potentially relevant to the potential 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 Cooperation 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 2001a). 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). The Enhanced Plan for Features, Events, and Processes (FEPs) at Yucca Mountain (Bechtel SAIC, LLC, 2002b) discusses proposed improvements to the DOE scenario analysis to enhance transparency in the identification, screening, and documentation of features, events, and processes.

A total of 1,808 entries, identified as primary, secondary, or classification, has been cataloged in the CRWMS M&O (2001b). Primary entries have been given broad definitions so they encompass multiple secondary entries. Screening arguments were developed mainly for primary features, events, and processes. A total of 328 primary features, events, and processes has been identified in the database (CRWMS M&O, 2001a). According to Bechtel SAIC, LLC (2002b), the number of entries may change as a result of redefining the scope of features, events, and processes, and minimizing overlap among definitions. Later revisions to the database will eliminate reference to secondary entries (Bechtel SAIC, LLC, 2002b).

DOE states 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). Bechtel SAIC, LLC (2002b) outlines a process for the tracking and consideration of new information that could result in the identification of new features, events, and processes and potential impacts to existing features, events, and processes.

The 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 potential high-level waste repository are not described in this list. However, these aspects not explicitly mentioned in the initial list of features, events, and processes (e.g., response of the drip shield to static loads and seismic excitation) are covered by existing key technical issue agreements [e.g., Subissue 1 of Container Life and Source Term Key Technical Issue Agreement 14 (Schlueter, 2000)]. NRC staff has not identified any relevant aspect that is not already considered in the initial set of features, events, and processes, or in existing key technical issue agreements. Recommendations in Bechtel SAIC, LLC (2002b) are intended to enhance the navigation structure of the list of features, events, and processes to facilitate identification of the technical aspects considered in the DOE analyses and eliminate the apparent lack of completeness in the initial list. Implementation of the enhanced plan is also intended to better define the scope of broad features, events, and processes (e.g., Section 2.3.13.01.00—Biosphere Characteristics). Broad-scope features, events, and processes overlap and frequently have associated dual screening decisions (i.e., particular aspects of the feature, event, and processes are included in the performance assessment while others are disregarded), clouding the identification of the aspects addressed by the performance assessment model.

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 DOE and NRC Technical Exchanges and Management Meetings on Total System Performance Assessment and Integration (Reamer, 2001a,b). At the May 15–17 meeting (Reamer 2001a), 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. Additionally, 10 CFR Part 63 requires that DOE "... provide the technical basis for either inclusion or exclusion of specific features, events, and processes... ." Because of the varying levels of information used to define the scope of primary features, events, and processes, it is difficult to judge the comprehensiveness of the existing database (Reamer, 2001a). Also, the current structure of the database did not permit clearly identifying where and how particular features, events, and processes were addressed in the performance assessment model.

At the August 6–10 meeting (Reamer 2001b), 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 (Reamer, 2001b).

Various agreements addressing the issues highlighted in Section 5.1.2.1.4.1 were reached at the May 15–17 and August 6–10, 2001 (Reamer, 2001a,b), DOE and NRC Technical Exchanges and Management Meetings on Total System Performance Assessment and Integration, and are listed in Section 5.1.2.1.5.

DOE submitted an Enhanced Plan for Features, Events, and Processes (FEPs) at Yucca Mountain (Bechtel SAIC, LLC, 2002b), in response to two agreements reached at DOE and NRC Technical Exchanges and Management Meetings. DOE reiterated elements of the Bechtel SAIC, LLC (2002b) related to the identification and classification of features, events, and processes in the Total System Performance Assessment–License Application Methods and Approach (Bechtel SAIC, LLC, 2002a). NRC requested additional details (Schlueter, 2002) on the Enhanced Plan for Features, Events, and Processes to fulfill Agreements TSPAI.2.05 and TSPAI.2.06. Information requested by NRC included clarification on the comprehensiveness of the DOE approach to the identification of features, events, and processes; screening and documentation of features, events, and processes considered for inclusion in the performance assessment; and the impact of new information on existing features, events, and processes. DOE submitted KTI Letter Response to Additional Information Needs on TSPAI.2.05 and TSPAI.2.06 (Bechtel SAIC, LLC, 2003). The NRC staff determined that this document adequately addressed the additional information needs for total system performance assessment and integration key technical issue Agreements 2.05 and 2.06 (Schlueter, 2004). The Enhanced Plan for Features, Events, and Processes (FEPs) at Yucca Mountain (Bechtel SAIC, LLC, 2002b) and the response to additional information needs (Bechtel SAIC, LLC, 2003) describe an adequate plan for the improvement of the existing list of features, events, and processes. The plan proposes an additional navigation structure, the use of keywords, and revisions to broad-scope features, events, and processes that should result in a more efficient identification of aspects covered by the features, events, and processes, as well as a more transparent identification of how particular features, events, and processes are addressed in the total system performance assessment.

Overall, the current information, along with agreements reached between DOE and NRC (Section 5.1.2.1.5), is sufficient to expect 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.

5.1.2.1.4.2 Screening of the Initial List of Features, Events, and Processes

After identification of features, events, and processes, the second step in the scenario analysis is the development of screening arguments for further consideration of features, events, and processes into the total system performance abstraction. Those features, events, and processes with the potential to affect dose risk should be included in the performance assessment, and those that are unlikely (less than one chance in 10,000 over 10,000 years) or noninfluential to dose risk can be excluded from further analysis. Therefore, staff will evaluate whether screening rationales are robust enough so that no feature, event, or process influential to repository performance is excluded from consideration in the performance assessment model.

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 screening arguments for features, events, and processes, which are listed in Table 5.1.2.1-1. Database entries were assigned to more than one analysis and model report because, in general, the 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 potential 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 indicated that they are planning to do so (CRWMS M&O, 2000a; Bechtel SAIC, LLC, 2002a).

Each primary database entry was screened as included or excluded on the basis of three criteria developed in the DOE Interim Guidance (Dyer, 1999). 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 (Dyer, 1999) 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).

Table 5.1.2.1-1. Set of Features, Events, and Processes Analysis and Model Reports for Developing Screening Arguments			
Analysis and Model Report Title	**Control Identification**	**Revision/IN**	**Year**
Features, Events, and Processes in Unsaturated Zone Flow and Transport	AN–NBS–MD–000001	01/00	2001
Features, Events, and Processes in Saturated Zone Flow and Transport	AN–NBS–MD–000002	01/00	2000
Evaluation of the Applicability of Biosphere-Related Features, Events, and Processes	AN–MGR–MD–000011	01/00	2001
Features, Events, and Processes: Screening for Disruptive Events	AN–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

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 (Dyer, 1999; CRWMS M&O, 2001a). As mentioned in Section 5.1.2.1.4.1, 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 (Reamer, 2001a,b), 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 5.1.2.1-1. Screening arguments in some analysis and model reports depend on assumptions yet to be verified (CRWMS M&O, 2000c, 2001c,d). 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 (2001e)}. 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 (Reamer, 2001a).

A summary of the detailed evaluation of the screening arguments is contained in Table 5.1.2.1-2, which lists the 328 primary features, events, and processes of CRWMS M&O (2001a), in ascending order of database tracking numbers. In Table 5.1.2.1-2, 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 (Reamer, 2001a), August 6–10 (Reamer, 2001b), and September 5 (Reamer, 2001c), DOE and NRC Technical Exchanges and Management Meetings, and agreements are available. The column labeled Technical Exchange in Table 5.1.2.1-2 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 5.1.2.1-2 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 5.1.2.1-2 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 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 (Reamer, 2001a) and August 6–10 (Reamer, 2001b) DOE and NRC Technical Exchanges and Management Meetings on Total System Performance Assessment and Integration, and at the September 5 (Reamer, 2001c) 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.

Table 5.1.2.1-2. 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 5.1.2.1-2. 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	–	–
1.5.03.01.00	Changes in the Earth's magnetic field	–	–	–	–	–	–	–	–	–	–	–	–	–	–	E/S	–

Table 5.1.2.1-2. 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
1.5.03.02.00	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 wasteforms	–	–	–	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 Yucca Mountain Project 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	–	–	–	–	–	–	–	–	–	–	–	–
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

Table 5.1.2.1-2. 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
2.1.03.03.00	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	*Ground water 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
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

Table 5.1.2.1-2. 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
2.1.08.05.00	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 ground water 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 wasteform 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 wasteform	–	–	–	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	–	–	–	–	–	–	–	–	–	–	–	–
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	–	–	–	–	–	–	–	–	–	–	–	–

Table 5.1.2.1-2. 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
2.1.09.18.00	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	–	–	–	–	–	–	–	–	–	–	–	–	–
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	–	–	–	–	–	–	–	–	–	–	–	–

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
Table 5.1.2.1-2. Summary of Features, Events, and Processes Screening Argument Evaluation (continued)																	
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 wasteform, top breach	–	–	–	E/A	–	–	–	–	–	–	–	–	–	–	–	74
2.1.14.04.00	Criticality *in situ*, waste package internal structures degrade at same rate as wasteform, top breach	–	–	–	E/A	–	–	–	–	–	–	–	–	–	–	–	74
2.1.14.05.00	Criticality *in situ*, waste package internal structures degrade slower than wasteform, top breach	–	–	–	E/A	–	–	–	–	–	–	–	–	–	–	–	74
2.1.14.06.00	Criticality *in situ*, wasteform 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, wasteform 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	–	–	–	–	–	–	–	–	–	–	–	–
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	–	–	–	–	–	–	–	–

Table 5.1.2.1-2. 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
2.2.06.03.00	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 ground water 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 ground water 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 ground water 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 ground water	–	–	–	–	–	–	–	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	Ground water 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
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

Table 5.1.2.1-2. 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
2.2.08.14.00	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 ground water 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
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

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
2.3.02.03.00	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	Ground water 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	–	–	–
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	–	–

Table 5.1.2.1-2. Summary of Features, Events, and Processes Screening Argument Evaluation (continued)

Table 5.1.2.1-2. 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
3.3.06.00.00	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 Forms
ENG4	Radionuclide Release Rates and Solubility Limits
UZ1	Climate and Infiltration
UZ2	Flow Paths in the Unsaturated Zone
UZ3	Radionuclide Transport in the Unsaturated 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

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 5.1.2.1.5.

Overall, the current information, along with agreements reached between DOE and NRC (Section 5.1.2.1.5), is sufficient to expect 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.

5.1.2.1.4.3 Formation of Scenario Classes Using the Reduced Set of Events

Those features, events, and processes or sequences of events or processes, screened for inclusion into the total system performance assessment model are further grouped into scenario or event classes. The staff will evaluate whether all relevant scenario classes have been identified.

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 system performance assessment (Swift, 2000; 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 appears reasonable. 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.

Overall, the current information, along with agreements reached between DOE and NRC (Section 5.1.2.1.5), is sufficient to expect 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.

5.1.2.1.4.4 Screening of Scenario Classes

After identification of scenario classes, probability or consequence arguments are developed to support consideration or disregard of the scenario classes into the total system performance assessment model. Therefore, staff will evaluate whether all relevant scenario classes have been incorporated into the total system performance assessment model.

DOE indicated that both the disruptive and nominal scenario classes are represented in the total system performance assessment (Swift, 2000; CRWMS M&O, 2000a,b). Thus, none of the scenario classes identified so far will be screened out from the performance assessment.

Overall, the current information, along with agreements reached between DOE and NRC (Section 5.1.2.1.5), is sufficient to expect 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.

5.1.2.1.5 Summary and Status of Key Technical Issue Subissues and Agreements

Table 5.1.2.1-3 provides related DOE and NRC agreements pertaining to the Scenario Analysis, as well as the status of the associated key technical issue subissues. 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 (Reamer, 2001a) and August 6–10 (Reamer, 2001b) DOE and NRC Technical Exchanges and Management Meetings, are presented in Appendix B.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (e.g., specified testing or analyses), indicates that information necessary to begin a technical review will likely be available at the time of a potential license application.

Table 5.1.2.1-3. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreement
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

Table 5.1.2.1-3. Related Key Technical Issue Subissues and Agreements (continued)			
Key Technical Issue	**Subissue**	**Status**	**Related Agreement**
Evolution of the Near-Field Environment	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 Igneous Activity	Closed-Pending	IA.1.01 IA.1.02
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

Table 5.1.2.1-3. Related Key Technical Issue Subissues and Agreements (continued)			
Key Technical Issue	**Subissue**	**Status**	**Related Agreement**
Total System Performance Assessment and Integration	Subissue 2—Scenario Analysis and Event Probability	Closed-Pending	TSPAI.2.01 through TSPAI.2.07
	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

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5.1.2.2 Identification of Events with Probabilities Greater Than 10^{-8} Per Year

5.1.2.2.1 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 in 10 CFR 60.113. (See requirements for performance assessment in 10 CFR 60.114.) The identification of events with probabilities greater than 10^{-8} per year include the following aspects: (i) appropriate definition of events and event sequences, (ii) appropriate determination of the annual probability of each event with sufficient technical bases, (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 potential repository at Yucca Mountain 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 previously were documented in CRWMS M&O (2000a,b). A summary of the current DOE approach is contained in technical basis documents for volcanic activity (Bechtel SAIC Company, LLC, 2003a) and criticality (DOE, 2003). DOE has not defined the current approach for seismicity, which is scheduled to be provided in the Technical Basis Document No. 14, Low Probability Seismic Events. Staff also reviewed portions of additional analysis and model reports, and other publicly available literature, to assess the current DOE approach for identification of events with probabilities greater than 10^{-8} per year.

5.1.2.2.2 Relationship to Key Technical Issue Subissues

Event classes identified as potentially significant for the potential repository system at Yucca Mountain include

- Igneous Activity
- Faulting
- Seismicity
- Nuclear Criticality

As specified in 10 CFR Part 63, the disruption of the repository by 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 previously has been captured within the framework of the eight 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-Hydrological-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 previous versions of the issue resolution status reports and also were the bases for technical exchanges with DOE where agreements were reached on 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.

5.1.2.2.3 Importance to Postclosure Performance

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 specifies, 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.

The probabilities of igneous disruption, faulting, seismicity, and criticality are important to postclosure performance calculations because analyses used to demonstrate compliance with licensing requirements must factor the likelihood of a potential disruptive event into the performance calculations, to determine a probability-weighted dose (i.e., risk). In addition, disruptive events with likelihoods of occurrence less than 1 in 10,000 during the 10,000-year postclosure performance period (equivalent to 10^{-8} per year for events with time-independent probabilities of occurrence) do not need to be included in the total system performance calculations.

The DOE model results (Bechtel SAIC Company, LLC, 2002; CRWMS M&O, 2000a) indicate igneous activity is one of the natural processes that could cause a significant number of waste package failures and thus result in a possible radiological dose to the receptor during the regulatory period of interest. Most DOE estimates for the probability of igneous disruption at the repository site range from on order of 10^{-10} to 10^{-8} per year (e.g., Bechtel SAIC Company, LLC, 2003a). In contrast, alternative annual probability estimates generally range from on the order of 10^{-8} to 10^{-7} per year (e.g., NRC, 1999a; Hill and Connor, 2000), to values as high as 10^{-6} per year using Bayesian methods (Ho, 1995; Ho and Smith, 1997). NRC sensitivity analyses (Appendix D of this report; Mohanty, et al., 2004) indicate the probability of igneous activity is a significant contributor to total system performance assessment results.

None of these probability models, however, has considered current uncertainties in the number and age of past igneous events (Hill and Stamatakos, 2002). Using a range of alternative conceptual models, Hill and Stamatakos (2002) described how these uncertainties may have negligible to order-of-magnitude effects on the igneous activity probability estimate. Because the probability of igneous activity is directly proportional to the risk from potential igneous activity, these unaccounted for uncertainties may result in negligible to order-of-magnitude effects on current risk estimates.

CRWMS M&O (2000a) identifies the probability of igneous intrusion as one of the eight principal factors for the Yucca Mountain potential repository system. With respect to other low-frequency events, the occurrence of seismic activity or faulting could result in failure of the waste package or drip shield. Earthquake-induced ground vibrations could lead to premature drift collapse or even direct damage to waste packages and drip shields, if these engineered systems were to collide with each other during a strong earthquake. Similarly, faulting could lead to drift degradation or, with substantial fault displacement across a drift, potential direct rupture of the waste packages or drip shields. Performance of the waste package and performance of the drip shield and drift invert system are also identified as principal factors for the potential Yucca Mountain repository system (CRWMS M&O, 2000a). Criticality events could generate additional radioactive inventory in the spent nuclear fuel or alter the rate of spent nuclear fuel dissolution, and this could affect dose estimates.

5.1.2.2.4 Technical Basis

NRC developed a review plan (NRC, 2003) consistent with the acceptance criteria and review methods found in the previous issue resolution status reports. A status assessment of the DOE approaches for including identification of events with probabilities greater than 10^{-8} per year is provided in the following subsections. This assessment is divided into four subsections: Igneous Activity, Faulting, Seismicity, and Nuclear Criticality. The assessment is organized according to the five review methods: (i) Event Definition, (ii) Probability Estimates, (iii) Probability Model Support, (iv) Probability Model Parameters, and (v) Uncertainty in Event Probability.

5.1.2.2.4.1 Igneous Activity

For the past 11 million years, basaltic volcanoes have formed in scattered locations throughout the area around the potential Yucca Mountain repository site. Many studies have been conducted on interpreting patterns of this past volcanic activity to calculate the likelihood of a new volcano forming at the potential repository site during the next 10,000 years (e.g., NRC, 1999a; Bechtel SAIC Company, LLC, 2003a). The DOE approach to evaluating the probability of igneous disruption is based on an expert elicitation conducted in 1995 (CRWMS M&O, 1996), which used the judgment of 10 subject-matter experts to interpret available information and develop numerical probability models. Most of the probability models developed during this elicitation used spatio-temporal patterns of past volcanic activity to calculate the likelihood of a subsurface igneous event intersecting the potential Yucca Mountain repository site. These conceptual models were based primarily on the location and age of basaltic volcanoes identified up to 1995. An important assumption during the development of these models was that the ages of all igneous events were reasonably well known, and there was a limited potential for buried but undetected events (CRWMS M&O, 1996).

Because there is no generally accepted methodology to evaluate the probability of future igneous events, a variety of different conceptual models have been developed by NRC (e.g., Connor and Hill, 1995; Connor, et al., 2000) and others (e.g., Ho, 1995; Ho and Smith, 1998). The technical bases of these probability models are reviewed in NRC (1999a). Based on insights gained from alternative probability models and interpretations of igneous processes in the Yucca Mountain region, the staff previously documented technical issues with respect to the DOE approach to evaluating the probability of future igneous activity (e.g., NRC, 1999a).

To resolve these technical concerns, NRC reached two agreements with DOE (Schlueter, 2000) that were sufficient to elevate the status of the probability subissue to closed-pending. DOE agreed to include, in any possible site recommendation and potential license application, for information purposes, the results of a single-point sensitivity analysis for extrusive and intrusive igneous activity at a probability of 10^{-7} per year. Use of this single-point value will provide staff with the information necessary to review the effects of the DOE probability distribution, and of alternative conceptual models, on the risk estimate. In addition, an aeromagnetic survey was conducted over the Yucca Mountain region (Blakely, et al., 2000). Interpretations of the aeromagnetic data showed that, in addition to the 7 buried volcanoes identified in 1995 (CRWMS M&O, 1996), approximately 13 additional volcanoes may be buried beneath the alluvium in this region and that additional volcanoes could remain buried but undetected (O'Leary, et al., 2002; Hill and Stamatakos, 2002). DOE also agreed to examine the results of this survey for the presence of previously unrecognized buried igneous features and to evaluate the effects of these possible igneous events on the CRWMS M&O (1996) probability estimate.

5.1.2.2.4.1.1 Event Definition

DOE documents the approach and technical basis for the definition of an igneous event in CRWMS M&O (2000b), which is summarized in Bechtel SAIC Company, LLC (2003b). The DOE estimate of the probability of an igneous event affecting the repository is based on the results of an expert elicitation conducted in 1995 to determine the probability of future igneous activity at Yucca Mountain (CRWMS M&O, 1996). DOE generally defines a volcanic event as a point in space representing a volcano and an associated intrusive dike having length, azimuth, and location extending from the volcano (CRWMS M&O, 2000c). Other igneous event definitions are possible, including, for example, the formation of volcano alignments as single events (e.g., NRC, 1999a). Each expert in the 1995 DOE elicitation, however, used different combinations of event characteristics to define igneous events (CRWMS M&O, 1996). Although the 1995 DOE elicitation assumed volcanic events have both an extrusive (i.e., eruptive volcano) and an intrusive component (i.e., dike), the output of this elicitation was the annual frequency of intersection of the repository only by an intrusive basaltic dike.

To derive the probability of a volcanic igneous event occurring within the repository, DOE subsequently developed a model for the distribution of volcanoes along a dike based on information in CRWMS M&O (1996) and some observed vent spacings in the Yucca Mountain region (CRWMS M&O, 2000c). In the model, volcanoes were allowed either to occur randomly along the length of a dike, or to preferentially localize near a repository drift (CRWMS M&O,

2000c). Using this approach, DOE concluded that an average of 77 percent of the repository-intersecting intrusive events would result in at least one volcano occurring within the repository footprint. This approach makes a clear and consistent distinction between intrusive and extrusive igneous events in the probability calculations.

Anomalies interpreted from aeromagnetic survey data may represent additional buried volcanoes that have not been considered in probability models (O'Leary, et al., 2002; Hill and Stamatakos, 2002). Approximately half of the recognized aeromagnetic anomalies form alignments or clusters, which are similar to some interpretations of igneous events in CRWMS M&O (1996). In an attempt to evaluate the effect of this new information on the DOE probability estimate, a sensitivity analysis in Ziegler (2002) used the judgment of DOE project staff to interpret how the elicitation experts likely would define the newly recognized aeromagnetic anomalies as igneous events. These new event counts were then propagated into the recurrence rate distributions used in CRWMS M&O (1996). Concerns regarding the appropriate and consistent definition of igneous events in the DOE probability estimate were discussed in Schlueter (2002a). Additional information to address these concerns was provided in Ziegler (2003), currently under evaluation by staff.

While some information on the identification of igneous activity events with probabilities greater than 10^{-8} per year, with respect to event definition, will be available at the time of a potential license application, the staff is currently of the view that DOE should provide additional information on the DOE consistency of definitions of igneous events used in the expert elicitation and the DOE subsequent analyses.

5.1.2.2.4.1.2 Probability Estimates

DOE documented the approach and technical basis for the definition of an igneous event in CRWMS M&O (2000b), which is summarized in Bechtel SAIC Company, LLC (2003a). The DOE estimate of the probability of an igneous event affecting the repository is based on the results of an expert elicitation conducted in 1995 to determine the probability of future igneous intrusive activity at Yucca Mountain (CRWMS M&O, 1996). Using various interpretations of past activity in the Yucca Mountain region, the DOE experts generally assumed volcanic events to have both an extrusive and intrusive component. The output of the DOE elicitation, however, was the annual frequency of intersection of the repository by only an intrusive event. To derive the probability of an extrusive event occurring within the repository, DOE developed a model for the distribution of volcanoes along an intrusion based on information in CRWMS M&O (1996) and some observed volcano spacings in the Yucca Mountain region (CRWMS M&O, 2000c). Volcanoes were allowed either to occur randomly along the length of an intrusion, or to localize preferentially near a repository drift (CRWMS M&O, 2000c). Using this approach, DOE estimates the mean extrusive and intrusive disruption probabilities are slightly greater than 10^{-8} per year (Bechtel SAIC Company, LLC, 2003a).

Previous DOE probability estimates for future igneous activity at the potential repository site were based on interpretations of past patterns of igneous activity in the Yucca Mountain region. Although basaltic igneous features preserved at the surface on or around Yucca Mountain appear well characterized, aeromagnetic surveys conducted after the DOE probability elicitation (Blakely, et al., 2000) indicate approximately 13 additional volcanoes may be buried in this area (O'Leary, et al., 2002; Hill and Stamatakos, 2002). Other igneous features may remain buried but undetected due to limited resolution capabilities of aeromagnetic surveys in this type of terrain. These potential volcanoes represent a larger uncertainty than that considered during the 1995 DOE elicitation with regard to spatial and temporal patterns of past igneous activity in the Yucca Mountain region.

Current uncertainties in the number and age of potential buried volcanoes may affect fundamental assumptions made during the 1995 DOE probability elicitation regarding temporal patterns of past activity. Within the limits of the 1995 information, some experts evaluated the hypothesis for nonhomogeneous temporal recurrence processes and concluded the available information did not support adoption of this hypothesis in their probability models (CRWMS M&O, 1996). The primary reason nonhomogeneous temporal recurrence rate processes were not adopted was because the number and age of past events were thought to be relatively well characterized and temporal patterns were not apparent. Logically, a large increase in uncertainty for the frequency of past events reasonably could affect an independent expert consideration of alternative models for temporal recurrence rate, including consideration of temporally nonhomogeneous processes to account for that uncertainty (Schlueter, 2002a). DOE considers that, because temporally nonhomogeneous models were not adopted in 1995, such models should not be considered in current sensitivity analyses regardless of current uncertainties in recurrence rate (Ziegler, 2003, 2002).

Similarly, current uncertainties in the number and age of potential buried volcanoes may affect fundamental assumptions made during the 1995 DOE probability elicitation regarding spatial patterns of past activity. Patterns of past events were used in CRWMS M&O (1996) to develop conceptual models for homogeneous and nonhomogeneous spatial recurrence rates. Current uncertainties on the number and age of past events reasonably could affect an independent expert conceptual model for spatial recurrence rates and would necessarily affect existing model parameters involving spatial density functions (Schlueter, 2002a). The DOE sensitivity analyses, however, (Ziegler, 2002) only consider different ranges and values for some model parameters, rather than effects on source-zone definitions, spatial density functions, or alternative source-zone models from the 1995 DOE elicitation.

DOE concluded that the effects of recently recognized potential buried volcanoes on spatial and temporal recurrence models were not significant to the DOE probability estimate (Ziegler, 2002). NRC has indicated that DOE should include a full evaluation of the effect of current model and data uncertainties on its probability estimate (Schlueter, 2002a). Staff currently are evaluating additional information provided in Ziegler (2003) regarding the significance of new uncertainties in past patterns of igneous activity.

Most of the DOE probability models in CRWMS M&O (1996) only considered basalt younger than 5 million years relevant to deriving patterns of igneous activity in the Yucca Mountain region. Current uncertainties on the number and age of buried igneous events greatly exceed the event uncertainties considered during the 1995 DOE elicitation. Multiple interpretations of current uncertainties are possible, which could, for example, cause an independent expert to consider patterns of basaltic events older than 5 million years relevant to understanding appropriate patterns of activity for use in probability models (Schlueter, 2002a). Ongoing work at CNWRA also suggests basalt in the Crater Flat Basin younger than 11 million years may have a common petrogenesis, whereas 7–11 million-year-old basalt formed outside the Crater Flat Basin may have a different petrogenesis that was strongly influenced by silicic caldera-forming processes. This new information indicates that Miocene basalt in the Crater Flat basin may provide relevant information for risk assessments, which was not considered in the 1995 DOE elicitation. DOE has concluded that current information would not affect the 1995 conclusions regarding the relevancy of events only younger than 5 million years (Ziegler, 2002). Additional information in Ziegler (2003) was provided to address a concern regarding the significance of new uncertainties in patterns of igneous activity on the DOE probability estimate expressed in Schlueter (2002a). Staff is currently evaluating this additional information.

The cumulative effect of these technical concerns leads to reasonable uncertainty in the estimate of the probability of igneous activity affecting the repository system (e.g., NRC, 1999a). To provide NRC staff with a simplified basis to evaluate the significance of these concerns, along with associated uncertainties and alternative probability models, DOE agreed to include, in the total system performance assessment–site recommendation and any potential 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 their review.

While some information provided on the identification of igneous activity events with probabilities greater than 10^{-8} per year, with respect to support for an appropriate technical basis for the probability estimates, will be available at the time of a potential license application, the staff is currently of the view that DOE should provide additional information on the DOE determination of the effect on the estimates of uncertainties in the number and age of potential buried igneous bodies.

5.1.2.2.4.1.3 Probability Model Support

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 effect on how probability models are formulated and the consequent results of probability models. This model is developed by DOE in CRWMS M&O (2000c, 1996) and Bechtel SAIC Company, LLC (2003a). During the 1995 probability elicitation, the DOE experts distinguished between deep (i.e., mantle source) and shallow (i.e., upper crustal structure and stress field) processes when considering regional and local scales of spatial control on volcanism. Many probability models in CRWMS M&O (1996), however, restricted the areas of above-background likelihood for future volcanic activity to the areas where previous volcanism has occurred. Although the basis for most of these source-zone models was expert judgment and not tectonic models, currently available geophysical data (gravity, aeromagnetic, and seismic) do not support many of the zone definitions used in the DOE probabilistic volcanic hazard assessment (CRWMS M&O, 2000c, 1996). After the 1995 elicitation, DOE justified these source-zone definitions by relating the zones to areas within the Crater Flat Basin that have undergone the greatest amount of shallow crustal extension (e.g., Fridrich, et al., 1999; CRWMS M&O, 2000c; Ziegler, 2002). Available data, however, indicate most of the shallow crustal extension occurred before the 11 million years and younger basalt formed within the Crater Flat Basin, which calls into question the role of prior crustal extension in necessarily restricting the future location of rising magma (Stamatakos, et al., 2000).

DOE has presented the probabilistic volcanic hazard assessment source-zone modeling approach without explicit validation. In currently available documentation (Bechtel SAIC Company, LLC, 2003a; Ziegler, 2003), DOE relies on the conclusions of the 1995 expert elicitation as its technical basis in support of the source-zone model, which uses source-zone definitions derived from expert judgments of patterns of sparse events. Since the elicitation, new aeromagnetic surveys indicate possibly 13 unexposed igneous bodies in the immediate area of Yucca Mountain (O'Leary, et al., 2002; Hill and Stamatakos, 2002). This number is large compared to the total number of igneous events considered by the elicitation experts (typically 5–15 events) and the relatively small uncertainty assigned for unexposed bodies (median of 10–20 percent additional "hidden events") in the elicitation. While it continues to use the conceptual volcanic source-zone model from the 1995 elicitation, DOE has not yet provided a technical basis as to why this model and its specific source-zone definitions remain appropriate in light of new information that may significantly change the spatial and temporal distribution of igneous bodies. DOE has proposed (Ziegler, 2003), and begun, a program of geophysical surveys, drilling, and laboratory analyses to constrain existing uncertainties in the number and age of potential buried igneous bodies in the region. Data and interpretations developed in this proposed program could contribute to the technical basis for the conceptual model of regional volcanism.

DOE states that there are no alternative conceptual models developed since the 1995 elicitation that either are considered plausible or would have a significant effect on the DOE probability estimate (Bechtel SAIC Company, LLC, 2003a). This raises two points. First, DOE dismisses or disregards models published in the peer-reviewed literature after 1995 (Ho, 1995; Ho and Smith, 1998, 1997). As part of review methods used by staff to evaluate the potential license application, staff will consider alternative conceptual models. In addition, volcanic source-zone models published in Ho (1995) and Ho and Smith (1998, 1997) are derived using the same basic methods of expert interpretation and judgment as the volcanic source-zones used in the DOE probability elicitation (CRWMS M&O, 1996). Published alternative probability models that use the same conceptual basis as DOE probability models (i.e., expert judgment to define volcanic source-zones) should be appropriately factored into the DOE probability estimate.

Additionally, there is an inconsistency between probability models from the 1995 DOE elicitation and current DOE probability models (Bechtel SAIC Company, LLC, 2003a). Volcanic source-zones in CRWMS M&O (1996) were clearly defined on interpretations of the timing and location of past extrusive volcanic events in a specific area. A new igneous event center (i.e., volcano; CRWMS M&O, 1996) can form only within a defined volcanic source-zone, whereas only a subsurface intrusion could potentially extend out of the source-zone and possibly intersect the repository. The models in CRWMS M&O (2000c) and Bechtel SAIC Company, LLC (2003b), however, permit new volcanoes to form outside of the predefined volcanic source-zone. Using the conceptual basis defined in CRWMS M&O (1996), new volcanoes should occur only within the volcanic source-zone at recurrences defined by past patterns of volcanic activity within that zone. None of the experts in CRWMS M&O (1996) discussed the possibility of a new volcano forming outside the volcanic source-zone, but originating within the predefined volcanic source-zone. Thus, the current DOE probability models appear to contradict their original conceptual basis by permitting new volcanic events to occur well outside the boundaries of their predefined volcanic source-zones.

The cumulative effect of these concerns regarding probability model support leads to reasonable uncertainty in the estimate of the probability of igneous activity affecting the repository system (e.g., NRC, 1999a). To provide NRC staff with a simplified basis to evaluate the significance of these concerns, along with associated uncertainties and alternative probability models, DOE agreed 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.

While some information provided on the identification of igneous activity events with probabilities greater than 10^{-8} per year, with respect to probability model support, will be available at the time of a potential license application, the staff is currently of the view that DOE should provide additional information on the DOE determination of the effect on the model of uncertainties in the number and age of potential buried igneous bodies.

5.1.2.2.4.1.4 Probability Model Parameters

DOE documented the approach and technical basis for defining probability model parameters in CRWMS M&O (2000b,c), which also is summarized in Bechtel SAIC Company, LLC (2003a). These parameters are based primarily on the results of an expert elicitation conducted in 1995, which determined the probability of future igneous intrusive activity at Yucca Mountain (CRWMS M&O, 1996). Other parameters are derived from interpretations of igneous features in the Yucca Mountain region (CRWMS M&O, 2000c). In general, parameters in the DOE probability models are constrained by traceable interpretations or data from past basaltic igneous events in the Yucca Mountain region.

In some cases, only a subset of the available data was selected for subsequent use, without explicit criteria or justification. For example, vent spacing (CRWMS M&O, 2000c, Section 6.5.2.2) only uses data from the 1-million- year Crater Flat and 0.3-million-year Sleeping Butte volcanoes, but ignores relevant information from the 3.7-million-year Crater Flat volcanoes, 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 in the DOE probability models (Bechtel SAIC Company, LLC, 2003a) that a relationship exists between the number of events and the number of dikes, which were considered independent parameters in the 1995 DOE probability elicitation (CRWMS M&O, 1996).

As discussed in Section 5.1.2.2.4.1.3 of this report, new information has increased the level of uncertainty about the number, age, and location of possible buried volcanoes in the Yucca Mountain region. DOE has not yet provided a technical basis as to why the conceptual model and its specific source-zone definitions from the 1995 elicitation remain appropriate given that the spatial and temporal distribution of past igneous events may be significantly different from that recognized in the elicitation. In addition, the ranges of important DOE model parameters, such as event length and orientation, may not account for uncertainties arising from the possible distribution of buried volcanoes. It is also not clear that the parameter ranges reflect current understanding of the age uncertainty associated with previously recognized buried events, or that they include the potential of buried but undetected events in the Yucca Mountain region. While DOE has produced several sensitivity analyses that conclude a lack of significance of post-1995 information for the DOE probability estimate (Ziegler, 2003, 2002), these analyses are limited to the source-zone model of the 1995 expert elicitation and do not capture all of the current uncertainty in conceptual models for Yucca Mountain volcanism. As previously noted, the program of work proposed in Ziegler (2003) and currently under way by DOE may help to constrain those aspects of the history of igneous activity which contribute to uncertainties in the estimate of future volcanism.

The cumulative effect of these concerns leads to reasonable uncertainty in the estimate of the probability models that could result in an inaccurate estimate of the probability of igneous activity affecting the repository system (e.g., NRC, 1999a). To provide NRC staff with a simplified basis to evaluate the significance of these concerns, along with associated uncertainties and alternative probability models, DOE agreed 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.

While some information provided on the identification of igneous activity events with probabilities greater than 10^{-8} per year, with respect to support for probability model parameters, will be available at the time of a potential license application, the staff is currently of the view that DOE should provide additional information on the DOE determination of the effect on the model of uncertainties in the number and age of potential buried igneous bodies.

5.1.2.2.4.1.5 Uncertainty in Event Probability

DOE documents the approach and technical basis for defining the probability estimate in CRWMS M&O (2000b,c), which also is summarized in Bechtel SAIC Company, LLC (2003a). This probability estimate is based primarily on the results of an expert elicitation conducted in 1995, which determined the probability of future igneous intrusive activity at Yucca Mountain (CRWMS M&O, 1996). Other parts of the volcanic probability estimate are derived from interpretations of igneous features in the Yucca Mountain region (CRWMS M&O, 2000c).

As discussed in Section 5.1.2.2.4 of this report, the effects of current uncertainties about the number, age, and location of possible buried volcanoes in the Yucca Mountain region have not been accounted for in the DOE probability estimate (Schlueter, 2002a). Event definitions have not been modified to account for current uncertainties in event counts or characteristics. The technical basis used to support DOE probability models does not encompass current uncertainties, which indicate different patterns of spatial or temporal clustering may be valid relative to the narrow range of uncertainty considered during the 1995 DOE elicitation. Model uncertainties related to new interpretations of temporal or spatial clustering have not been included in the DOE probability estimate. The DOE probability models have not undergone a formal model validation process. In addition, alternative conceptual probability models have been published in the peer-reviewed literature since the 1995 elicitation, but have been disregarded by DOE. Thus, the effects of credible alternative conceptual models have not been evaluated in the DOE probability estimate. Ranges of important DOE model parameters, such as event lengths and orientations, do not account for current uncertainties in the number, age, and location of possible buried volcanoes. In addition, these parameter ranges do not reflect current understandings of the age uncertainty associated with previously recognized buried events, or for the potential of buried but undetected events in the Yucca Mountain region. DOE has produced several sensitivity analyses that conclude a lack of significance of the effects of any new post-1995 information on the DOE probability estimate (Ziegler, 2003, 2002). The NRC staff does not agree with this conclusion (Schlueter, 2002a), and continues to evaluate information supplied by DOE to resolve these concerns.

The cumulative effect of these concerns is that the DOE probability estimate may inaccurately represent the probability of igneous activity affecting the repository system (e.g., NRC, 1999a; Schlueter, 2002a). To provide the staff with a simplified basis to evaluate the significance of these concerns, along with associated uncertainties and alternative probability models, DOE agreed 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.

While some information provided on the identification of igneous activity events with probabilities greater than 10^{-8} per year, with respect to probability model uncertainty, will be available at the time of a potential license application, the staff is currently of the view that DOE should provide additional information the DOE determination of the effect on the model of uncertainties in the number and age of potential buried igneous bodies.

5.1.2.2.4.2 Faulting

The potential effect of direct fault displacement of the engineered barrier systems is one of several disruptive scenarios currently being evaluated by DOE with respect to postclosure performance at Yucca Mountain, Nevada. To address this potential disruptive scenario, DOE assessed both the probability and consequences of faulting. Probability estimates of faulting at Yucca Mountain were developed as part of the DOE expert elicitation on seismicity and faulting (see Section 5.1.3.2, Mechanical Disruption of Engineered Barriers, for a detailed description of the elicitation results). In that elicitation, the experts derived probabilistic fault displacement hazard curves for a series of demonstration points at or near Yucca Mountain. These demonstration points were selected to represent faulting and related fault deformation in the subsurface and near the proposed surface facility sites. DOE is currently using the results of that expert elicitation to evaluate the potential consequences of faulting on repository performance. At present, DOE considers faulting within the repository to be too infrequent and fault displacements too small to impact repository performance, and as such has screened the faulting disruptive event from consideration in their total system performance assessment.

To evaluate the DOE analyses of faulting within a potential license application for Yucca Mountain, the staff has reviewed the DOE probabilistic fault displacement results and associated DOE analyses of the potential consequences of faulting. Based on this review of the DOE analyses coupled with risk insights gained from an independent consequence analysis of faulting (Stamatakos, et al., 2003) staff concluded that DOE has assembled sufficient information on the issue of direct faulting in the prelicensing period for NRC to conduct a review of a potential license application.

Overall, the available information is sufficient to expect that the information necessary to assess the probability of faulting affecting the repository system will be available at the time of a potential license application. The staff considers the faulting subissue, as defined within the Structural Deformation and Seismicity Key Technical Issue, to be closed.

5.1.2.2.4.2.1 Event Definition

The approach and technical basis for defining faulting events are contained in CRWMS M&O (2000b). DOE divides faulting events into separate features, events, and processes based on their potential consequence. DOE considers that faulting events could potentially alter ground water flow around and below the drift or could potentially disrupt engineered barriers in the repository system. When considering the effects of faulting on ground water 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.

Overall, the available information is sufficient to expect that the information necessary on faulting with respect to event definition will be available at the time of a potential license application.

5.1.2.2.4.2.2 Probability Estimates

The approach and technical basis for defining the probability of faulting affecting the potential repository system are contained in CRWMS M&O (2000b) and the analysis and model reports in CRWMS M&O (2000d–f). The basis for the estimates of the probability of faulting events affecting the potential repository system is the result of an expert elicitation documented in CRWMS M&O (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 (1999b) 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 [0.039 in] or less at the 10^{-5} annual exceedence 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} exceedence 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 exceedence probability.

DOE concluded faulting affecting ground water 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 potential 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 exceedence 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. The sufficiency of the DOE information on the Mechanical Disruption of Engineered Barriers is considered in Section 5.1.3.2. 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 ground water flow.

The staff reviewed the data, conceptual models, and assumptions developed by DOE in the probabilistic seismic hazard assessment (CRWMS M&O, 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 (CRWMS M&O, 1998), appear 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.

Overall, the available information is sufficient to expect that the information necessary on faulting with respect to probability estimates will be available at the time of a potential license application.

5.1.2.2.4.2.3 Probability Model Support

The support for the probability model is contained in CRWMS M&O (2000b) and the analysis and model reports (CRWMS M&O, 2000d,f,g). 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.

The 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. In addition, the probability estimates of faulting were derived from an expert elicitation, in which the individual experts considered a wide range of alternative faulting models.

Overall, the available information is sufficient to expect that the information necessary on faulting with respect to probability model support will be available at the time of a potential license application.

5.1.2.2.4.2.4 Probability Model Parameters

The technical basis for the parameters used in the probability model is contained in CRWMS M&O (2000a,b) and the CRWMS M&O (2000g) reports. 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.

The staff review indicates DOE has 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 has 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) and implemented in the probabilistic fault displacement hazard assessment (CRWMS M&O, 1998), appear geologically consistent and reasonable. The experts 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 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 DOE observations and interpretations.

Overall, the available information is sufficient to expect that the information necessary on faulting with respect to probability model parameters will be available at the time of a potential license application.

5.1.2.2.4.2.5 Uncertainty in Event Probability

The technical basis for the estimate of uncertainty in the probability model is contained in CRWMS M&O (2000b) and the CRWMS M&O (2000g) analysis and model report. 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 range of expert predictions for low probability (<10^{-6} per year) fault displacements.

Overall, the available information is sufficient to expect that the information necessary on faulting with respect to uncertainty in event probability will be available at the time of a potential license application.

5.1.2.2.4.3 Seismicity

The probability of seismicity at Yucca Mountain is important to both preclosure seismic safety assessment and postclosure performance calculations. Similar to many natural phenomena, earthquakes consist of a continuum of sizes and recurrence intervals that span the full range of probability, from very frequent micro-earthquakes to very large and extremely rare mega-earthquakes. Seismic events have the potential to affect performance through three effects: (i) rockfall causing direct damage to engineered barriers, (ii) failure of cladding, and (iii) changes to the ground water flow system. These effects depend on the amount of ground motion produced at the site by seismic events. As discussed in Appendix D, the drip shields and waste packages may be breached by the accumulation of damage from multiple seismic loading events. Seismic events will thereby increase the effective static load because of rockfall on the drip shield and waste packages. However, uncertainty associated with the threshold of earthquake loads needed to generate appreciable drip shield and waste package mechanical damage is large. Thus, for site characterization, DOE developed hazard estimates that encompassed the full range of earthquake probability, from once per year to 1 chance in 10,000 in 10,000 years (~10^{-8} per year).

DOE conducted a probabilistic seismic hazard assessment to assess seismic hazards at Yucca Mountain (CRWMS M&O, 1998). The probabilistic seismic hazard assessment provided probabilistic hazard curves in which increasing levels of vibratory ground motion (usually expressed in units of acceleration) are plotted as a function of progressively smaller annual exceedence probabilities. Details of the staff evaluation of the DOE expert elicitation and probabilistic seismic hazard assessment results are provided in Section 5.1.3.2, Mechanical Disruption of Engineered Barriers, of this Issue Resolution Status Report. Additional information is also presented in Section 4.1.1, Site Description As It Pertains to Preclosure Safety Analysis, and Section 7.4, Expert Elicitation.

Although the probabilistic seismic hazard assessment was completed in 1998, DOE has recently indicated that it plans to revise the ground motion expert elicitation results, especially as they pertain to postclosure performance assessments. The revision concerns the earthquake ground motions from the DOE seismic hazard study at low annual exceedence probabilities (between approximately 10^{-6} and 10^{-8}). DOE is taking this action because most technical experts (including comments from the NRC and CNWRA staff) conclude that the ground motion values at small annual exceedence probabilities are unrealistically large. For example, in the DOE postclosure performance assessment, strong motion recordings of acceleration and velocity that were scaled to the seismic hazard at 10^{-7} annual exceedence probability yield peak ground acceleration as high as 20 g [~640 ft/s^2] and peak ground velocities up to 1,800 cm/sec [~60 ft/s]. These values are beyond the limits of existing earthquake accelerations and velocities from even the largest recorded earthquakes worldwide, and they are about an order of magnitude larger than those observed for earthquakes with moment magnitudes between 6.5 and 7.0. These large ground motions also are deemed physically unrealizable because they require a combination of stress drop, strain, and rupture propagation that cannot be sustained without wholesale fracturing of the bedrock (e.g., Kana, et al., 1991). Finally, these unrealistic ground motions are difficult to incorporate into meaningful performance assessments because little is known about how the natural environment would be altered by such large ground shaking (see Section 5.1.3.2 of this report for additional discussion on how these low probability ground motions impact performance assessment of the mechanical barrier system).

The overly conservative earthquake ground motions arise in the DOE study because the seismic hazard curves are constructed as unbounded lognormal distributions. In past practice, probabilistic seismic hazard curves were used to estimate ground motions with annual exceedence probability down to 10^{-4} (a typical annual exceedence probability value designated for nuclear power plant design). Ground motions for hazards at the 10^{-4} level matched expected values for the largest earthquakes that could affect a given site. For Yucca Mountain, however, the seismic hazard curves are extrapolated to estimate ground motions with annual exceedence probabilities as low as 10^{-8}. At these low probabilities, the seismic hazard estimates are driven by the tails of the untruncated Gaussian distributions of the input ground motion attenuation models (e.g., Bommer, et al., 2004). As pointed out by Anderson and Brune (1999), overestimates of the hazards may also arise because experts improperly distributed uncertainty in the inputs between aleatory and epistemic uncertainties. The way in which the experts distributed uncertainty among their ground motion estimates, and how those uncertainties were accounted for by the composite ground motions hazard results in the DOE hazard assessment, forms the underlying technical basis of the Structural Deformation and Seismicity Agreement 2.01 (see Section 5.1.3.2 for a more complete discussion of this issue).

Similar comments and questions about the seismic hazard were raised at the 2003 Nuclear Waste Technical Review Board joint meeting on natural system and engineered systems on seismic issues (United States Nuclear Waste Technical Review Board, 2003). The joint meeting focused on the very large vibratory ground motions predicted by the DOE probabilistic seismic hazard assessment at annual exceedence probabilities below 10^{-6} per year. In a letter from the Nuclear Waste Technical Review Board to DOE (Coraddini, 2003), the Board expresses concern that

> "... although the probabilistic seismic hazard assessment is, in general, sound, extending it to very low probabilities results in ground-motion estimates about which there are serious technical questions. These relate to the lack of physical realism and the implication of these unrealistic estimates for performance assessment, design, and scientific confidence."

The Board notes that application of a physically unrealistic or highly conservative approach, even if acknowledged as such by DOE, could lead to a number of problems including a skewed understanding of repository behavior and the significance of different events; consideration of events for which there is little or no understanding or engineering practice; and undermined confidence in the scientific basis of the process under consideration.

Overall, based on previous prelicensing discussions, the available information is sufficient to expect that the information necessary to assess the probability seismic events affecting the repository system will be available at the time of a potential license application. DOE has indicated significant changes to its approach for definition of seismic events; if it does, DOE should provide information that supports those changes.

5.1.2.2.4.4 Nuclear Criticality

Based on the low probability of criticality events, DOE will develop screening arguments to screen criticality events from the performance assessment model. Alternatively, DOE may evaluate consequences of criticality events to screen them from the performance assessment model based upon limited effect to dose estimates. DOE intends to base the screening argument on the criticality analysis methodology summarized in DOE (2003). The NRC staff identified a number of concerns with the original criticality analysis methodology in DOE (1998); concerns summarized in NRC (2000d). The DOE criticality analysis methodology is described in detail in 17 model validation reports, 5 of which (Bechtel SAIC Company, LLC, 2003a,c,d, 2001a,b) were available to NRC for the writing of this report. Criticality computations, probability or consequences, will be provided with the potential license application. Documents with screening arguments on features, events, and processes related to criticality events will be published by DOE in August 2004.

The approach and technical basis for defining criticality events are contained in DOE (2003). DOE considers three major categories of criticality events: in-package events, near-field events, and far-field events. The division of criticality events based on the event location (e.g., in-package, near-field, and far-field) adequately characterizes the range of possible events.

DOE provided two estimates of the probability of criticality (CRWMS M&O, 2000h; Bechtel SAIC Company, LLC, 2001c). Several concerns with these estimates have been identified by the NRC (NRC, 2002; Schlueter, 2002b; Rom, 2003). In general, probability estimates followed an approach inconsistent with the methodology in DOE (1998) and did not use a systematic approach to identify possible configurations, their likelihood, and whether criticality events can result from the configurations.

DOE will evaluate the probability of criticality events through the use of configuration classes. DOE has proposed to calculate the probability of occurrence of configuration classes or show that configuration classes are subcritical. It is not clear that configurations classes, as defined, account for non-intact fuel conditions potentially more reactive that intact fuel (Elam, et al., 2003). DOE appears to use engineering judgment to determine the subcriticality of several configuration classes. A solid screening argument requires that the full range of potentially critical configurations, within a configuration class, are identified.

DOE has developed isotopic and criticality models for determining the reactivity of configuration classes. DOE has proposed accounting for the extent of burnup in the isotopic models, including the presence of fission products. DOE has also proposed taking exceptions to the use of consensus standards in their criticality model. The NRC staff is currently evaluating the technical basis for these deviations.

DOE has identified parameters to be used in evaluating the probability of critical configurations (DOE, 2003). However, DOE has not identified an approach to determine the most reactive configurations within a configuration class and the full range of potentially critical configurations within a configuration class. It is also important to consider effects of correlated parameters on the probability of criticality. Finally, DOE should provide parameter ranges in criticality probability that are consistent with those used in the total system performance assessment.

Alternatively to the development of probability arguments to screen criticality events from the performance assessment model, DOE may evaluate consequences of criticality events. DOE has indicated it will use the potential power and duration of steady-state events to determine the incremental change in inventory of radionuclides in the wasteform and to estimate temperature-related effects, such as potentially increased corrosion rates of the waste package or wasteform (DOE, 2003). DOE has not discussed processes that may affect the release and transport of radionuclides, such as radiolysis (except radiolytic formation of nitric acid) and evaporation. Radiolysis from alpha decay has been shown to produce changes in uranium-bearing solutions, specifically the formation of uranyl peroxides (Kubato, et al., 2003) while evaporation may concentrate impurities in the water such as carbonates.

Only partial information in DOE criticality methodology is currently available. DOE has consistently indicated that criticality calculations will be provided with the license application. Overall, available information and information DOE agreed to provide to address key technical issues agreements and criticality event analyses supporting a potential license application are sufficient to support an acceptance review.

5.1.2.2.5 Summary and Status of Key Technical Issue Subissues and Agreements

Table 5.1.2.2-1 provides the status of DOE and NRC agreements pertaining to the Identification of Events with Probability Greater Than 10^{-8} Per Year. Note the status and the detailed agreements pertaining to all the key technical issue subissues are provided in Table 1.2-1 and Appendix A. Additional agreements from the DOE and NRC technical exchange on August 6–10, 2001, are summarized in Appendix B.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (e.g., specified testing or analyses), indicates that information necessary to begin a technical review will likely be available at the time of a potential license application. As noted in this section of the report, however, further information should be provided on the effect on the estimate of probability of igneous activity of uncertainties in the number and age of potential buried igneous bodies. 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 5.1.2.2-1. Related Key Technical Issue Subissues and Agreement

Key Technical Issue	Subissue	Status	Related Agreement*
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

Table 5.1.2.2-1. Related Key Technical Issue Subissues and Agreement (continued)			
Key Technical Issue	**Subissue**	**Status**	**Related Agreement***
Structural Deformation and Seismicity	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 CLST.5.05
Evolution of the Near-Field Environment	Subissue 4—Effects of Coupled Thermal-Hydrological-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
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 review methods.			

5.1.2.2.6 References

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5.1.3 Model Abstraction

To facilitate review of prelicensing DOE total system performance assessments of a potential repository at Yucca Mountain, Nevada, the staff has defined a comprehensive set of model abstractions that integrate those features, events, and processes affecting the engineered, geosphere, and biosphere systems significant to waste isolation. These 14 model abstractions are identified as integrated subissues in NUREG–1804, Yucca Mountain Review Plan, and are used to organize the staff review of significant assumptions, models, and data that support any DOE performance assessment submitted as part of a potential license application.

Each of the following sections (5.1.3.1 through 5.1.3.14) documents the current NRC understanding of the model abstractions developed by DOE for inclusion into its total system performance assessment. For each of the 14 model abstractions, the staff assessment is focused on those aspects that are significant to waste isolation based on risk insights identified to date, which are included in Appendix D to this report. For each abstraction, the staff is solely concerned with determining whether the information gathered during the prelicensing phase to support the assumptions underlying the models used in the total system performance assessment are likely to be documented in sufficient depth to allow the staff to conduct a detailed technical review.

All 14 model abstractions follow a consistent format. The following brief summary is intended to provide an overview of the content in each section and serve as a guide to the different model abstractions.

Description of Issue

In this section, the integrated subissue is described, and its relationship to the other model abstractions is presented. Where appropriate, a physical description of the features, events, and processes that compose the model abstraction is presented.

Relationship to Key Technical Issue Subissues

In this section, the model abstraction is presented in the context of the NRC key technical issue subissues. The key technical issue subissues formed the primary topical areas for previous versions of the issue resolution status reports and were also the subject of public technical exchanges at which DOE and NRC reached formal agreements on the additional information DOE would provide to NRC to close a key technical issue subissue.

Importance to Postclosure Performance

After linking the model abstraction to the key technical issue subissues and illustrating the relationships among different model abstractions, staff describe the importance of the model abstraction to postclosure performance using information from the DOE investigations and NRC confirmatory studies. In this discussion, information from Appendix D is used to identify high-level topical areas of high and medium significance to waste isolation.

Technical Basis

The staff assessment of the technical basis for the DOE approach to each model abstraction is based on the five generic review methods from the NUREG–1804, Yucca Mountain Review Plan. As defined in Section 2.3, the five review methods address (i) Model Integration (including system description), (ii) Data and Model Justification, (iii) Data Uncertainty, (iv) Model Uncertainty, and (v) Model Support.

The staff has identified those aspects of the model abstraction significant to waste isolation in Appendix D. This section provides the necessary context for risk-informing the assessment of the current DOE approach.

The staff provides an assessment of available DOE documents that explains to what extent the model abstraction addresses the particular review method. In this section, the staff also assesses the degree to which DOE has been consistent in treating similar features, events, and processes in other model abstractions. The staff also assesses whether individual agreements related to the model abstraction have been met. At the end of this assessment, the staff indicates whether there will be sufficient information may be available at the time of submittal of a potential license application for the staff to conduct a detailed technical review of the model abstraction.

Summary and Status of Key Technical Issue Subissues and Agreements

After each of the five review methods has been applied to the model abstraction being considered staff summarize the overall status of all key technical issue subissues related to the model abstraction. Finally, the staff provides a brief concluding statement that indicates whether or not the DOE-proposed approach, together with the DOE agreements to provide NRC with additional information indicates that information necessary to begin a technical review will likely be available at the time of a potential license application.

References

This section includes references to NRC, DOE, and other technical reports that support the staff assessment of the DOE approach to each model abstraction. Only documents that were available at the end of March 2004 are considered in the assessments presented in this report.

5.1.3.1 Degradation of Engineered Barriers

5.1.3.1.1 Description of the Issue

The Degradation of Engineered Barriers Integrated Subissue addresses features and processes that affect the engineered barrier system degradation, including drip shield and waste package corrosion processes. The relationship of this integrated subissue to other integrated subissues is depicted in Figure 5.1.3.1-1. The overall organization and identification of all the integrated subissues are depicted in Figure 1.1-2. The DOE description and technical bases for abstraction of degradation of the waste package and drip shield were documented previously in CRWMS M&O (2000a) and several supporting analysis and model reports cited throughout the following sections. Also, DOE has published Technical Basis Documents (Bechtel SAIC Company, LLC, 2003a,b) and a supporting analysis and model report (Bechtel SAIC Company, LLC, 2003c) that provides the most current description of the DOE conceptual model for the evolution of the environment within the emplacement drifts, and a description of the waste package and drip shield degradation models, as well as a summary of supporting data and analyses. This section documents the current NRC understanding of the abstractions DOE developed to incorporate waste package and drip shield degradation processes into its total system performance assessment. The assessment is focused on those aspects most important to repository safety based on the risk insights gained to date (Appendix D). The scope of the assessment presented here is limited to examining if data gathered and methodologies developed by DOE are likely to be adequately documented for the staff to undertake a detailed technical review. This assessment is not a regulatory compliance determination review of a potential license application.

5.1.3.1.2 Relationship to Key Technical Issue Subissues

The Degradation of Engineered Barriers Integrated Subissue incorporates subject matter previously described in the following 13 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 and 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 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)
- Thermal Effects on Flow: Subissue 2—Thermal Effects on Temperature, Humidity, Saturation, and Flux (NRC, 2000a)

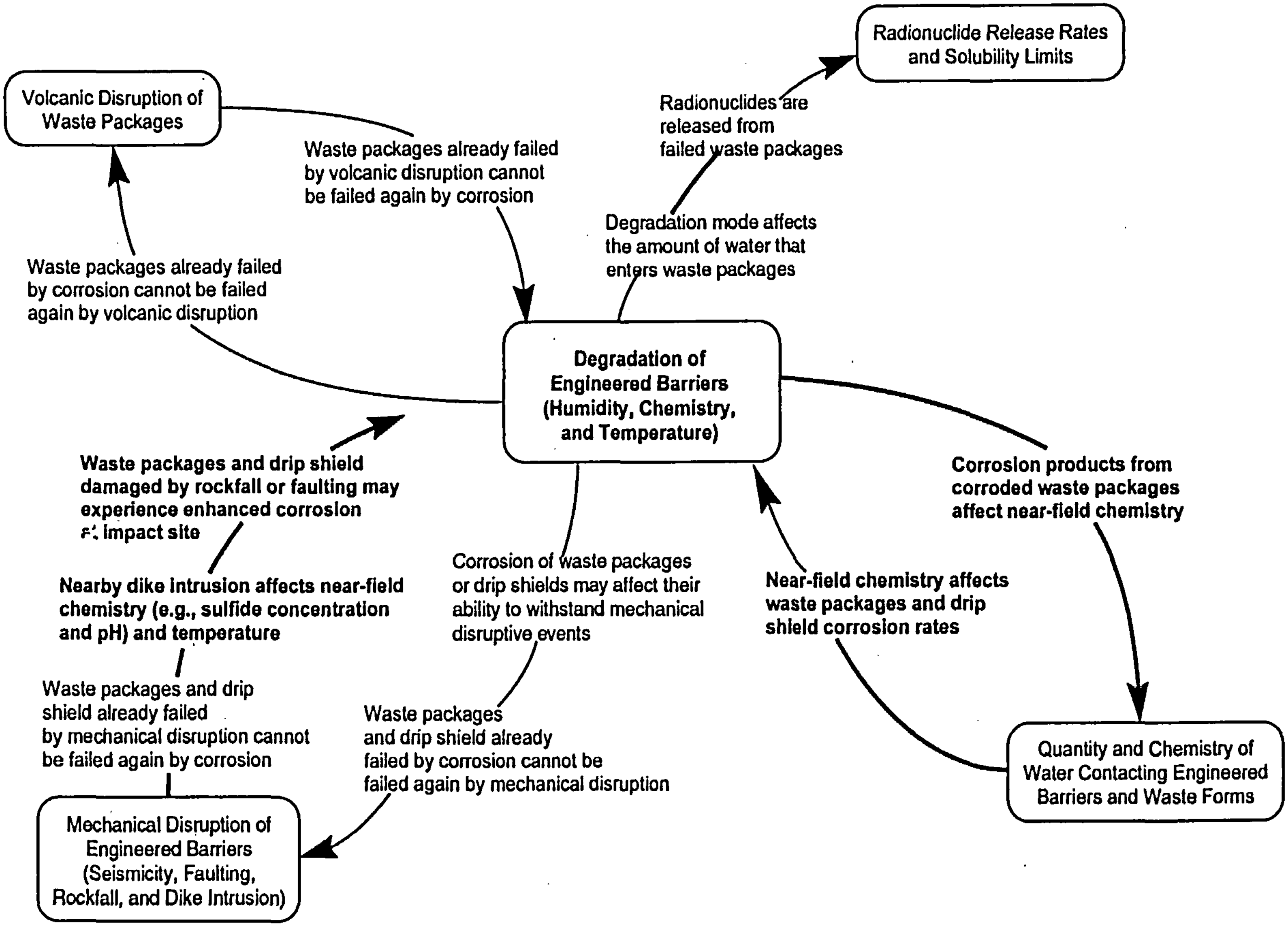

Figure 5.1.3.1-1. Diagram Illustrating the Relationship Between Engineered Barrier Degradation and Other Integrated Subissues. Material in Bold Is Identified in the Text.

- Evolution of the Near-Field Environment: Subissue 2—Effects of Coupled Thermal-Hydrological-Chemical Processes on the Waste Package Chemical Environment (NRC, 2000b)

- Evolution of the Near-Field Environment: Subissue 3—Effects of Coupled Thermal-Hydrological-Chemical Processes on Chemical Environment for Radionuclide Release (NRC, 2000b)

- Evolution of the Near-Field Environment: Subissue 5—Effects of Coupled Thermal-Hydrological-Chemical Processes on Potential Nuclear Criticality in the Near Field (NRC, 2000b)

- Repository Design and Thermal-Mechanical Effects: Subissue 3—Thermal-Mechanical Effects on Underground Facility Design and Performance (NRC, 2000c)

- Total System Performance Assessment and Integration: Subissue 1—System Description and Demonstration of Multiple Barriers (NRC, 2000d)

- Total System Performance Assessment and Integration: Subissue 2—Scenario Analysis and Event Probability (NRC, 2000d)

- Total System Performance Assessment and Integration: Subissue 3—Model Abstraction (NRC, 2000d)

- Total System Performance Assessment and Integration: Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards (NRC, 2000d)

DOE has not included nuclear criticality within the waste package as part of the degradation of engineered barriers model abstraction. DOE has indicated that it intends to exclude nuclear criticality events from the performance assessment based on low probability. The DOE evaluation of nuclear criticality is assessed in Section 5.1.2.2, Identification of Events with Probabilities Greater Than 10^{-8} Per Year, of this report

The key technical issue subissues formed the bases for the previous 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 a 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.

5.1.3.1.3 Importance to Postclosure Performance

One aspect regarding risk-informing the NRC review is to determine how this integrated subissue on degradation of engineered barriers relates to the DOE repository safety strategy. The primary components of the engineered barrier system are the drip shield and the waste package. Risk insights pertaining to the degradation of waste package and drip shield indicate that the persistence of a passive film on the surface of a waste package is of high significance to waste isolation. Waste package failure mode, drip shield integrity, and stress corrosion cracking are assigned medium significance, and juvenile failures of the waste package is assigned low significance. The details of the risk insights ranking are provided in Appendix D.

The drip shield and waste package can protect the wasteform from dripping water while they remain intact, thereby limiting both the timing and magnitude of radionuclide release. The drip shield may also limit the exposure of the waste package to aggressive chemical environments resulting from thermal-hydrological-chemical processes, as well as mitigate mechanical damage to the waste package from falling rocks. These engineered barriers will eventually fail at some time in the future by corrosion and mechanical degradation processes. Mechanical disruption of engineered barriers is addressed in Section 5.1.3.2. Release of radionuclides from the breached waste packages will depend on the location and cross-sectional area of the breaches through the waste packages.

In CRWMS M&O (2000c, Section 5.3.2), analyses were performed by DOE to compare a degraded waste package barrier with an enhanced waste package barrier using the basecase as reference. The parameters for the degraded waste package case were set at the 95th percentile value of their uncertainty distribution and, for the enhanced waste package case, at the 5th percentile. The parameters included the general corrosion of Alloy 22, the microbially influenced corrosion factor for the Alloy 22 general corrosion rate, the multiplication factor for the Alloy 22 general corrosion rate due to aging and phase instability, the residual hoop stress state and stress intensity factor at the closure-lid welds, and the number of manufacturing defects per waste package at the closure-lid welds. The enhanced waste package case yielded no waste package failure and therefore no dose, whereas the degraded waste package case exhibited a relatively large fraction of failed waste packages (0.01) in 10,000 years. The first waste package failure occurred at 7,000 years. In contrast, the basecase displayed waste package failures only beyond a period of 10,000 years. For the degraded case, there is a 50-percent probability that 1 percent of the waste packages will fail at approximately 10,000 years. Because the degraded case is associated with more waste packages failing in 10,000 years than the basecase, the mean dose is higher {just below 0.01 mSv/yr [1 mrem/yr] at the end of 10,000 years in the degraded case}.

Similar analyses were performed by Bechtel SAIC Company, LLC, (2002a) comparing the basecase waste package performance for the nominal scenario with three computational variations. The computational variations corresponded to different general corrosion rates. Higher corrosion rates yielded higher doses at earlier times, which is an intuitive result. Full neutralization of the waste package (with the drip shield remaining intact) resulted in a maximum dose of 0.1 mSv/yr [10 mrem/yr] (at around 2,000 years) in a 10,000 year period.

Performance assessment sensitivity analyses by NRC (Mohanty, et al., 2002) using the TPA Version 4.1 code also indicated the importance of the waste package on the performance of the whole repository. One of the 10 most influential parameters in the sensitivity analyses for the basecase performance scenario is the waste package flow multiplication factor, which is the fraction of dripping water entering a breached waste package. The defective fraction of waste packages, related to juvenile failure, is another influential parameter.

A degraded drip shield was compared with an enhanced drip shield using the basecase as reference (CRWMS M&O, 2000c, Section 5.3.2). As in the case of the waste package, the parameters for the degraded drip shield case were set at the 95th percentile value of their uncertainty distribution and for the enhanced drip shield case at the 5th percentile. Only general corrosion was considered, because it is the only degradation process of the drip shield accounted for in the basecase. The results showed intuitive trends such as the enhanced case yielding later drip shield failure times than the basecase and degraded case. The results show practically no influence of the drip shield on the calculated mean dose, because the model abstraction for waste package degradation is independent of the drip shield performance, although radionuclide release could be a function of drip shield protection.

Bechtel SAIC Company, LLC (2002a) also compared the basecase for the nominal scenario with two computational variations (enhanced drip shield degradation and neutralization of the drip shield). The dose curve for the basecase exhibited some differences from that calculated in the Total System Performance Assessment for the site recommendation. The dose is close to the 1×10^{-6} mSv/yr [1×10^{-4} mrem/yr] calculated for the basecase after approximately 3,000 years. If the drip shield is fully neutralized (with the waste package remaining intact), the dose increases to 1×10^{-5} mSv/yr [1×10^{-3} mrem/yr] in fewer than 1,000 years, slowly increasing as a function of time by less than an order of magnitude at the end of a 10,000-year period.

In summary, DOE identified degradation of the waste package as one of eight principal model components of the total-system performance assessment for the potential license application (Bechtel SAIC Company, LLC, 2002b), whereas degradation of the drip shield is not considered important to repository performance. In the sensitivity analysis conducted by NRC and the Center for Nuclear Waste Regulatory Analyses (CNWRA) staffs using the TPA Version 4.1 code (Mohanty, et al., 2002), degradation of engineered barriers was rated at the top of the other three model abstractions, related to the engineered system, for its contribution to performance. Appendix D designates the degradation of engineered barriers as having a medium to high significance to waste isolation. The following assessment of the DOE characterization and performance assessment abstraction of degradation of engineered barriers was conducted at a level of detail commensurate with the assigned degree of significance.

5.1.3.1.4 Technical Basis

NRC developed a review plan (NRC, 2003) consistent with the acceptance criteria and review methods found in the previous issue resolution status reports. A status assessment of the DOE approaches for including the degradation of engineered barriers in total system performance assessment abstractions is provided in the following subsections. This assessment is organized according to the five review methods identified in Section 2.3: (i) Model Integration (including system description), (ii) Data and Model Justification, (iii) Data Uncertainty, (iv) Model Uncertainty, and (v) Model Support.

Each review method requires a reviewer to evaluate, verify, or confirm what information DOE has presented in support of its specific model abstractions or design features of the potential repository. In updating this status report, staff considered how the current information informs understanding of the DOE view of the potential repository system as it relates to the specific review methods. Specifically, the staff examined the information available to date to determine if DOE has provided the information listed in the agreements or is in the process of acquiring such information.

Drip Shield and Waste Package Environments. Based on the thermal seepage model, DOE considers that when the drift temperature is above a threshold value of 100 °C [212 °F], no ground water will reach the waste package or drip shield by seepage (Bechtel SAIC Company, LLC, 2003a). Above this threshold temperature, the chemical environment on the surfaces of the waste package and drip shield will be dominated by the chemistry of the dusts that may be deposited onto the waste package or the drip shield during the ventilation period or after the closure of the drift, and the ability of these dusts to form an aqueous solution by deliquescence in moist air. Below this threshold temperature, ground water may reach the waste package and drip shield by seepage and thus the chemistry of the ground water will dominate the chemical environment on the surfaces of the waste package and drip shield. Therefore, aqueous corrosion of the waste package and drip shield was considered in two scenarios, the dust deliquescence scenario and the crown seepage scenario (Bechtel SAIC Company, LLC, 2003a). A summary of these scenarios is presented next. A detailed evaluation of these scenarios is presented in Section 5.1.3.3, Quantity and Chemistry of Water Contacting Engineered Barriers and Waste Forms.

In the dust deliquescence scenario, an aqueous solution is formed by the sorption of moisture from the air by hygroscopic salts in the dusts when the in-drift relative humidity is at or above the mutual deliquescence relative humidity of the salt mixtures. The aqueous solution will be concentrated and its chemistry will be controlled by the composition of the salts and the in-drift relative humidity. In the crown seepage scenario, concentrated aqueous solutions could be directly formed from the seepage water. When the in-drift relative humidity is low, the seepage water dripping on the waste package or drip shield surfaces will be evaporated and concentrated until the water in the aqueous solution is at equilibrium with the water in the gas phase. Thus, the chemistry of the seepage solution will be determined by the seepage water composition and the in-drift relative humidity.

For the seepage solution, DOE modeled evaporation using the EQ3/6 program on ground waters that were considered representative of the seepage water compositions based on the thermal-hydrologic-chemical model (Bechtel SAIC Company, LLC, 2003a). Based on the final compositions of the evaporated waters, DOE grouped these waters into 11 bins to encompass the range of the seepage waters in the potential repository system (Bechtel SAIC Company, LLC, 2003a). Based on the number of waters in each bin and the number of chemistries each water represents, each bin is given a probability of occurrence in 20,000 years.

For the solutions generated from dust, DOE similarly modeled evaporation on the leachates of 52 dust samples collected from the exploratory studies facility. As with the seepage waters, DOE grouped the leachates into six bins which encompass the range of the dust leachates. The composition of each water that was chosen to represent the range of waters in each of the 11 seepage water bins and the composition of each of the dust leachates chosen to represent the chemistry range of the leachates in each of the six leachate bins are discussed in Section 5.1.3.3. The end point brines evaporated from the representative waters in the 11 seepage water bins can be classified into the following six types of concentrated brines:

- Chloride dominated by Ca-Cl constituents
- Chloride dominated by Ca-Cl and K-Cl constituents
- Sulfate dominated by K-NO_3 and Na-NO_3 constituents
- Sulfate dominated by Na-Cl and K-Cl constituents

- Carbonate dominated by Na-Cl, $Na-NO_3$, and K-Cl constituents
- Carbonate dominated by $K-NO_3$ and K-Cl constituents

The end point brines evaporated from the representative dust leachates in the six leachate bins can be classified into the following four types of concentrated brines:

- Chloride dominated by $Ca-NO_3$ constituents
- Sulfate or carbonate dominated by $K-NO_3$ and $Na-NO_3$ constituents
- Sulfate dominated by $Na-NO_3$ constituents
- Carbonate dominated by $K-NO_3$ constituents

As mentioned previously, the concentrations of these brines depend on the in-drift relative humidity. Above the deliquescence point of the salt mixture (or the mutual deliquescence relative humidity), the lower the in-drift relative humidity, the more concentrated the brine solution. When the in-drift relative humidity is below the mutual deliquescence relative humidity, salts will crystallize and no aqueous solution will be present.

For the seepage scenario, the lowest relative humidity that needs to be considered is the value corresponding to the seepage threshold drift temperature. In the DOE analysis, 100 °C [212 °F] was considered the threshold temperature (Bechtel SAIC Company, LLC, 2003d); in recent work[1] 105 °C [221 °F] was used as the threshold temperature. According to the models developed independently by DOE (Bechtel SAIC Company, LLC, 2003e) and CNWRA (Fedors, et al., 2004), the relative humidity corresponding to the threshold temperatures {100 to 105 °C [212 to 221 °F]} is approximately 60 to 65 percent for the case of no-drift degradation. Therefore, the lowest in-drift relative humidity during the seepage scenario in the case of no-drift degradation would be 60 percent and the concentration of the salts will be bounded by the salt solution in equilibrium with the water vapor in the gas phase at 60-percent relative humidity. However, for the dust deliquescence scenario, the lowest relative humidity to consider is the mutual deliquescence relative humidity of the salt mixtures. Therefore, the most concentrated brine, for the dust deliquescence scenario, will be the solution that is at equilibrium with the water vapor in the gas phase at the mutual deliquescence relative humidity of the salt mixture.

The above mentioned brines derived by DOE to represent the aqueous solutions formed during the evaporation of seepage water or dust leachates may be divided into two types, those that contain calcium and those that do not contain calcium. As discussed in Section 5.1.3.3, the mutual deliquescence relative humidity of the calcium-containing salt mixtures is approximately 15 percent and is independent of temperature. However, these salts are unstable beyond a threshold temperature, which is known to be lower than 150 °C [302 °F]. The analysis in Section 5.1.3.3 also indicates that the likely mutual deliquescence relative humidities of the noncalcium salt mixtures are approximately 35 to 25 percent at temperatures from 100 to 145 °C [212 to 293 °F] and are expected to be lower at temperatures above 145 °C [293 °F]. The decomposition products of the unstable calcium-containing salts at temperatures above the threshold value may contain acid gases such as HCl and HNO_3. A large quantity of such

[1]Browning, L., R. Fedors, L. Yang, O. Pensado, R. Pabalan, C. Manepally, and B. Leslie. "Estimated Effects of Temperature-Relative Humidity Variations on the Composition of In-Drift Water in the Potential Nuclear Waste Repository at Yucca Mountain, Nevada." Materials Research Society Symposium CC: Scientific Basis for Nuclear Waste Management XXVIII, San Francisco, California, April 12–16, 2004. L. Browning and J. Hanchar, eds. Warrendale, Pennsylvania: Materials Research Society. In press. 2004.

calcium-containing salts may generate enough acid gases to alter the acidity of the brine solution or the condensate vapor. However, the drift wall and the dusts that are deposited in the drift or mixed with the calcium-containing salts have large surface areas and may adsorb or react with the acid gases. Sufficient information should be available to evaluate the effect of this decomposition on the acidity of the brine solution or the condensate.

Based on modeling of the brine chemistry during evaporation and the composition of the above-mentioned evaporation end-point brines, DOE used 5 complex waters and several simple solutions that were composed of $CaCl_2$, $CaCl_2 + Ca(NO_3)_2$, or NaCl salt to bound the compositions of the brines from the 11 bins of seepage waters and 6 bins of dust leachates and their chemistries during the evaporation at different plausible relative humidities in terms of corrosion behaviors. These five waters were derived based on the studies of J–13 Well water, which was the composition adopted as a reference for corrosion testing in the Yucca Mountain Project (CRWMS M&O, 2000d). The chemical compositions of these waters are given in Table 5.1.3.1-1 (CRWMS M&O, 2000d; Bechtel SAIC Company, LLC, 2003a).

Degradation of the Drip Shield. The drip shield design DOE has proposed to use in a potential license application 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) plates and structural members made of Titanium Grade 24 (Ti-6Al-4V-0.15Pd) for long-term structural support (CRWMS M&O, 2000e). 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 crushed tuff as ballast. The emplacement drifts will have perforated stainless steel sheets and rock bolts for ground support of the drift walls and roof.

Table 5.1.3.1-1. Molar Concentration of Key Species in Simulated Diluted Water, Simulated Concentrated Water, Simulated Saturated Water, Simulated Acidified Water, and Basic Saturated Water*†

Species	Simulated Diluted Water	Simulated Concentrated Water	Simulated Saturated Water	Simulated Acidified Water	Basic Saturated Water
K^+	0.0009	0.09	3.62	0.09	1.73
Na^+	0.0178	1.78	2.12	1.78	4.60
Mg^{2+}	0.0000	0.00	0.00	0.04	0.00
Ca^{2+}	0.0000	0.00	0.00	0.02	0.00
F^-	0.0007	0.07	0.00	0.00	0.08
Cl^-	0.0019	0.19	3.62	0.68	3.69
NO_3^-	0.0010	0.10	21.1	0.37	22.52
SO_4^{2-}	0.0017	0.17	0.00	0.40	0.15
HCO_3^-	0.0155	1.15	0.00	0.00	0.00
pH	9.8 to 10.2	9.8 to 10.2	5.5 to 7	2.7	12

Performance of the drip shield should be considered as an important factor regarding safety because DOE has stated it intends to incorporate it into the design of the engineered barrier system to provide defense-in-depth by limiting the amount of water contacting the waste package as a result of dripping of seepage water and providing additional protection to the waste package from mechanical loads as a consequence of rockfall. Mechanical disruption of engineered barriers is addressed in Section 5.1.3.2. The quantity and chemistry of the water contacting the waste package and effects on corrosion modes and rates will depend on the integrity of the drip shield. Hence, the initiation of aqueous corrosion of waste packages can be delayed, resulting in a significantly longer container lifetime. Before drip shield failure, however, aqueous solutions may develop by hydration of salts present in the dust above a certain critical value of the relative humidity, which is dependent on the salt or salt mixtures and temperature. 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 transport of the released radionuclides could be limited by the presence of the drip shield, even if partially damaged.

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, 2000a) include thermal embrittlement, dry-air oxidation, humid-air corrosion, uniform aqueous corrosion including accelerated corrosion in the presence of fluoride-containing ground waters, localized (pitting and crevice) aqueous corrosion, and environmentally assisted cracking (consisting of stress corrosion cracking and hydrogen embrittlement or hydrogen-induced cracking).

Degradation of the Waste Package. The waste package, composed of the containers and the wasteforms, is the primary engineered barrier controlling the release of radionuclides to the geosphere. It should be noted that, in contrast to the definitions of 10 CFR Part 63, DOE defines the waste package with the exclusion of the wasteforms. 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. In the proposed DOE reference design for the various types of spent nuclear fuel and high-level waste glass to be included in the potential license application (Anderson, et al., 2003), the waste package is composed (in addition to the various wasteforms) of two concentric containers of different alloys 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.

For environmental conditions where the passive oxide film is stable, the corrosion rate of Alloy 22 is slow, and DOE models project container lifetimes greater than 10,000 years projected. Aggressive environmental conditions can disrupt passive film stability and may decrease container lifetimes. Corrosion processes potentially important in the degradation of the waste package and drip shield include humid-air and uniform aqueous corrosion, localized (pitting, crevice, and intergranular) corrosion, microbially influenced corrosion, stress corrosion cracking, and hydrogen embrittlement. 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, however, the rate of dry-air oxidation under anticipated repository conditions is expected to be low. 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 aqueous corrosion listed previously. The corrosion failure mode and morphology will effect significantly the amount of water that can enter the waste package and, in turn, can alter the release rate of radioactive material.

Review of Processes Screened Out. The possibility for thermal embrittlement of titanium used in drip shield construction was excluded from 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 5.1.2.1. Potential effects on the temperature of the drift shield from degradation of the emplacement drifts were also screened out by DOE (CRWMS M&O, 2000g). This type of drift degradation process, however, may have an important effect on the integrity of the drip shield and should be considered, as discussed in detail in Section 5.1.3.2, Mechanical Disruption of Engineered Barriers.

The possibility of degradation of the titanium drip shield by dry-air oxidation was excluded from further analysis by DOE because the dry-air oxidation rate is insufficient to penetrate the drip shield in 10,000 years (Bechtel SAIC Company, LLC, 2003b). The increase in the oxide film thickness was calculated using Fick's first law and assuming a linear concentration gradient across the oxide film of thickness. Data on the titanium oxide film thickness as a function of time and temperature (Schutz and Thomas, 1987) were used to determine the activation energy for oxidation. At a constant temperature of 200 °C [392 °F], the titanium oxide thickness is expected to be approximately 0.0021 mm [8.3×10^{-5} in] in 10,000 years. At 400 °C [752 °F], the oxide layer thickness was calculated to be 0.087 mm [3.4×10^{-3} in] in 10,000 years. Based on these calculations, the oxide film growth rate was determined to be insignificant to drip shield performance at temperatures below 400 °C [752 °F].

The possibility of microbially influenced corrosion of the titanium drip shield was excluded from further analysis because titanium alloys are generally immune from microbially influenced corrosion processes (Bechtel SAIC Company, LLC, 2003b). Titanium alloys are reported to be susceptible to biofouling but the stability of the TiO_2 passive film provides immunity from microbially influenced corrosion (Revie, 2000). The DOE assessment is supported by the results of other investigations that indicate titanium is extremely resistant to localized corrosion in particularly severe microbial environments (Geesey and Cragnolino, 1995). Titanium has been extensively used in corrosive marine environments, and no failures have been reported. Titanium samples tested for 20 years in marine environments showed no sign of corrosion (Schutz, 1991). Sufficient information should be available to evaluate if the titanium drip shield is susceptible to microbially influenced corrosion under the expected repository conditions.

5.1.3.1.4.1 Passivity and Uniform Corrosion of the Drip Shield

5.1.3.1.4.1.1 Model Integration

The integrity of the drip shield will influence the quantity and chemistry of the water that can develop on the waste package and the potential effects on corrosion modes and rates. Analyses performed by DOE for the total system performance assessment for site recommendation show the drip shield has little effect on repository performance. However, the role of the drip shield to control the formation of aggressive environments on the waste package surface was not included in the DOE model. Higher doses observed with accelerated drip shield failure are attributed to water seepage, which is not diverted by the drip shield, and hence contacts the breached waste packages.

While the drip shield is intact, water that contacts the waste package may be limited to condensed water with low concentrations of aggressive species that are unlikely to enhance corrosion. The uniform corrosion rate of titanium alloys, however, is dependent on the fluoride concentration. Faster corrosion rates and shorter failure times may occur on drip shield sections exposed to solutions with fluoride concentrations greater than 10^{-4} M (Brossia, et al., 2001). However, titanium corrosion may be limited by the availability and supply of fluoride from dripping water, and not strictly by the concentration threshold for accelerated corrosion (Lin, et al., 2003). Failure of the drip shield by corrosion degradation processes in combination with mechanical disruption may allow the formation of aggressive environments in contact with the waste package surface and lead to accelerated failure of the waste package. The formation of aggressive environments depends on many factors, with different degrees of uncertainties, that are related to the deposition of deliquescent salts, the rate of evaporation, and the composition of the seepage water.

The model abstraction for the general corrosion of the drip shield, corresponding to uniform passive dissolution, is modeled based on a cumulative distribution function derived from weight-loss data obtained from experiments conducted in the long-term corrosion test facility (Bechtel SAIC Company, LLC, 2003b). The data were obtained using Titanium Grade 16 instead of Titanium Grade 7. It is argued by the DOE that the corrosion rate of Titanium Grade 16 (titanium-palladium alloy, which contains 0.04–0.08 wt% palladium) bounds that of Titanium Grade 7 (which contains 0.12–0.25 wt% palladium) (Bechtel SAIC Company, LLC, 2003b). Specimens were exposed for 1 year to simulated diluted water, simulated concentrated water, and simulated acidified water and for 5 years to simulated diluted water and simulated concentrated water at 60 and 90 °C [140 and 164 °F]. A slight influence of temperature was observed but there was not a noticeable effect from the testing environment.

Since the temperature dependence of corrosion rate was only tested at two temperatures, further confirmation of the lack of dependence may be useful. In general, however, the information provided by DOE on its empirical model of general corrosion of the drip shield appears to be sufficient for use in developing a potential license application.

Two types of specimens were used in experiments at the long-term corrosion test facility to evaluate the corrosion rate. One was a flat coupon for weight loss measurements and the other incorporated a crevice former. The data from the coupon specimens were used to develop the cumulative distribution function that describes the distribution of general corrosion rates in the inner surface of the drip shield. The combined data obtained using the flat coupons and the crevice specimens were used to develop the cumulative distribution function representing the general corrosion rates on the outer surface of the drip shield. It is argued by the DOE that the outer surface of the drip shield will be exposed to a more complicated water chemistry because dust, salt deposits from evaporation of the seepage water, or both may form crevices with occluded environments (Bechtel SAIC Company, LLC, 2003b). In contrast, the inner surface of the drip shield will be exposed only to condensed water, possibly with dust.

While the general approach adopted by DOE for general corrosion of the drip shield appears to be sufficient, DOE should provide additional information beyond that in Bechtel SAIC Company, LLC (2003b) regarding the conditions expected on the outer drip shield surface, in terms of possible crevice enhancement effects on corrosion (e.g., effect of rubble from rockfall).

Taking into account the results obtained by Brossia, et al. (2001) regarding the detrimental effect of fluoride on the stability of the passive film on titanium and its alloys, DOE has argued that the effect is only noticeable in freshly polished specimens but attenuated in the case of oxide covered specimens (Bechtel SAIC Company, LLC, 2003b). No evidence of this assertion is presented, however, with the exception of the results obtained in the long-term corrosion test facility indicating that no accelerated corrosion was observed despite the fact that the simulated concentrated water contains a fluoride concentration of approximately 0.085 mol/L [0.32 mol/gal]. The lack of accelerated corrosion is attributed to the effect of the oxide layer formed by prolonged exposure of the specimens to the atmosphere prior to corrosion testing or through higher temperature exposure also prior to the test (Hua, et al., 2004). However, only the first situation can be considered a valid argument because no specimen preoxidized at high temperature was used by DOE in the long-term corrosion test facility testing. Other arguments were based on the consumption of fluoride anions by the SiO_2 present in the test solutions or the precipitation by Ca^{2+} cations. With the exception of the effect of Ca^{2+} cations, Brossia, et al. (2001) have demonstrated that the other anions such as nitrate and sulfate present in ground waters do not inhibit the accelerating effect of fluoride on the uniform corrosion rate. However, as suggested by Lin, et al. (2003), the availability and rate of supply of fluoride to the drip shield could be a controlling factor limiting the accelerated corrosion of the drip shield, an aspect that requires further evaluation.

Additional information, presumably contained in the revised version of CRWMS M&O (2000h), is needed to conclude that generalized passivity breakdown by the action of fluoride will not occur under repository conditions. According to agreement CLST.6.01, DOE should confirm that no deleterious effect of fluoride on the uniform corrosion rate of titanium drip shields exists. Because it is known that fluoride increases the corrosion rate, the availability and supply of fluoride ions to the drip shield surface should be estimated by DOE to evaluate the extent of the detrimental effect of fluoride on drip shield integrity.

Although galvanic coupling of titanium to active metals such as carbon steels may cause hydrogen absorption by titanium and its alloys, Hua, et al. (2004) emphasize that galvanic coupling of Titanium Grade 7 with passive metals or alloys is unlikely to promote accelerated corrosion. The argument is plausible because the corrosion potential of Titanium Grade 7 would be very close to that of passive metals or alloys such as stainless steels. However, DOE has not provided data or appropriate references to support the conclusion that the driving force is insufficient to promote accelerated corrosion.

Certain aspects related to the definition of the environment that may contact the inner and outer surfaces of the drip shield—such as the concentration and availability of fluoride—lack an adequate technical basis. The removal of fluoride from evaporating seepage waters by precipitation of CaF_2 needs further justification considering the competitive precipitation of $CaCO_3$. To address these concerns, DOE agreed (Schlueter, 2000) to provide additional information on credible environmental conditions, including the composition of the water contacting the drip shield, and the concentration and availability of fluoride, which is one of the most deleterious species.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.1.5), is sufficient to expect that the information necessary to assess the passivity and uniform corrosion of the drip shield with respect to system description and model integration will be available at the time of a potential license application.

5.1.3.1.4.1.2 Data and Model Justification

Experimental results used by DOE in its data and model justification are summarized in Bechtel SAIC Company, LLC (2003b) and its appendixes. The corrosion rate data were obtained using Titanium Grade 16 coupon and crevice specimens exposed for 1 year to simulated diluted water, simulated concentrated water, and simulated acidified water and for 5 years to simulated diluted water and simulated concentrated water at 60 and 90 °C [140 and 164 °F]. A wide variation in the measured weight loss, resulting in corrosion rates of approximately −1,700 to 150 nm/yr [-6.7×10^{-2} to 5.9×10^{-3} mpy], was reported for the coupons after a 1-year exposure. It is apparent from the negative values that the data include specimens exhibiting significant weight gain. The variability was explained as a result of differences in the postexposure cleaning procedures used to remove corrosion product buildup. Creviced specimens were simultaneously tested but no significant attack observed under the crevice former. In this case, rates ranging from −350 to 320 nm/yr [-1.4×10^{-2} to 1.3×10^{-2} mpy] were calculated.

The maximum corrosion rate for the distribution of the coupon specimens after 5 years exposure was 58 nm/yr [2.3×10^{-3} mpy], with most values below 30 nm/yr [1.2×10^{-3} mpy]; the creviced specimens exhibited a maximum of approximately 77 nm/yr [3.0×10^{-3} mpy], with most values below 30 nm/yr [1.2×10^{-3} mpy]. The median corrosion rate was found to be 5 nm/yr [2.0×10^{-4} mpy] for the planar specimens and 10 nm/yr [4.0×10^{-4} mpy] for the creviced specimens. The 5-year exposure tests revealed the median corrosion rate increased from 0 nm/yr [0 mpy] after 1 year to 5 and 10 nm/yr [2.0×10^{-4} and 4.0×10^{-4} mpy] for the planar and crevice specimens, respectively. The median corrosion rate of 0 nm/yr [0 mpy] after 1 year is the result of the weight gain experienced by a large number of specimens. The maximum corrosion rate decreased with exposure time, indicating a narrowing of the distribution of corrosion rates with time. This effect may result from the different posttest treatments of the specimens for the 1- and 5-year tests. The 1-year test specimens were cleaned with distilled water and a nylon brush, whereas the 5-year specimens were cleaned thoroughly using a chemical method.

The corrosion rate data obtained in the 1-year test were conservatively adopted for the model abstraction after eliminating from the distribution the negative values (Bechtel SAIC Company, LLC, 2003b). This treatment of the data yields uniform corrosion rates with a median of 18 nm/yr [7.0×10^{-4} mpy] and a maximum of 113 nm/yr [4.4×10^{-3} mpy] for the coupon specimens and a median of 25 nm/yr [1.0×10^{-3} mpy] and a maximum of 319 nm/yr [1.24×10^{-2} mpy] for the combined distribution using coupon and crevice specimens.

The values of the corrosion rate adopted for the model abstraction seem to be reasonable. Additional information provided by DOE in response to CLST.1.07 addressing validity of data obtained in immersion tests and the use of alternative methods (Bechtel SAIC Company, LLC, 2003b, Appendix A) are being evaluated, but the relevant information regarding Titanium Grade 7 should be available in a revised version of CRWMS M&O (2000h).

A slight influence of temperature was observed between the results at 60 and 90 °C [140 and 194 °F] and there was not a significant effect of the testing environment. Since the corrosion rates were similar for the uniform corrosion coupons and the crevice corrosion coupons, DOE assumed that the main corrosion mode for the creviced specimens was also uniform passive corrosion of the exposed surfaces.

The values of uniform corrosion rates adopted for the model abstraction are considered by the DOE to be independent of both temperature and the composition of the aqueous environment (Bechtel SAIC Company, LLC, 2003b). This criterion seems to be justifiable on the basis of the limited data available. It should be noted, however, that the corrosion data were obtained in three environments at two temperatures below the boiling point at the potential repository horizon. Although the uniform corrosion rate is expected to be similar in widely different environments if passivity is maintained, high temperatures may have a noticeable effect on corrosion rates. Additional information will be needed to estimate the effect of temperature on passive corrosion rate if temperatures above 100 °C [212 °F] persist over prolonged periods on the drip shield surfaces covered with an aqueous electrolyte.

DOE refers to a study of Covington and Schutz (1981) conducted in a marine environment to confirm that a maximum corrosion rate of about 25 nm/yr [1.0×10^{-3} mpy] could be expected for titanium. It is also noted that approximately half of the specimens were exposed to the vapor above the aqueous phase and the other half were submerged, a few samples were located at the water line. No difference in corrosion rates was observed among the three environmental conditions.

Using potentiostatic methods, electrochemical measurements of passive current density yield corrosion rates of approximately 870 nm/yr [3.4×10^{-2} mpy] after 1,150 hours exposure at 95 °C [203 °F] (Brossia, et al., 2001; Brossia and Cragnolino, 2001a). This and other measurements resulting in values one order of magnitude lower (Brossia, et al., 2001; Brossia and Cragnolino, 2001a; Cragnolino, et al., 1999) indicate that the values adopted by DOE, although somewhat low, are consistent with data in the open literature.

Although the data and the model abstraction for the general corrosion seem adequate, there are certain aspects related to the conditions of the tests and the influence of the crevice in the results that require additional explanation as requested in CLST.6.01. To address these concerns, DOE agreed (Schlueter, 2000) to provide the additional information on the tests conditions.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.1.5), is sufficient to expect that the information necessary to assess the passivity and uniform corrosion of the drip shield with respect to data being sufficient for model justification will be available at the time of a potential license application.

5.1.3.1.4.1.3 Data Uncertainty

The most important implication of data uncertainty is related to the estimate of the distribution of drip shield failure times. The maximum error in determining the corrosion rates from weight loss measurements in the case of a 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. There is no simple approach to estimate the uncertainties associated with environmental variations.

DOE has opted to use the 1-year test data as upper bound values for the uniform corrosion rate. Though data have been obtained to determine the rates associated with uniform 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 or with data from field tests or other applications. This confirmation is particularly important because the analysis and model report (CRWMS M&O, 2000h) states that the weight-loss measurements are at or below the reliable detection limit, yet these measurements are used as bounding values. NRC is unable to evaluate the data uncertainty related to drip shield corrosion based on the information currently provided by DOE (Bechtel SAIC Company, LLC, 2003b). However, it is expected that the information will be included in the revised version of CRWMS M&O (2000h). In principle, the low corrosion rates measured from weight-loss experiments need to be confirmed with other tests designed to sensitively measure the passive corrosion rate or experience on field tests or other applications.

DOE has agreed to provide information through their responses to agreements CLST.1.07 and 6.01, to confirm uniform corrosion rates using alternative methods.

In addition, uncertainties related to the potential 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 such a 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 (Schlueter, 2000) to provide the additional information on the uniform corrosion from alternative test methods and on the fluoride concentration of the ground water in contact with drip shields and its effects on accelerated drip shield corrosion and hydrogen uptake.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.1.5), is sufficient to expect that the information necessary to assess the passivity and uniform corrosion of the drip shield with respect to data uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.1.4.1.4 Model Uncertainty

The corrosion rates measured (approximately 10 to a few hundreds of nanometers per year) using weight loss methods appear to be more reliable in the 5-year tests than those measured in 1-year tests, presumably due to the extended testing period and the improvement in the cleaning method used for the specimens exposed for 5 years. As noted previously, DOE has used 1-year test data to bound the uniform corrosion rate used in the model abstraction.

DOE has not presented alternative models for measuring general passive corrosion. The model used is empirical and based only on the experimental determination by weight loss of corrosion rates (CRWMS M&O, 2000h). As a result, model uncertainty may affect the confidence to predict life of the drip shield for thousands of years. Most of the models on passivity are empirical in nature, although some of them have a substantial level of mechanistic support that can be found in the open corrosion literature and could be applicable to titanium alloys.

Therefore, a comprehensive database based on laboratory corrosion tests, in-service measurements, and other sources of data could be useful to reduce these potential model uncertainties and gain confidence on drip shield lifetime estimates. It is expected that the revised version of CRWMS M&O (2000h) will contain such information. To address this concern, DOE agreed (Schlueter, 2000) to provide sufficient data on uniform corrosion from more sensitive and alternative test methods.

The effect of the environment variables is not considered explicitly in the model abstraction because only three environments and two temperatures are used in the DOE corrosion tests. A single distribution of corrosion rates was used regardless of the composition of the environment and temperature. While information to evaluate the uncertainty related to drip shield corrosion is not fully available in Bechtel SAIC Company, LLC (2003b), DOE has agreed to provide additional information in their responses to those agreements related to the drip shield.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.1.5), is sufficient to expect that the information necessary to assess the passivity and uniform corrosion of the drip shield with respect to model uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.1.4.1.5 Model Support

The potential detrimental effects of fluoride on the corrosion behavior of titanium should be further explored. 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. DOE should provide a final evaluation of the amount and concentration of available fluoride in the evaporated ground water and the flow rate toward the drip shield, which controls the supply of fluoride to its surface. DOE's current approach to the chemistry of water that may contact the engineered barrier system is discussed in more detail in Section 5.1.3.3.

This subject is also important in relation to the mechanical disruption of the engineered barriers integrated subissue, as described in Section 5.1.3.2. 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, the DOE agreed (Schlueter, 2000) to provide the additional information on the effect of wall thinning from accelerated uniform corrosion.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.1.5), is sufficient to expect that the information necessary to assess the passivity and uniform corrosion of the drip shield with respect to model abstraction output being supported by objective comparisons will be available at the time of a potential license application.

5.1.3.1.4.2 Localized Corrosion of the Drip Shield

5.1.3.1.4.2.1 Model Integration

Localized corrosion of titanium alloys is assumed to occur when the E_{corr} is greater than the E_{crit} (Bechtel SAIC Company, LLC, 2003b). The value of E_{crit} is determined in simulated ground water solutions using cyclic potentiodynamic polarization tests. Only crevice corrosion is considered; pitting corrosion is disregarded as a plausible degradation process because it is not observed in studies at the long-term corrosion test facility. Initiation and threshold potentials were obtained in cyclic potentiodynamic polarization tests in a variety of electrolytes based on modifications of J–13 Well water at temperatures up to 120 °C [248 °F].

For chloride concentrations up to 4 M NaCl for a pH range 2–14 and temperatures up to 107 °C [225 °F], the difference between E_{crit} and E_{corr} was sufficiently large to preclude the occurrence of crevice corrosion (Bechtel SAIC Company, LLC, 2003b). The difference between E_{corr} and E_{crit} is not strongly dependent on either temperature or chloride concentration. At temperatures greater than 107 °C [225 °F] in concentrated chloride solutions, the E_{crit} is likely to be greater than E_{corr}. No localized corrosion was observed in additional tests conducted in concentrated $CaCl_2$ solutions at 150 °C [302 °F]. Results from tests conducted in concentrated $CaCl_2$ solutions indicate that the difference between E_{corr} and E_{crit} is in the range of 1.5 to 3.5 V.

The localized corrosion model for the drip shield includes consideration of environmental factors such as chloride concentration and temperature. The model does not consider the effect of fluoride, which is known to promote generalized passivity breakdown; however, the combination of chloride and fluoride has not been shown to increase the localized corrosion susceptibility of Titanium Grade 7 (Brossia and Cragnolino, 2001a,b). Effects of fabrication processes (e.g., welding and postweld treatments) are not considered in the localized corrosion model. Results from Brossia and Cragnolino (2001b) suggest that fabrication processes such as welding do not significantly alter the localized corrosion resistance of Titanium Grade 7.

Overall, the available information is sufficient to expect that the information necessary to assess the localized corrosion of the drip shield with respect to system description and model integration will be available at the time of a potential license application.

5.1.3.1.4.2.2 Data and Model Justification

Cyclic potentiodynamic polarization experiments were performed to examine localized corrosion susceptibility. Based on experiments conducted with Titanium Grade 7 in simulated saturated water at 120 °C [248 °F] and in simulated concentrated water at 90 °C [164 °F] (the nominal compositions for these solutions are shown in Table 5.1.3.1-1), 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, 2000h). Crevice corrosion tests lasting 8 weeks were conducted in basic saturated water at temperatures ranging from 60 to 105 °C [140 to 221 °F]; no crevice corrosion was observed (Hua, et al., 2002).

The model parameters that affect the localized corrosion susceptibility of Titanium Grade 7 are based on supporting data that consider the range of expected environmental conditions within the emplacement drifts of the potential repository. The crevice corrosion susceptibility of

Titanium Grade 7 has been evaluated at temperatures up to 150 °C [302 °F] in solutions that encompass the range of representative water chemistries that may contact the drip shields.

Although the effects of fabrication processes such as welding and postweld heat treatments have not been reported, results from Brossia and Cragnolino (2001b) suggest that the fabrication processes do not significantly alter the localized corrosion resistance of Titanium Grade 7. Similar tests have not been performed using Titanium Grade 24; however, the resistance of titanium alloys to localized corrosion in chloride-containing solutions is well known and similar alloys are frequently used in chloride solutions at elevated temperatures in heat exchangers and seawater desalination plants.

Overall, the available information is sufficient to expect that the information necessary to assess the localized corrosion of the drip shield with respect to data being sufficient for model justification will be available at the time of a potential license application.

5.1.3.1.4.2.3 Data Uncertainty

The determination of the localized corrosion susceptibility of Titanium Grade 7 is based on a comparison of the critical potentials for crevice corrosion (E_{crit}) and corrosion potentials (E_{corr}). The difference between the E_{crit} and the E_{corr} is dependent on temperature, chloride concentration, and pH. Because no localized corrosion of titanium was observed in the cyclic polarization tests, the values of E_{crit} are conservatively based on the initiation of another electrochemical reaction at high anodic potentials (i.e., oxygen evolution). Actual values for the crevice corrosion repassivation potentials reported by Brossia and Cragnolino (2001b) for Titanium Grade 7 were approximately 1.4 V versus a saturated calomel electrode (V_{SCE}) in 1 M NaCl at 165 °C [329 °F] and greater than 5 V_{SCE} in a 5 M NaCl solution at 95 °C [203 °F]. In addition to temperature, other factors can influence the E_{corr}, such as radiolysis and water chemistry, which are not considered in the assessment of E_{corr}. It is unlikely, however, that uncertainty in the values of E_{corr} are significant in the assessment of localized corrosion susceptibility of Titanium Grade 7 as a result of the well-established resistance to crevice attack in chloride-containing solutions. Effects of fabrication processes have not been reported. Processes such as welding and postweld heat treatments may alter the localized corrosion susceptibility of titanium-palladium alloys; however, the resistance to localized corrosion in the expected range of potential repository conditions is expected to remain high (Brossia and Cragnolino, 2001b). The localized corrosion resistance of Titanium Grade 24 is expected to be similar to that of Titanium Grade 7.

Overall, the available information is sufficient to expect that the information necessary to assess the localized corrosion of the drip shield with respect to data uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.1.4.2.4 Model Uncertainty

No alternate conceptual model for the initiation of localized corrosion of the drip shield is considered. The evaluation of localized corrosion resistance of Titanium Grade 7 based on the potential difference between E_{crit} and E_{corr} in solutions with chloride concentrations up to 4 M NaCl for a pH range 2–14, and temperatures up to 107 °C [225 °F], indicates that crevice corrosion will not be initiated. The selection of E_{crit} is conservative because it is not based on the initiation of localized corrosion but on another electrochemical reaction, (i.e., the evolution of oxygen) observed at high anodic potentials. Additional tests in concentrated $CaCl_2$ solutions support the DOE model that predicts the titanium drip shield will be resistant to localized corrosion. The minimum difference between the E_{crit} and E_{corr} is approximately 800 mV (Bechtel SAIC Company, LLC, 2003b). Considering a large confidence interval of four standard deviations, the minimum difference between the E_{crit} and E_{corr} is approximately 400 mV.

Effects of fabrication processes have not been reported. Although welding and postweld heat treatments may alter the localized corrosion susceptibility of titanium-palladium alloys, results for Titanium Grade 7 suggest that fabrication processes do not result in a significant increase in localized corrosion susceptibility (Brossia and Cragnolino, 2001b). The localized corrosion resistance of Titanium Grade 24 is expected to be similar to that of Titanium Grade 7.

In summary, the DOE drip shield localized corrosion model appears conservative. No alternative conceptual model for localized corrosion of the drip shield was considered. Effects of fabrication processes and alloy composition have not been evaluated, however, the large confidence interval is expected to sufficiently bound model uncertainty for the drip shield materials.

Overall, the available information is sufficient to expect that the information necessary to assess the localized corrosion of the drip shield with respect to model uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.1.4.2.5 Model Support

The DOE model considers that relevant environmental parameters necessary to evaluate the localized corrosion susceptibility of the drip shield and is consistent with previous investigations for similar alloys. Similar investigations have shown that Titanium Grade 7 has a high crevice corrosion resistance in aggressive solutions (Brossia and Cragnolino, 2001a,b). Conditions where localized corrosion has been observed are limited to high potentials that are not obtainable under natural conditions (Brossia and Cragnolino, 2001b). Fabrication processes such as welding have not been shown to significantly decrease the crevice corrosion repassivation potentials.

Overall, the available information is sufficient to expect that the information necessary to assess the localized corrosion of the drip shield with respect to model abstraction output being supported by objective comparisons will be available at the time of a potential license application.

5.1.3.1.4.3 Environmentally Assisted Cracking of the Drip Shield

5.1.3.1.4.3.1 Model Integration

Environmentally assisted cracking was examined by DOE (Bechtel SAIC Company, LLC, 2003b) considering two main processes: stress corrosion cracking and hydrogen-induced cracking. The process model report, corresponding analysis and model reports, and other technical documents (CRWMS M&O, 2000a,i,j,k) made a clear distinction between stress corrosion cracking and hydrogen-induced cracking. It is stated by DOE that the potential repository environment is unique—presumably in comparison to other industrial and nuclear facilities—because, in the absence of disruptive events, there is no source of dynamic mechanical loading. Within this framework, the only viable source of stress needed for stress corrosion cracking to occur results from accumulated rockfall, because it is stated 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. However, DOE (Bechtel SAIC Company, LLC, 2003b) concluded that tight cracks produced by stress corrosion cracking will be plugged by scale deposits that will seal the cracks in approximately 3,400 years, impeding the flow of water and its contact with the waste package surface.

The primary environmentally assisted cracking mode for the drip shield is hydrogen absorption leading to embrittlement as a result of hydrogen-induced cracking. The approach taken by DOE to evaluate hydrogen-induced cracking is based on the assumption 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 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. The critical hydrogen concentrations for Titanium Grades 7 and 24 were estimated to be 1,000 and 400–600 ppm, respectively (Bechtel SAIC Company, LLC, 2003b). 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 hydrogen-induced cracking does not have a significant effect on the drip shield life expectancy for more than 10,000 years.

DOE has considered the two predominant forms of environmentally assisted cracking of titanium alloys, stress corrosion cracking and hydrogen-induced cracking, and the main factors affecting these two processes. However, the argument regarding the plugging of cracks has a limited technical basis and lack of experimental evidence. DOE has indicated it will provide the necessary justification in its response to agreement TSPAI.3.03 (Reamer, 2001).

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 by promoting accelerated corrosion and hydrogen entry into the alloy. To address these concerns, DOE agreed (Schlueter, 2000) to provide additional information on credible environmental conditions including the composition of the contacting water (e.g., fluoride content).

Additional examination of possible galvanic interactions with iron-based components in the potential 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 so that the concentration would be low. In any event, the consequence for both stress corrosion cracking and hydrogen-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.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.1.5), is sufficient to expect that the information necessary to assess environmentally assisted cracking of the drip shield with respect to system description and model integration will be available at the time of a potential license application.

5.1.3.1.4.3.2 Data and Model Justification

No stress corrosion cracking failure was observed by DOE in constant deflection tests using U-bend specimens of both Titanium Grades 7 and 12 in the long-term corrosion testing facility after a 4-year exposure to simulated dilute water, simulated concentrated water, and simulated acidified water at 90 °C [194 °F] (Gordon, 2002). In constant load tests conducted in a variation of simulated concentrated water at 105 °C [221 °F], however, stress corrosion cracking of Titanium Grade 7 specimens stressed at near the tensile strength occurred in a short period (ranging from a few days to a few months). Welded Titanium Grade 12 specimens were also observed to suffer stress corrosion cracking in simulated concentrated water at 90 °C [194 °F] (Fix, et al., 2004). Using precracked compact tension specimens exposed to an air-saturated alkaline solution (pH 13.4) with a composition similar to basic saturated water at 110 °C [230 °F], stress corrosion crack growth rates ranging from 7.9×10^{-8} to 4.0×10^{-8} mm/s [3.1×10^{-6} to 1.6×10^{-6} mils/s] were measured by Andresen, et al. (2001) at $K_I = 30$ MPa·m$^{1/2}$ [27.3 ksi·in$^{1/2}$] under cyclic loading. The crack growth rate decreased only slightly to 1.3×10^{-8} mm/s [5.1×10^{-7} mils/s] under constant load after the initiation under cyclic loading. No experimental work has been conducted by the DOE to examine the stress corrosion cracking susceptibility of Titanium Grade 24. Since titanium cracking susceptibility is generally related to the strength of the alloy, Titanium Grade 24 could be more susceptible to stress corrosion cracking than either Titanium Grades 7 or 12. Therefore, the susceptibility of Titanium Grade 24 to stress corrosion cracking needs to be evaluated.

The critical hydrogen concentrations for Titanium Grades 7 and 24 were estimated to be 1,000 and 400–600 ppm, respectively (Bechtel SAIC Company, LLC, 2003b). The selection of these critical hydrogen concentration values is justified, based on the literature data for titanium alloys and the effects of palladium additions to titanium alloys. The critical hydrogen concentration is determined using compact tension specimens under slow strain rate conditions and it could be lower if slow crack growth occurs. Slow crack growth requires relatively high stress intensities to occur {> 50 MPa·m$^{1/2}$ [45.5 ksi·in$^{1/2}$]}. However, the possible increase of hydrogen uptake by Titanium Grade 7 in the presence of a large amount of fluoride has not been evaluated, leading to the possibility of enhanced susceptibility to hydrogen-induced cracking.

To address these concerns, DOE agreed (Schlueter, 2000) to provide additional information on environmentally assisted cracking of titanium alloys as well as credible environmental conditions, including composition of the contacting water. In particular, DOE agreed to address the potential detrimental effect of fluoride anions leading to accelerated drip shield dissolution and concurrent hydrogen uptake, and possibly hydride cracking.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.1.5), is sufficient to expect that the information necessary to assess environmentally assisted cracking of the drip shield with respect to data being sufficient for model justification will be available at the time of a potential license application.

5.1.3.1.4.3.3 Data Uncertainty

Stress corrosion cracking, due to residual stresses arising from fabrication processes and applied stresses from accumulated rockfall as a result of drift degradation, is a possible failure mode for the drip shield. DOE has reported the occurrence of stress corrosion cracking of Titanium Grade 7 in tests conducted in basic saturated water and simulated concentrated water (Andresen, et al., 2001; Fix, et al., 2004). However, DOE considers stress corrosion cracking of the drip shield as having a low consequence because of presumed crack plugging by corrosion deposits. This claim needs to be evaluated further, because it is unclear how corrosion product buildup will occur such that any cracks that developed are plugged with corrosion products. The DOE approach has neither extensively evaluated the effect of accumulated rockfall or cyclic stress on stress corrosion cracking of titanium alloys nor considered the consequence of the crack presence on subsequent rockfall events where an existing crack acts as the nucleation point for a substantial opening in the drip shield. DOE has agreed (Schlueter, 2000) to provide additional information on evaluation of the effect of rockfall or potential repository on environmentally assisted cracking of drip shield.

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 such cases, the propagation of data uncertainty can affect evaluation of the potential occurrence of delayed hydrogen cracking as a coupled failure mode. Error propagation from data uncertainties that originate from possible acceleration of uniform corrosion and hydride embrittlement in the presence of fluoride ions was considered. To address this concern, DOE agreed (Schlueter, 2000) to provide additional information on the fluoride concentration of the ground water in contact with drip shields and its effects on accelerated drip shield corrosion and hydrogen uptake/hydride cracking, which could be affected by the availability and rate of supply of fluoride.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.1.5), is sufficient to expect that the information necessary to assess environmentally assisted cracking of the drip shield with respect to data uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.1.4.3.4 Model Uncertainty

DOE has considered two alternative models for stress corrosion cracking of titanium alloys and a model for hydrogen-induced cracking. The issue of drip shield failure by environmentally assisted cracking is important in relation to the mechanical disruption of the engineered barriers integrated subissue as described in Section 5.1.3.2. The effect of accumulated rockfall on mechanical failure of the drip shield will be affected by consideration of the drip shield wall thinning as a result of uniform corrosion and simultaneous hydrogen absorption leading to hydride precipitation and embrittlement of titanium alloys. The assumption of crack plugging to dismiss the possibility of stress corrosion cracking needs to be adequately justified. To address this concern, DOE agreed (Schlueter, 2000) to provide additional information on the effect of wall thinning from corrosion and hydride precipitation on the mechanical failure induced by accumulated rockfall.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.1.5), is sufficient to expect that the information necessary to assess environmentally assisted cracking of the drip shield with respect to model uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.1.4.3.5 Model Support

Even though environmentally assisted cracking of titanium-palladium alloys has not been extensively examined, recent data indicate that Titanium Grade 7 exhibits crack growth in basic saturated water and simulated concentrated water. Although 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, a better understanding for the environmental and mechanical conditions leading to stress corrosion cracking in simulated waters is needed. DOE, however, considers stress corrosion cracking and hydrogen-induced cracking to be separate mechanisms. In fact, DOE is considering two possible models for stress corrosion cracking (stress intensity threshold and slip/film rupture dissolution). It is unclear how these stress corrosion cracking models fit into the more 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.

Additional technical bases have been provided for the fraction of hydrogen absorbed by titanium during corrosion processes (CRWMS M&O, 2000l). 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. DOE should examine the possibility of enhanced hydrogen uptake and absorption in the palladium-bearing titanium alloys, especially Titanium Grade 7 rather than Titanium Grade 16, because the differences in the palladium content of these materials could make a difference in the measured hydrogen uptake rates. The lower threshold value of Titanium Grade 24 (or Titanium Grade 5) is expected to be fully justified in the revised version of CRWMS M&O (2000j). The possibility of enhanced hydrogen uptake in the presence of fluoride through destabilization of the TiO_2 oxide should be evaluated also.

The belief that stress corrosion cracking and hydrogen-induced cracking of the drip shield have low consequences because of presumed crack plugging by corrosion or calciferous deposits should be evaluated further. Although it may be possible that any cracks forming on the drip shield eventually will be plugged so that no water transport through the crack is possible, the consequence of subsequent rockfall events on the crack and any plugging material 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. The DOE assessment of the environmentally assisted cracking of drip shields is unclear regarding the hydrogen uptake process, and the proposed mechanism for crack plugging by corrosion or calciferous deposits as a means for crack arrest.

To address these concerns, DOE agreed (Schlueter, 2000) to provide additional information on rationales for the possibility of drip shield stress corrosion cracking and for the efficiency of hydrogen uptake, as well as the potential effects of crack plugging by corrosion or by calciferous deposits on the further development of stress corrosion cracking.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.1.5), is sufficient to expect that the information necessary to assess environmentally assisted cracking of the drip shield with respect to model abstraction output being supported by objective comparisons will be available at the time of a potential license application.

5.1.3.1.4.4 Dry-Air Oxidation and Humid-Air Corrosion of the Waste Package

5.1.3.1.4.4.1 Model Integration

For undisturbed repository conditions, corrosion is expected to be the primary degradation process limiting the life of Alloy 22 engineering barriers. Dry-air oxidation is assumed to occur during the initial period after waste package emplacement when the radioactive decay heat keeps moisture away from the gaseous environment surrounding the waste package (Bechtel SAIC Company, LLC, 2003b,c; CRWMS M&O, 2000d,g).

The expected in-drift environment including typical waste package temperature and humidity histories has been reported by DOE (Bechtel SAIC Company, LLC, 2003b,c). Dry-air oxidation of the Alloy 22 engineering barrier is expected to occur when the relative humidity of the repository is less than the critical relative humidity for the initiation of humid-air corrosion. If no drift degradation occurs prior to the thermal pulse, the maximum waste package temperature is expected to be below 200 °C [392 °F] (Bechtel SAIC Company, LLC, 2003b). The rate of dry-air oxidation is modeled assuming that mass transport of reacting species is limited by solid-state diffusion through a tightly adherent passive oxide film. The predominant oxide is a uniform, protective Cr_2O_3 film. Small amounts of other elements may also be present in the oxide. The growth of the oxide film follows a parabolic rate law and the oxide film thickness at any specific time is proportional to the square root of time (Fehlner, 1986; Welsch, et al., 1996).

Another mode of oxidation degradation of alloys in dry-air environment at elevated temperatures is internal oxidation. Oxygen may diffuse inward and form internal oxides in the alloy matrix or form internal precipitates along grain boundaries. Formation of internal oxides along grain boundaries, also known as intergranular oxidation, has been reported for Fe-21Cr-32Ni alloy after oxidation at 900 °C [1,652 °F] for 3,000 hours (Ahn, 1996; Shida and Moroishi, 1992). However, neither internal oxidation or intergranular oxidation is likely to be significant at the proposed operating conditions of the potential repository. Dry-air oxidation is not a performance limiting process of the waste package Alloy 22 material and is not considered in the waste package performance analysis for the potential repository.

For humid-air corrosion, DOE assumes that an aqueous solution exists on the waste package surfaces when the relative humidity is higher than the critical relative humidity. Humid air corrosion is characterized by general corrosion under a thin film of liquid. At a given temperature, the existence of liquid water on the waste package surface depends on the hygroscopic nature of any salts or minerals deposited on the surface. In the presence of such a deposit, a liquid-phase surface brine film can be established by deliquesced salts with water from the atmosphere and form at a higher temperature and lower RH than otherwise possible. The critical relative humidity is based on the mutual deliquescence point, that is the lowest relative humidity at which a saturated solution of the salt mixture can be maintained at a given temperature. The humid air corrosion rate is temperature-dependent, and time-independent at a given temperature (Bechtel SAIC Company, LLC, 2003b,c; CRWMS M&O, 2000d).

Dry air oxidation during the dry out period is not a life limiting factor in waste package performance. Humid air corrosion is one form of general corrosion that is bounded by slow rates of aqueous corrosion, and is also not a life limiting factor for waste package performance.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.1.5), is sufficient to expect that the information necessary to assess the dry-air oxidation and humid-air corrosion of the waste package with respect to system description and model integration will be available at the time of a potential license application.

5.1.3.1.4.4.2 Data and Model Justification

DOE models indicate that the highest waste package temperature is expected to be less than 200 °C [392 °F], which is well below 350 °C [662 °F], the boundary temperature used in the calculation of oxidation rate. Using this parabolic rate constant, the oxide film thickness in the first year on the Alloy 22 surfaces was estimated to be approximately 9.3 nm [3.7×10^{-4} mils]. Measurements of Alloy 22 oxide thickness using the atomic force microscope result in a limiting oxide thickness of approximately 3.4 nm [1.3×10^{-4} mils] after a 7-month exposure to air at 200 °C [392 °F] (Gordon, 2002). These results suggest that Alloy 22 surfaces in the potential repository will undergo dry-air oxidation during the high-temperature dry-out period. However, the oxidation rate is low at the waste package temperatures predicted by DOE after waste emplacement, and dry-air oxidation does not limit waste package lifetime.

DOE has measured corrosion rates in the long-term corrosion test facility at Lawrence Livermore National Laboratory for 5 years using weight loss experiments (Bechtel SAIC Company, LLC, 2003b,c). DOE has performed long-term weight loss experiments using a variety of samples of diverse geometry, under two temperatures {60 °C [140 °F] and 95 °C [203 °F]}, and using solutions with various concentrations of halides (e.g., chloride) and oxyanions such as nitrate. Approximately half the samples are submersed and half are in the saturated vapor representative of humid-air corrosion. The weight loss method yields humid-air corrosion rates on the order of 5 nm/yr [1.96×10^{-4} mpy].

The thinning of the waste package surface by humid air oxidation is equal to the general corrosion rate at the temperature multiplied by the time the waste package surface is at that temperature. Based on measured corrosion rate, the material loss due to humid air oxidation is estimated to be 50 μm [1.96 mils] during a 10,000-year period. The estimation is considered conservative because the corrosion rates of metals and alloys tend to decrease with time. This low material loss rate demonstrated that humid air oxidation if not a life limiting factor in determining waste package performance.

Overall, the available information is sufficient to expect that the information necessary to assess the dry-air oxidation and humid-air corrosion of the waste package with respect to data being sufficient for model justification will be available at the time of a potential license application.

5.1.3.1.4.4.3 Data Uncertainty

After the 5-year test at the long-term corrosion test facility, it was reported that the corrosion rates of Alloy 22 were generally lower for those specimens exposed to vapor than immersed in liquid, regardless of the test temperature or electrolyte solution. Moreover, for the weight loss coupons, there appears to be no effect of the presence of welds on the corrosion rate (Bechtel SAIC Company, LLC, 2003b,c). The results showed that the humid-air corrosion rate is bounded by the general corrosion rate in electrolyte solution. The uncertainties in the general corrosion rate of Alloy 22 resulted from insufficient resolution of the weight-loss measurement because of the extremely low corrosion rate of Alloy 22. For the weight loss samples, approximately 89 percent of the variation in the measured corrosion was due to variation among specimens, and the rest from measurement uncertainty. The combined standard uncertainty is estimated to be 0.314 nm/yr [1.24×10^{-5} mpy].

No corrosion rate data are available for Alloy 22 beyond 5 years. However, a slow general corrosion rate has been observed for similar nickel-chromium-molybdenum type alloys. Recent results show that a similar alloy, Alloy C, maintains a mirror-like finish and passive film general corrosion behavior after 50 years of exposure at Kure Beach, North Carolina (McCright, 1998). During the more than 50 years of exposure, the specimens have been subjected to a range of ambient temperatures, humidities, and alternate wetting and drying cycles.

Overall, the available information is sufficient to expect that the information necessary to assess the dry-air oxidation and humid-air corrosion of the waste package with respect to data uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.1.4.4.4 Model Uncertainty

For dry-air oxidation, the oxide growth rate follows a parabolic rate law. It is reasonable to expect an Arrhenius relationship between temperature and oxidation rates. Although there is uncertainty in activation energies, DOE indicates that the rate of dry-air oxidation is too low to cause waste package failure during the dryout period.

For humid-air corrosion, the maximum corrosion rate is limited to the uniform corrosion rate under aqueous conditions. The experimental corrosion rates in the long-term corrosion test facility show a steady decrease with time. Therefore, DOE considers that the selection of the 5-year data to define corrosion rate distributions is conservative. It is reasonable to expect an Arrhenius relationship between temperature and passive corrosion rates. Although there is uncertainty in activation energies, DOE indicates that the rate of humid-air corrosion is too low to be a life limiting factor for the waste package.

Environmental thermogravimetric analysis has been used by DOE to evaluate the corrosion of Alloy 22 underneath deliquescence of deposited $CaCl_2$ at 150 °C [302 °F] and 22.5-percent relative humidity. No sustained oxidation of Alloy 22 is evident from the thermogravimetric analysis data (Bechtel SAIC Company, LLC, 2003b,c). However, no convincing conclusions can be drawn from this test because $CaCl_2$ is decomposed at 150 °C [302 °F] in less than 10 hours under flowing air conditions.

Overall, the available information is sufficient to expect that the information necessary to assess the dry-air oxidation and humid air corrosion of the waste package with respect to model uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.1.4.4.5 Model Support

DOE has stated that during the preclosure period, the waste package will be kept dry by air ventilation. For the dryout period after closure of the potential repository, also known as the thermal pulse period, temperatures within the repository drifts would be less than 200 °C [392 °F]. Dry-air oxidation at these temperatures will not be a limiting factor for waste package life.

After peak temperature is reached, the waste package will begin to cool, resulting in increased relative humidity. Humid-air corrosion alone will not be a limiting factor for waste package life. However, coupling of relative humidity with deliquescent brine may result in an increased corrosion rate. Tests by DOE showed that no localized corrosion of Alloy 22 has been observed beneath films of $CaCl_2$ deliquescent brines at 150 °C [302 °F] and 22.5-percent relative humidity (Bechtel SAIC Company, LLC, 2003b,c).

Overall, the available information is sufficient to expect that the information necessary to assess the dry-air oxidation and humid-air corrosion of the waste package with respect to model abstraction output being supported by objective comparisons will be available at the time of a potential license application.

5.1.3.1.4.5 Passivity and Uniform Corrosion of the Waste Package

5.1.3.1.4.5.1 Model Integration

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_{crit}). No mechanistic model is used to calculate corrosion rates within this regime. An empirical model for the general corrosion rates was derived from weight loss data obtained from the long-term corrosion test facility where numerous Alloy 22 test specimens have been exposed to aqueous solutions based on modifications of J–13 Well water (Bechtel SAIC Company, LLC, 2003c; McCright, 1998). The corrosion rate decreased with time and, after 5 years, the corrosion rates were less than 23 nm/yr [9.0×10^{-4} mpy] in all test solutions at 60 and 90 °C [140 and 194 °F]. The mean general corrosion rate for the model abstraction is 7.3 nm/yr [2.9×10^{-4} mpy] with a standard deviation of 5.0 nm/yr [2.0×10^{-4} mpy], based on the 5-year data for the crevice specimens. Although the corrosion rates measured by weight loss in the long-term corrosion test facility were independent of temperature, the effect of temperature is considered using an activation energy obtained from electrochemical tests. For Alloy 22, the apparent activation energy of 26 kJ mol^{-1} [6.2 kcal mol^{-1}] was derived from tests of a range of metallurgical conditions, temperatures, and solution chemistries (Bechtel SAIC Company, LLC, 2003b,c).

Acceleration of the corrosion rate as a result of microbial activity is treated using a microbially influenced corrosion factor, G_{MIC}, that has a uniform distribution from 1 to 2. The condition for the occurrence of microbially influenced corrosion is a threshold relative humidity of 90 percent.

The current DOE general corrosion model abstraction discounts the thermal aging effect on general corrosion based on limited test results in 5 M $CaCl_2$ and 5 M $CaCl_2$ + 0.5 M $Ca(NO_3)_2$ solutions and the insignificant thermal aging effect under the anticipated repository conditions (Bechtel SAIC Company, LLC, 2003b). DOE should provide additional evaluations to justify the exclusion of the enhancement factor accounting for the thermal aging effect.

DOE addressed the issue of long-term passive film stability through a limited number of short-term electrochemical tests and studies of the structure and composition of the Alloy 22 passive film using x-ray photoelectron spectroscopy (Bechtel SAIC Company, LLC, 2003b, 2001). A modeling effort based on the point defect model for passivity is under way. Results from tests, either to evaluate long-term effects on corrosion rates under passive dissolution or taking into consideration the formation of aggressive (concentrated salts with low pH) solutions on the waste package surface that may accelerate the uniform corrosion rate, have not been fully reported. Limited study of a sample of Josephinite (a rock containing a naturally occurring nickel-iron alloy) was conducted to provide technical support to the long-term stability of passive films through the use of a natural analog. DOE agreed (Schlueter, 2000) to provide the technical basis that supports the long-term passive film stability.

While some information on the passivity and uniform corrosion of the waste package with respect to model integration may be available at the time of a potential license application, the staff is currently of the view that DOE should provide additional information on the effect of fabrication processes, and support for the calculated passive corrosion rates over the entire temperature range of intended use.

5.1.3.1.4.5.2 Data and Model Justification

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 (ASTM International, 1999a) after exposures of 0.5, 1, 2, and 5 years. Weight gain was observed on up to 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. From the 5-year data for the crevice specimens, the abstracted general corrosion rate for the Alloy 22 outer container was found to be 7.3 ± 5.0 nm/yr [$2.9 \times 10^{-4} \pm 2.0 \times 10^{-4}$ mpy] (Bechtel SAIC Company, LLC, 2003b,c). Specimens exposed for 5 years had corrosion rates less than 23 nm/yr [9.0×10^{-4} mpy]. The decrease in the corrosion rate as a function of time is supported by electrochemical measurements. Corrosion rates of Alloy 22 in simulated acidified water measured using polarization resistance decrease by a factor of 10 after a 1-week exposure. Decreasing corrosion rates also were observed in potentiostatic measurements (Lian, et al., 2003).

The lack of an observed temperature dependence for the corrosion rates of specimens exposed in the long-term corrosion test facility was addressed by obtaining the temperature dependence of the corrosion rates measured using the linear polarization or polarization resistance method. For steady-state measurements, the activation energy for uniform corrosion was determined to be 26 ± 3 kJ mol^{-1} [6.2 ± 0.7 kcal mol^{-1}] (Bechtel SAIC Company, LLC, 2003b,c). The base corrosion rates were measured at temperatures less than 100 °C [212 °F]. The presence of a water film sufficient to support corrosion processes may be possible at higher temperatures with the formation of salts with a low deliquescence relative humidity. Corrosion rates in concentrated solutions formed by the deliquescence of deposited salts that may disrupt passivity have not been evaluated by DOE.

Welding and fabrication processes typically decrease the localized corrosion resistance of passive chromium containing alloys. Increased corrosion rates were reported by Rebak, et al. (2002) and Summers, et al. (2002, 2000) for welded and aged Alloy 22 using standardized tests designed to detect intergranular corrosion sensitivity. Although fabrication processes can be expected to reduce the localized corrosion resistance of Alloy 22 (Heubner, et al., 1989), no effect of fabrication processes on localized corrosion resistance was observed in solutions with high chloride concentrations. Waste package fabrication processes are not considered in the general corrosion rate model (Bechtel SAIC Company, LLC, 2003b).

The DOE general corrosion rate model assumes that the material remains passive under the various testing conditions (Bechtel SAIC Company, LLC, 2003c). Aggressive environments characterized by high chloride concentrations, low pH, and high temperatures, however, can disrupt passivity of Alloy 22 and cause accelerated uniform corrosion. The range of environments in contact with the waste package is not adequately justified. DOE also recognizes the need of some fundamental understanding of corrosion processes of passive film to extrapolate to long times. Additionally, the calculated passive corrosion rates have not been verified over the entire temperature range of intended use.

Although the data and model abstraction for the passivity and uniform corrosion of the waste package seem adequate, there are certain aspects related to conditions of the tests and the influence of the passivity and uniform corrosion that require additional explanation. The extrapolation of uniform corrosion rates to extended periods should be supported with modeling of passive film stability. Calculated corrosion rates at elevated temperatures should also be verified with measured corrosion rates. To address these concerns, DOE agreed (Schlueter, 2000) to provide the additional information on the test conditions.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.1.5), is sufficient to expect that the information necessary to assess the passivity and uniform corrosion of the waste package with respect to data being sufficient for model integration will be available at the time of a potential license application.

5.1.3.1.4.5.3 Data Uncertainty

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 also is 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] at the 50th percentile (CRWMS M&O, 2000a) 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 1.25×10^{-3} g/yr [4.9×10^{-5} 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 no 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. In addition, the corrosion data indicated that surface preparation of the test specimens may alter corrosion rates. Data from specimens with nonstandard preparation were used selectively and the uncertainty associated with this effect was not characterized. Furthermore, the highest corrosion rates measured, if not accounted for in the distribution, would lead to container failure times much shorter than those currently estimated in the total system performance assessment for the site recommendation.

Higher corrosion rates have been observed for nickel-chromium-molybdenum alloys at elevated temperatures in concentrated solutions [Smailos (1993), Bickford and Corbett (1985), Harrar, et al. (1978, 1977)]. For example, Smailos (1993) reported corrosion rates of Alloy C-4 in brine environments containing 25.9-percent sodium chloride at 150 °C [302 °F] 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]. Bickford and Corbett (1985) reported corrosion rates of Alloy 22 to be 0.05 mm/yr [2 mpy] at 40 °C [104 °F] and 0.012 mm/yr [0.47 mpy] at 90 °C [194 °F] in near neutral environments containing chloride, fluoride and sulfate. Harrar, et al. (1978, 1977) estimated general corrosion rates of 1.5×10^{-3} mm/yr [5.9×10^{-2} mpy] for Alloy C-276 in the Salton Sea geothermal field ground water at 100 °C [212 °F]. In summary, the distribution of corrosion rates used by 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 conditions.

The relative corrosion rates of welded and base metal Alloy 22 also were 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 (Schlueter, 2000) to use a larger surface area in corrosion testing, including welded samples cut from mockups. Similarly, the corrosion rates of crevice specimens (welded and mill-annealed) were significantly greater than the corrosion rates for the weight loss specimens, even though crevice corrosion was not observed. The higher corrosion rates for the crevice specimens may indicate that the uniform corrosion rates under crevices are greater than the corrosion rates in freely exposed conditions. The analysis of the weight loss data does not consider the accelerated corrosion that may have occurred in the crevice regions.

Although the uncertainty in the corrosion rates obtained by weight loss appears to be insufficiently characterized, the corrosion rate is low for conditions where the passive film on Alloy 22 is stable, and long waste package lifetimes are expected. The presence of a crevice and welds is not expected to decrease waste package lifetimes significantly as long as passivity is maintained. Short-term corrosion rates measured using electrochemical methods under nonsteady-state conditions are approximately 100 nm/yr [3.9×10^{-3} mpy] and significantly greater than the long-term corrosion rates (Rebak, et al., 2002). Nevertheless, currently available information indicates that if passivity is maintained, the Alloy 22 outer container will not fail within 10,000 years.

Corrosion rates calculated from weight loss measurements were not sensitive to the effect of temperature. The DOE evaluation of the temperature dependence for the uniform corrosion of Alloy 22 was obtained using polarization resistance measurements. Actual corrosion rates obtained using linear polarization were not used in the assessment of the lifetime of the Alloy 22 outer containers. Instead, the baseline corrosion rates from the long-term corrosion test facility weight loss specimens were combined with the uniform corrosion rate activation energy obtained using electrochemical measurements. Although the corrosion rates obtained using electrochemical tests are greater than the corrosion rates measured by the weight loss method, the corrosion rates under passive conditions are low and decrease with time. Measured passive corrosion rates are limited to temperatures less than 100 °C [212 °F]. After emplacement and closure of the potential repository, the surface temperature of the waste packages is expected to be near 160 °C [320 °F]. Uncertainties in the corrosion rates at higher temperatures are not fully evaluated by the DOE uniform corrosion model. Extrapolation of passive corrosion rates to higher temperatures may not be representative of the actual waste package corrosion rates at temperatures greater than 100 °C [212 °F]. Although higher temperatures and concentrated chloride solutions formed by evaporation and concentration of dissolved species are not expected to prevail in the potential repository, if such conditions occur they may disrupt the passive film and lead to significantly higher corrosion rates and shorter waste package lifetimes.

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 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.

Although the data and model abstraction for the passivity and uniform corrosion of the waste package seem adequate, there are certain aspects related to conditions of the tests and the influence of the passivity and uniform corrosion that require additional explanation including passive film stability and corrosion rates in concentrated solutions with elevated boiling temperatures. To address these concerns, DOE agreed (Schlueter, 2000) to provide the additional information on the test conditions.

While some information on the passivity and uniform corrosion of the waste package with respect to data uncertainty may be available at the time of a potential license application, the staff is currently of the view that DOE should provide additional information on the DOE use of corrosion rates in concentrated solutions that may disrupt passivity, and support for calculated passive corrosion rates over the entire temperature range of intended use.

5.1.3.1.4.5.4 Model Uncertainty

The DOE alternative conceptual model for general corrosion is based on the assumption of the time-dependent general corrosion behavior (Bechtel SAIC Company, LLC, 2003c). Because this alternative conceptual model considers the rate decreases with time, it is assumed to be less conservative than the base general corrosion model.

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 and passive corrosion rates obtained using electrochemical methods. The lower corrosion rates calculated from the weight loss of specimens exposed in the long-term corrosion test facility may be valid if the corrosion rates decrease with time or as a result of silica deposition during testing. Decrease in the corrosion rate as a function of time is supported by electrochemical measurements. Corrosion rates of

Alloy 22, measured using polarization resistance (Rebak, et al., 2002) and potentiostatic measurements (Lian, et al., 2003), were determined in short-term tests. Although corrosion rates obtained after short-term exposures may be higher, the increased corrosion rate is insufficient to cause failure within 10,000 years if passivity is maintained.

Corrosion rate data used by DOE do not reflect 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 alloying elements has not been obtained from the long-term corrosion test facility using weight loss specimens.

Overall, the available information is sufficient to expect that the information necessary to assess the passivity and uniform corrosion of the waste package with respect to model uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.1.4.5.5 Model Support

The DOE data for the corrosion rates of Alloy 22, obtained in the long-term corrosion test facility, are less definitive due to the deposition of silica and the limitations of the weight loss measurements to evaluate the effects of welding. 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, electrochemical methods such as steady-state anodic current density measurements obtained under potentiostatic conditions can be used to determine corrosion rates, according to ASTM International G102 (1999b). Limited electrochemical tests conducted by DOE showed the corrosion rates under passive conditions are sufficiently low so that failure within 10,000 years is not expected as a result of passive corrosion of the Alloy 22 outer container (Rebak, et al., 2002; Lian, et al., 2003). The activation energy for the uniform corrosion rate of Alloy 22 was obtained using electrochemical tests; the base corrosion rates are obtained from weight-loss measurements. The use of source data in the models appears to be inconsistent. The calculated passive corrosion rates for the general corrosion model, which assume passive film stability, have not been verified over the entire temperature range of intended use. Extrapolation of passive corrosion rates to environmental conditions where passivity cannot be maintained will significantly underestimate corrosion rates.

While some information on the passivity and uniform corrosion of the waste package with respect to model support may be available at the time of a potential license application, the staff is currently of the view that DOE should provide additional information on the DOE determination of corrosion rates in concentrated solutions that may disrupt passivity, and support for the use of calculated passive corrosion rates over the entire temperature range of intended use.

5.1.3.1.4.6 Localized Corrosion of the Waste Package

5.1.3.1.4.6.1 Model Integration

Localized corrosion of Alloy 22 is assumed to occur when the corrosion potential (E_{corr}) is greater than the critical potential (E_{crit}). The values of E_{corr} and E_{crit} were measured in laboratory tests using standard test procedures (Bechtel SAIC Company LLC, 2003b,c). Welded U-bend specimens and rod specimens used to evaluate E_{corr} of Alloy 22 as a function of immersion time were machined from sheet and bar stock. The E_{corr} of Alloy 22 was measured in simulated dilute water, simulated acidified water, simulated concentrated water at temperatures ranging from 25 to 90 °C [77 to 194 °F] for periods of up to 500 days. In additional tests, the E_{corr} of Alloy 22 was measured in 5 M $CaCl_2$ at 120 °C [248 °F] and in 1 M $CaCl_2$ with additions of $Ca(NO_3)_2$ at 90 °C [194 °F] for periods of up to 330 days. Values of the E_{crit} were obtained in simulated dilute water, simulated acidified water, simulated concentrated water, simulated saturated water, basic saturated water, NaCl solutions, and $CaCl_2$ solutions with and without the addition of $Ca(NO_3)_2$.

The E_{corr} dependence on pH is attributed to the influence of the hydrogen reduction reaction, especially in acidic solutions. Decreasing E_{corr} with increasing chloride concentration is attributed to the tendency of chloride to attack the passive film and lower oxygen solubility in concentrated solutions. The E_{corr} of Alloy 22, increases slightly with temperature. The increase in E_{corr} is apparent for the results obtained in simulated acidified water. For simulated concentrated water and simulated dilute water, the E_{corr} was not observed to increase with temperature, however, these data were limited to temperatures of 60 and 90 °C [140 to 194 °F]. Increased E_{corr} for Alloy 22 at elevated temperatures were attributed to the passive film becoming more defect free at higher temperature because the defect repair processes in the passive film could be accelerated at higher temperatures (Bechtel SAIC Company, LLC, 2003c).

The empirical model includes several relevant environmental parameters that influence the E_{corr}. The reported decrease in the E_{corr} as a result of increasing chloride concentration is expected as a result of the lower oxygen solubility in concentrated solutions. Conversely, the passivating effect of nitrate can increase the E_{corr}. Hydrogen ion reduction at low pH expected to increase E_{corr} as predicted by the empirical model. The increase in E_{corr} with temperature is not consistent with the expected increase in passive corrosion rates and decrease in oxygen solubility at elevated temperatures but should conservatively predict higher corrosion potentials at elevated temperature.

The empirical model was developed with data from fully immersed test specimens. Ground water contacting the waste packages will likely be spread over the cylindrical body of the outer container forming a thin water film. Increased oxygen reduction rates through thin water films can occur when the thickness of the water film is less than the diffusion layer thickness. The increased oxygen reduction rates through the thin solution layer may increase the E_{corr} by several hundred millivolts. In addition, the empirical model does not consider the effect of other oxidants such as ferric species formed from the corrosion of ground support materials. Small concentrations of such oxidants can increase the corrosion potential of the Alloy 22 outer barrier and promote localized corrosion initiation. However, with thin water films under open-circuit conditions, the decrease of cathodic throwing power will decrease the tendency for localized corrosion.

Three possible criteria to determine the value of E_{crit} from cyclic potentiodynamic polarization tests were evaluated (Bechtel SAIC Company, LLC, 2003b,c). The repassivation potential for crevice corrosion (E_{rcrev}) determined by the crossover of the forward and reverse scan was selected as the criteria for determining E_{crit}. Other methods evaluated were the breakdown potential of the passive film based on current density of 20 μA/cm² [1.9×10^{-2} A/ft²] and the repassivation of the surface based on a fixed current density of 1 μA/cm² [9.3×10^{-4} A/ft²]. Based on data obtained in NaCl solutions, and $CaCl_2$ solutions without the addition of $Ca(NO_3)_2$, the E_{rcrev} for Alloy 22 is determined to be a function of temperature, pH, and chloride concentration. The E_{rcrev} decreases with increasing chloride concentration and temperature. A weak dependence on pH was noted with increased E_{rcrev} values associated with increasing solution pH. The addition of nitrate significantly increases the value of the E_{rcrev} (Bechtel SAIC Company, LLC, 2003c).

The DOE empirical model for the initiation of crevice corrosion considers the environmental effects of temperature, chloride concentration, and nitrate concentration. Other anions such as reduced sulfur species may also promote localized corrosion of nickel-chromium-molybdenum alloys such as Alloy 22. Such species are not considered in the empirical model and are not expected under the oxidizing conditions of the potential repository without the presence of sulfate-reducing bacteria. In addition to nitrate, other anionic species such as carbonate, bicarbonate, and sulfate may act as inhibitors of crevice corrosion if present in sufficient concentrations with respect to chloride. The empirical model developed by DOE does not consider the inhibiting effects of other anions.

The localized corrosion propagation rate is assumed to be constant with time. The propagation rate is based on the localized penetration of Alloy 22 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]. The distribution is characterized with a mean penetration rate of 127 μm/yr [5 mpy], a minimum penetration rate of 12.7 μm/yr [0.5 mpy], and a maximum penetration rate of 1270 μm/yr [50 mpy] (Bechtel SAIC Company, LLC, 2003c).

The description of the waste package 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, 2000m). 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 expected in 10,000 years at 300 °C [572 °F] (Bechtel SAIC Company, LLC, 2003b). The formation of full grain boundary coverage of precipitates is deemed a worst-case scenario that would be equivalent to a 100-hour exposure at 700 °C [1,262 °F]. As noted in Bechtel SAIC Company, LLC (2001), the early stages of topologically close-packed phase precipitation on grain boundaries (i.e., 15-, 50-, and 80-percent grain boundary coverage) are also not expected at temperatures below 300 °C [572 °F] in 10,000 years. No long-range ordering is predicted if the temperature remains below 260 °C [500 °F].

The effects of container fabrication processes on the localized corrosion susceptibility of Alloy 22 was evaluated by comparing the E_{rcrev} for mill-annealed and as-welded Alloy 22 in 5 M $CaCl_2$ at 120 °C [248 °F]. The E_{rcrev} for the mill-annealed material was in the range −154 to −227 $mV_{Ag/AgCl}$ whereas the E_{rcrev} for the as-welded material was in the range −165 to −185 $mV_{Ag/AgCl}$ (Bechtel SAIC Company, LLC, 2003b). Based on the similar values of the E_{rcrev} for the mill-annealed and the as-welded material, waste package fabrication processes were determined to have no effect on the crevice corrosion susceptibility of Alloy 22. The E_{rcrev} obtained for the mill-annealed material were determined to be applicable for welded Alloy 22.

In summary, the DOE model for localized corrosion of the waste packages allows consideration of some of the environmental factors that can affect localized corrosion susceptibility. The model allows consideration of the effects of temperature, chloride concentration, and the inhibiting effects of nitrate. The model does not consider the effects of any oxidizing species that, if present, may increase the corrosion potential of the waste packages and promote localized corrosion. The DOE model does not account for the increased localized corrosion susceptibility of waste packages as a result of fabrication and closure processes.

Overall, the available information is sufficient to expect that the information necessary to assess the localized corrosion of the waste package with respect to system description and model integration will be available at the time of a potential license application.

5.1.3.1.4.6.2 Data and Model Justification

The empirical model for the E_{corr} was compared to data obtained for Alloy 22 in concentrated NaCl. Measured values of the corrosion potential were slightly lower than values predicted by the empirical model (Bechtel SAIC Company, LLC, 2003c). The dependence of the E_{corr} on pH was also reported by Dunn, et al. (2003). Although the data obtained in studies reported by Dunn, et al. suggest that the E_{corr} is not a function of chloride concentration, the values of the E_{corr} are similar to those reported by Bechtel SAIC Company, LLC (2003c).

The empirical model for the E_{rcrev} was compared to E_{rcrev} data that were not used to develop the model. The empirical model was determined to adequately predict the E_{rcrev} for Alloy 22 in chloride solutions reported by Dunn, et al. (1999). In addition, the model predicts high values of the E_{rcrev} in simulated dilute water, simulated concentrated water, and simulated acidified water at 90 °C [194 °F]. Because the E_{rcrev} values are above the E_{corr} in these test solutions, localized corrosion is not predicted to occur, which is in agreement with the results obtained for 5-year tests (Bechtel SAIC Company, LLC, 2003b).

Welding and fabrication processes typically decrease the localized corrosion resistance of passive chromium-containing alloys. Increased corrosion rates were reported by Rebak, et al. (2002) and Summers, et al. (2002, 2000) for welded and aged Alloy 22 using standardized tests designed to detect intergranular corrosion sensitivity. The localized corrosion susceptibility of welded Alloy 22 was evaluated in concentrated $CaCl_2$ solutions at 120 °C [248 °F]. Based on similar values of the E_{rcrev} for mill-annealed and as-welded Alloy 22, fabrication processes were not considered to cause an increase in the localized corrosion susceptibility of the waste package outer container. At high chloride concentrations and high temperatures, the E_{rcrev} for the mill-annealed and as-welded Alloy 22 are expected to be similar. However, the critical chloride concentration for crevice corrosion can be much lower for as-welded Alloy 22. Solution-annealed welds may also be susceptible to crevice corrosion in dilute chloride solutions at elevated temperatures (Dunn, et al., 2004). The increased crevice corrosion susceptibility of welded Alloy 22 at lower chloride concentrations is not apparent from a comparison of the E_{rcrev} for welded and mill-annealed Alloy 22 at high temperatures in concentrated chloride solutions.

Anions such as nitrate, carbonate, bicarbonate, and sulfate can inhibit the localized corrosion of Alloy 22[2] (Dunn, et al., 2003). Higher molar concentration ratios of inhibitors to chloride are required to inhibit localized corrosion of welded and thermally aged Alloy 22 compared with Alloy 22 in the mill-annealed condition. 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.

No localized corrosion of Alloy 22 was observed on the specimens tested in the long-term corrosion test facility. Crevice corrosion was observed in the electrochemical tests with concentrated calcium chloride (Bechtel SAIC Company, LLC, 2003c), however, propagation rates for localized corrosion cannot be obtained from accelerated electrochemical tests. The alternative conceptual model for localized corrosion propagation is based on a time-dependent propagation rate, which is consistent with diffusion controlled propagation of localized corrosion. Because of the lack of data on localized corrosion propagation rates in freely corroding conditions, the localized corrosion propagation rates are based on the measured penetration of Alloys 22 and C-276 in aggressive oxidizing chloride solutions under fully immersed conditions. The assumed propagation rates are probably conservative because propagation of localized corrosion of passive alloys is typically controlled by diffusion and decreases with time. In addition, formation of a thin water film on the waste package surface may limit the available cathodic surface area necessary for localized corrosion propagation.

Although the data and model abstraction for localized corrosion of the waste package seem adequate, DOE should further consider certain aspects related to waste package fabrication and the range of conditions expected in the emplacement drifts. The model parameters that can affect localized corrosion such as temperature and the evolution of ground water composition are based on project data; however, the DOE approach to assessing the effects of fabrication processes on the localized corrosion susceptibility of Alloy 22 waste package outer containers is not supported with sufficient data. The increased susceptibility to localized corrosion of the fabrication and closure welds should be included in the assessment of waste package performance.

While some information on the localized corrosion of the waste package with respect to data being sufficient for model justification may be available at the time of a potential license

[2]Dunn, D.S., L. Yang, C. Wu, and G.A. Cragnolino. "Effect of Inhibiting Oxyanions on the Localized Corrosion Susceptibility of Waste Package Container Materials." Materials Research Society Symposium CC: Scientific Basis for Nuclear Waste Management XXVIII, San Francisco, California, April 12–16, 2004. L. Browning and J. Hanchar, eds. Warrendale, Pennsylvania: Materials Research Society. In press. 2004.

application, the staff is currently of the view that DOE should provide additional information on the DOE determination of the effect of fabrication processes on localized corrosion rates of the waste package.

5.1.3.1.4.6.3 Data Uncertainty

Values of the E_{corr} for Alloy 22 specimens were obtained by DOE in a variety of solutions that are reported to be representative of solutions that may evolve at the container surface. These solutions that include simulated concentrated water, simulated acidified water, simulated dilute water, and basic saturated water are complex solutions that contain chloride, carbonate, bicarbonate, sulfate, nitrate, and fluoride as the principal anions and sodium, potassium, magnesium, and calcium as cations. In addition, long-term E_{corr} measurements were also obtained in concentrated $CaCl_2$ solutions with additions of $Ca(NO_3)_2$. Replicate measurements of the E_{corr} are consistent and values for a fixed condition fall within a 50 to 120 mV range. The corrosion potential was found to be a function of temperature, pH, chloride concentration, and nitrate concentration with pH being the most significant parameter. The pH of the test solutions encompasses the expected range of pH for solutions that may evolve at the waste package surface.

The E_{corr} was found to increase with temperature; however, this was only observed in the data for simulated acidified water that was obtained at temperatures of 25, 60, and 90 °C [77, 140, and 194 °F]. In other test solutions, the E_{corr} decreased with increasing temperature. For passive metals in solutions, the corrosion potential was dependent on the passive dissolution rate and the reduction reaction kinetics. The passive dissolution rate increases with temperature, which will tend to decrease corrosion potentials. In near-neutral or alkaline solutions, oxygen reduction is the primary reduction reaction and the oxygen concentration decreases with increasing temperature. The direct relationship between temperature and E_{corr} in the empirical model may be limited to acidic conditions, where the reduction of hydrogen ions is the primary reduction reaction. Nevertheless, the positive temperature coefficient in the empirical model for the E_{corr} is conservative with respect to the initiation of localized corrosion and waste package performance.

The determination of the localized corrosion susceptibility of Alloy 22 is based on a comparison of the crevice corrosion repassivation potentials (E_{rcrev}) and corrosion potentials (E_{corr}). The E_{rcrev} is dependent on temperature, chloride concentration, and nitrate-to-chloride concentration ratio. In addition to temperature, other factors can influence the E_{corr} 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. Recent information on crevice corrosion repassivation potentials (Bechtel SAIC Company, LLC, 2003c) and evolution of corrosion potentials (Estill, et al., 2003) appears to provide a better assessment of the electrochemical and environmental conditions needed for localized corrosion initiation.

Measurements of E_{rcrev} were conducted using similar solution chemistries used for E_{corr} measurements. The E_{rcrev} was determined to be dependent on temperature, chloride concentration, pH, nitrate concentration, and the nitrate-to-chloride concentration ratio. The E_{rcrev} tests were conducted in solutions that encompass the range of temperature and chloride concentrations in ground water that may contact the waste packages. The E_{rcrev} tests were conducted for a limited range of pH, from 4.1 to 6.4. Additional data from Brossia, et al. (2001) are used to develop the empirical model for E_{rcrev}. Solution pH values expected as a result of the evaporation of seepage waters range from 4.5 to 10.5 (Bechtel SAIC Company, LLC, 2003a). The effect of nitrate on the E_{rcrev} for Alloy 22 was evaluated at nitrate-to-chloride molar concentration ratios of 0.01 and 0.1; however, larger ratios are expected in both seepage waters and dusts that may deliquesce.

The assessment of the effects of welding on the localized corrosion susceptibility is limited to a single test condition that does not allow the evaluation of fabrication effects over the complete range of expected environmental conditions. The dependence of the E_{rcrev} on environmental parameters for the welded material may be substantially different from the mill-annealed material (Dunn, et al., 2003, 2004).

Propagation rates for localized corrosion are based on rates obtained in standardized acidic chloride solutions. Localized corrosion rates in solution chemistries that may evolve on the waste package surfaces as a consequence of deliquescence of dust or by evaporation of seepage water have not been determined, in part, because no localized corrosion was observed in long-term corrosion tests. Characterization of localized corrosion rates in the range of possible near-field solutions may not be necessary because the chemistry that develops in the occluded localized corrosion cells is not strongly influenced by the bulk environmental chemistry. Propagation rates for localized corrosion are largely independent of the chemistry of the external environment as long as the chemistry of the occluded region remains aggressive and promotes active dissolution within the crevice.

In summary, the technical basis for the parameter values used to determine the critical potentials for localized corrosion are justified according to the results of laboratory experiments. The values for localized corrosion propagation rates used in the DOE model are not based on the results of laboratory experiments and are obtained from propagation rates in acidic oxidizing chloride solutions. Selection of constant values of propagation rates is conservative because localized corrosion propagation rates typically decrease with penetration depth. The effects of fabrication processes are based on a limited set of data that does not account for the increased localized corrosion susceptibility of welded Alloy 22 in less concentrated solutions.

While some information provided by DOE on the localized corrosion of the waste package with respect to data uncertainty may be available at the time of a potential license application, the staff is currently of the view that DOE should provide additional information on the DOE determination of the effect of fabrication processes on localized corrosion rates of the waste package.

5.1.3.1.4.6.4 Model Uncertainty

Uncertainty in the empirical models for E_{corr} and E_{rcrev} are assumed to result from measurement uncertainty. An uncertainty of ± 2 standard deviations was determined to be sufficient to encompass 95 percent of the data used to develop the empirical models, provided that random variations are the only source of error (Bechtel SAIC Company, LLC, 2003c). The magnitude of this uncertainty is approximately 50 to 100 mV. The range of model uncertainty is consistent with independent assessments of the E_{rcrev}. Observed variations in the corrosion potentials are within 50 mV under acidic conditions with greater variations observed in alkaline solutions (Dunn, et al., 2003).

Environmental factors in the localized corrosion model include chloride concentration, nitrate concentration, temperature, and pH. The localized corrosion abstraction is valid for solutions that develop on the waste package surfaces as a consequence of the deliquescence of dust that contains chloride and nitrate as the primary soluble species. Analyses of dust chemistry (Bechtel SAIC Company, LLC, 2003a) indicate that solutions that result from the deliquescence of dust are expected to have chloride-to-nitrate concentration ratios less than 10. Dust may also contain other less soluble (compared to nitrate) inhibiting anions such as carbonate and sulfate.

The model abstraction does not explicitly consider the effects of fabrication processes on the localized corrosion susceptibility of the waste package container materials based on a comparison of the repassivation potentials for mill annealed and as-welded Alloy 22 in concentrated chloride solutions. The DOE approach is not consistent with independent assessments of the effects of fabrication processes on the localized corrosion susceptibility of Alloy 22 (Dunn, et al., 2004).

Alternative conceptual models for localized corrosion include the critical crevice corrosion temperature and the critical pitting temperature for Alloy 22. Data for critical temperatures for localized corrosion are obtained in environments that are not directly related to the expected environments within the emplacement drifts of the potential repository. As a result of the lack of repository relevant data, the alternative conceptual models for localized corrosion initiation are not considered as valid alternatives to the critical potential model (Bechtel SAIC Company LLC, 2003c). Critical pitting and crevice corrosion temperatures are known to be dependent on the test environments. Standardized tests for measuring critical crevice and pitting temperatures are used to rank or compare the relative corrosion resistance of alloys. Data cited in Bechtel SAIC Company, LLC (2003c) indicate that, in an acidic oxidizing solution containing 24,300 ppm chloride, the critical crevice corrosion temperature for Alloy 22 is 102 °C [216 °F]. Lower critical crevice corrosion temperatures have been reported for Alloy 22 in concentrated $FeCl_3$ solutions. While such tests are valid measures of crevice corrosion susceptibility, the composition of the environment has a strong influence of localized corrosion initiation. With the presence of inhibitor anions, localized corrosion was not observed in concentrated chloride solutions under potentiostatic conditions consistent with strongly oxidizing conditions.[3] The role of environment chemistry, which is included in the critical potential model, should be considered in the assessment of localized corrosion susceptibility.

The alternative conceptual model for localized corrosion propagation is based on a time-dependent growth rate. The propagation rate for localized corrosion is generally accepted to be diffusion controlled. As a result, the localized corrosion rate decreases with time. While the time dependent growth rate is a more accurate description of the propagation rate under conditions where localized corrosion can be initiated, sufficient data are not available to use the alternative model. The constant propagation rate model is conservative with respect to propagation rate and waste package penetration time.

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[3]Dunn, D.S., L. Yang, C. Wu, and G.A. Cragnolino. "Effect of Inhibiting Oxyanions on the Localized Corrosion Susceptibility of Waste Package Container Materials." Materials Research Society Symposium CC: Scientific Basis for Nuclear Waste Management XXVIII, San Francisco, California, April 12–16, 2004. L. Browning and J. Hanchar, eds. Warrendale, Pennsylvania: Materials Research Society. In press. 2004.

In summary, no alternative conceptual model was developed for the localized corrosion of the waste package. The localized corrosion model contains sufficient environmental parameters to evaluate the effects of coupled processes such as the deliquescence of dust that contains nitrate and chloride. The localized corrosion model does not explicitly consider the effects of the alloy compositional variations or the effects of fabrication processes on the localized corrosion susceptibility.

While some information on the localized corrosion of the waste package with respect to data being sufficient for model uncertainty may be available at the time of a potential license application, the staff is currently of the view that DOE should provide additional information on the DOE determination of localized corrosion rates of the waste package.

5.1.3.1.4.6.5 Model Support

In general, values of E_{corr} and E_{rcrev} calculated using the empirical models are consistent with independent measurements of these potentials for Alloy 22 in the mill-annealed condition. The determination that the crevice corrosion susceptibilities of Alloy 22 in the mill-annealed and as-welded conditions is not supported by previous investigations. Formation of topologically closed-packed phases has been reported 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), and a lower critical pitting temperature for welded Alloy 22 (Sridhar, 1990), do not support the DOE conclusion on the susceptibility to localized corrosion after thermal aging. The effect of solution annealing on the microstructural stability, localized corrosion resistance, and stress corrosion cracking susceptibility of Alloy 22 welds has not been fully characterized. Dunn, et al. (2003) have shown that solution annealing may not improve the crevice corrosion susceptibility of Alloy 22. In addition, variations in the annealing parameters, base and filler metal compositions, and welding parameters may exacerbate microstructural alterations and further reduce the stress corrosion cracking and localized corrosion resistance of the alloy.

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-ppm chloride with 30-ppm 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 suggest that the propagation rates used by DOE sufficiently bound the range of propagation rates for similar nickel-chromium-molybdenum alloys. Because the propagation rates selected by DOE are constant and do not decrease with time, the propagation rates are conservative.

In summary, the environmental parameters in the localized corrosion model are consistent with independent assessments of the key environmental variables for the initiation of localized corrosion of waste package container materials (Dunn, et al., 2004). The influence of a metallurgical condition on the localized corrosion susceptibility is not in agreement with previous investigations. Propagation rates in the localized corrosion model are conservatively calculated using a distribution of constant penetration rates obtained from standardized tests in acidic oxidizing chloride solutions.

Overall, the available information is sufficient to expect that the information necessary to assess the localized corrosion of the waste package with respect to model abstraction output being supported by objective comparisons will be available at the time of a potential license application.

5.1.3.1.4.7 Microbially Influenced Corrosion of the Waste Package

5.1.3.1.4.7.1 Model Integration

Microbially influenced corrosion is known as a problem affecting metallic materials used in many engineering applications (Thierry and Sand, 1995) including high-level waste disposal (Geesey and Cragnolino, 1995; Bachofen, 1990, 1991). Microbially influenced corrosion is usually manifested in the form of localized corrosion (Lewandowski, 2000; Little, et al., 2000) that tends to be catastrophic in effect. Microorganisms may produce extreme environments. These environments may be concentrated on metal surfaces, especially near the weak points such as welds or heat affected zones.

In the potential repository, microbially influenced corrosion is considered impossible during the low-relative-humidity and high-temperature phase because the microorganisms associated with microbially influenced corrosion would not be active under these conditions. However, as the temperature decreases and when the relative humidity reaches a threshold value (90 percent), certain microorganisms may become active and potentially cause microbially influenced corrosion to the engineered barrier system (Bechtel SAIC Company, LLC, 2003b,c).

Stainless steels are known to be susceptible to microbially influenced corrosion (Amaya, 2003; Amaya and Miyuki, 1999). Nickel-based alloys, such as Alloy 625, were also found to be susceptible to localized corrosion in natural seawater at electrochemical potentials that were observed by a microbial ennoblement effect (Martin, et al., 2003). However, there has been no credible evidence for microbially influenced corrosion of Alloy 22. From the extensive studies on the localized corrosion of Alloy 22, the repassivation potential of Alloy 22 at or slightly above the critical temperature {70 °C [158 °F]} is greater than 0.70 V_{SCE} for 5 M $CaCl_2$ + 0.1M $NaNO_3$ solution (Bechtel SAIC Company, LLC, 2003b). For temperatures lower than 70 °C [158 °F], no localized corrosion could be initiated at any potentials even in nitrate-free 5 M $CaCl_2$ solutions (Bechtel SAIC Company, LLC, 2003b). It has been reported that microbial activities were responsible for the ennoblement of stainless steels and nickel-based alloys in natural seawater to near 0.40 V_{SCE} (Martin, et al., 2003; Amaya, 2003) and caused the initiation of localized corrosion. However, it is unlikely for localized corrosion to initiate for Alloy 22 by ennoblement caused by microbial activity because the repassivation potential for Alloy 22 is extremely high.

Limited experimental studies have been conducted by DOE on the susceptibility of Alloy 22 to microbially influenced corrosion in the presence of Yucca Mountain bacteria (Bechtel SAIC Company, LLC, 2003b). In a 5-month immersion experiment (Lian, et al.,1999), no signs of localized corrosion for Alloy 22 were observed. The corrosion potential of the Alloy 22, and all other metals tested in the experiment, was found to be lower in the bacteria-containing solution than in the sterile solution (Bechtel SAIC Company, LLC, 2003c). Based on these observations and the high repassivation potentials of Alloy 22, any microbially influenced effect on localized corrosion was not considered in the DOE analysis.

In immersion experiments using electrochemical polarization methods, however, Lian, et al. (1999) measured a higher corrosion rate for Alloy 22 in the bacteria-containing solution than in the sterile solution. They also observed similar higher corrosion rates in the presence of Yucca Mountain bacteria than in abiotic solutions for other corrosion resistant alloys such as Type 304 stainless steel and Alloy 625. The increases in electrochemically measured corrosion rate were attributed to general corrosion.

To account for the uncertainties, a microbially influenced corrosion factor (f_{MIC}) uniformly distributed between one and two is applied to the waste package outer container general corrosion abstraction when the relative humidity at the waste package outer container surface is above 90 percent, which is considered the threshold relative humidity in the DOE model analysis (CRWMS M&O, 2000n).

The model for microbially influenced corrosion of the waste package considered the effect of bacterial activity on uniform corrosion. However, no consideration is given to the microbially induced effects on the localized corrosion susceptibility of Alloy 22, especially at welded areas. DOE has acknowledged, in the course of the ongoing review by the NRC staff of Technical Basis Document 6: Waste Package and Drip Shield Corrosion (Bechtel SAIC Company, LLC, 2003b) and its appendixes, that additional information should be provided on microbially influenced effects on localized corrosion. The specific information is being developed as part of the ongoing staff review of the documents and key technical issues agreements.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.1.5), is sufficient to expect that the information necessary to assess microbially influenced corrosion of the waste package with respect to system description and model integration will be available at the time of a potential license application.

5.1.3.1.4.7.2 Data and Model Justification

Based on recent work by Yang and Cragnolino (2004), the corrosion rate increases observed by Lian, et al. (1999) using electrochemical techniques in the presence of sulfate-reducing bacteria may not be due to the corrosion of the metal. At least part of the increase was due to the oxidation reaction of the reducing species produced by the microorganisms. Therefore, the value of the microbially influenced corrosion factor, f_{MIC}, for general corrosion derived from the experiments by Lian, et al. (1999) is a conservative value because it contains contributions from the oxidation reactions of the chemical species formed by the microorganisms.

On the other hand, microbially influenced corrosion is usually manifest with localized corrosion. Attributing the high corrosion rate observed in the presence of microorganisms to only general corrosion is not reasonable. If the observed increase in corrosion is true, localized corrosion should be considered. As discussed previously, the observed increase with electrochemical methods may produce artifacts; other methods should be used to verify the measurements. Solution analysis is a good method for this purpose. Lian, et al. (1999) also conducted the measurement with the solution analysis method and reported high values for chromium {1.05 mg/L [1.05 ppm]} and nickel {0.1 mg/L [0.1 ppm]} in the solution containing microorganisms versus nondetectable readings in the sterile solution for Alloy 22. The increases in both chromium and nickel may be an indication of localized corrosion.

The data provided by DOE account for the effects of microbial activity on the uniform corrosion of waste packages, but not fully consider the effects on the susceptibility to localized corrosion. DOE has acknowledged, in the course of the ongoing review by the NRC staff of Technical Basis Document 6: Waste Package and Drip Shield Corrosion (Bechtel SAIC Company, LLC, 2003b) and its appendixes, that additional information should be provided on microbially influenced effects on localized corrosion. The specific information is being developed as part of the ongoing staff review of the documents and key technical issue agreements.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.1.5), is sufficient to expect that the information necessary to assess justifications for microbially influenced corrosion of the waste package with respect to data being sufficient for model justification will be available at the time of a potential license application.

5.1.3.1.4.7.3 Data Uncertainty

In a 5-month exposure experiment (Lian, et al.,1999), no localized corrosion of Alloy 22 and other corrosion resistant metals including Type 304 stainless steel and Alloy 625 was observed. However, for Alloy 22, the solution contents of chromium and nickel increased from not detectable level in the sterile test cell to 1.05 and 0.1 mg/L, respectively, in the test cell containing the microorganisms. These increases, if continued with time, may be an indication of localized corrosion, even though this process is considered unlikely based on the repassivation potential measurements at temperatures lower than the critical temperatures.

In the same experiment for a less corrosion-resistant alloy, Type 304 stainless steel, the increases were also from not detectable to 1.03 mg/L [1.03 ppm] for chromium and from not detectable to 0.04 mg/L [0.04 ppm] for nickel. The increases for the Type 304 stainless steel are slightly less than for Alloy 22. This is an important indication of uncertainty of the data. More experiments and longer term experiments should be conducted to verify the dissolution rate of Alloy 22 and to test if the dissolution rate would continue in the presence of microorganisms.

The technical bases for the microbially influenced corrosion rate factor and distribution used in the model abstraction are reasonable and account for experimental uncertainty. However, the influence of fabrication processes is not considered in the evaluation of data uncertainty. DOE has acknowledged, in the course of the ongoing review by the NRC staff of Technical Basis Document 6: Waste Package and Drip Shield Corrosion (Bechtel SAIC Company, LLC, 2003b) and its appendixes, that additional information should be provided on microbially influenced effects on localized corrosion. The specific information is being developed as part of the ongoing staff review of the documents and key technical issue agreements.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.1.5), is sufficient to expect that the information necessary to assess the microbially influenced corrosion of the waste package with respect to data uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.1.4.7.4 Model Uncertainty

For general corrosion, the DOE model is conservative because the microbially influenced corrosion factor was distributed between one and two. The upper bound was the value derived from the maximum corrosion rate measured with the electrochemical methods in the presence of microorganisms including sulfate reducing bacteria (Lian, et al., 1999). Based on the work by Yang and Cragnolino (2004), the corrosion rate measured by Lian, et al. (1999) in the presence of sulfate-reducing bacteria was inevitably enhanced by the oxidation of reducing species produced by the microbial activities. Therefore, the enhancement factor for general corrosion obtained with the electrochemical method is conservative. No alternative conceptual models for microbially influenced corrosion were considered.

Localized corrosion as a result of microbial activity was not considered (Bechtel SAIC Company, LLC, 2003b) because the corrosion potential of Alloy 22 measured in the presence of Yucca Mountain bacteria was much lower than the repassivation potential of Alloy 22 (Bechtel SAIC Company, LLC, 2003b). In addition, the temperatures at which the microorganisms are believed to be active are lower than the critical temperature for Alloy 22 to be susceptible to localized corrosion. No model abstraction was considered for microbially influenced localized corrosion. DOE has acknowledged, in the course of the ongoing review by the NRC staff of Technical Basis Document 6: Waste Package and Drip Shield Corrosion (Bechtel SAIC Company, LLC, 2003b) and its appendixes, that additional information should be provided on microbially influenced effects on localized corrosion. The specific information is being developed as part of the ongoing staff review of the documents and key technical issue agreements.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.1.5), is sufficient to expect that the information necessary to assess microbially influenced corrosion of the waste package with respect to model uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.1.4.7.5 Model Support

The model abstraction used for the microbially influenced corrosion effect for general corrosion in the DOE performance analysis (Bechtel SAIC Company, LLC, 2003b,c) is adequately conservative. The arguments for the exclusion of a microbially influenced corrosion effect on localized corrosion based on the repassivation measurements also seem adequate. However, it is not known if the repassivation potentials measured with an electrochemical potential polarization method sufficiently bounds the repassivation potentials under steady-state conditions in the presence of microorganisms. According to measurements by Yang and Cragnolino (2004), the repassivation potentials for the Type 304 stainless steel obtained with the polarization methods were −0.15 to +0.06 V_{SCE}, and they were not affected by the presence of sulfate-reducing bacteria. However, localized corrosion on this type of metal was observed at potentials below these values in the immersion tests with the presence of sulfate-reducing bacteria by Ringas and Robinson (1988) and Rao and Satpathy (2000). Therefore, the repassivation potential measured by the electrochemical polarization methods may not include the effect of the local chemical species produced by the microbial activity and adsorbed onto the metal surface because these chemical species would be oxidized during the potential hold or sweep at higher values (Jain, et al., 2003). As a result, when the potential is decreased to determine the repassivation potential, the build-up of the chemical species resulting from the bacteria metabolic activity on the localized surface would not be available to cause the effect.

An immersion test for Alloy 22 in the presence of microorganisms would answer the question conservatively, as the thin film of water may limit the bacterial growth, especially in the presence of inhibitors. However, the DOE immersion test was too short (5 months) to sufficiently demonstrate the effect of microbially influenced corrosion for Alloy 22. Rao and Satpathy (2000) showed a small pit of nearly round shape {approximately 0.08 mm [0.003 in] in diameter} for a Type 304 stainless steel specimen after 25 days of immersion in a solution containing sulfate-reducing bacteria at room temperature. Ringas and Robinson (1988) observed pitting corrosion for Type 304L stainless steel specimens after 4 months of immersion in solutions containing sulfate reducing bacteria at room temperature. It may take much longer for Alloy 22, which is a far more corrosion resistant alloy than stainless steels, to develop localized corrosion in the presence of microorganisms if it is susceptible to microbially influenced corrosion. Therefore, the DOE immersion test for 5 months may not be sufficient to demonstrate the resistance of Alloy 22 to microbially influenced localized corrosion.

The technical basis for the microbially influenced effect on general corrosion is supported on an empirical correlation and this correlation appears valid for the range of repository conditions to be expected in the emplacement drifts. The information presented indicates that the enhancement factor for microbially influenced corrosion does not underestimate the actual degradation and failure of the waste packages. However, the model abstraction does not bound the effect of microbial activity on localized corrosion more data should be provided to support the exclusion of localized corrosion of Alloy 22, especially at fabrication affected areas. DOE has acknowledged, in the course of the ongoing review by the NRC staff of Technical Basis Document 6: Waste Package and Drip Shield Corrosion (Bechtel SAIC Company, LLC, 2003b) and its appendixes, that additional information should be provided on microbially influenced effects on localized corrosion. The specific information is being developed as part of the ongoing staff review of the documents and key technical issue agreements.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.1.5), is sufficient to expect that the information necessary to assess microbially influenced corrosion of the waste package with respect to model abstraction output being supported by objective comparisons will be available at the time of a potential license application.

5.1.3.1.4.8 Stress Corrosion Cracking of the Waste Package

5.1.3.1.4.8.1 Model Integration

Stress corrosion cracking is one of the potential failure modes of the Alloy 22 outer container. Stress corrosion cracking requires the combination of a susceptible material or microstructure, an aggressive environment, and an applied or residual tensile stress. Although nickel-base alloys are known to be resistant to environmentally assisted cracking in hot chloride solutions, stress corrosion cracking of Alloy 22 has been reported in simulated ground water solutions that may contact the waste packages (Andresen, et al., 2001, 2003; King, et al., 2002; Estill, et al., 2002). DOE proposed two models to evaluate stress corrosion cracking susceptibility: stress intensity threshold model and the slip dissolution/film rupture model (CRWMS M&O, 2000a). The stress corrosion cracking stress intensity 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, 2000i). For both the slip dissolution/film rupture model and the stress corrosion cracking stress intensity threshold model, through-wall radial cracking is predicted as a result of the high values of the calculated stress intensity factor. Both crack initiation and propagation are based on the slip dissolution and film rupture theory (Bechtel SAIC Company, LLC, 2003b).

The concept of threshold stress intensity factor is used to define crack arrest or propagation of preexisting cracks from either manufacturing flaws or incipient cracks. A crack blunting criterion is used to determine the threshold stress intensity factor, assuming that stress corrosion cracking will cease as the crack blunts when the general corrosion rate exceeds the crack propagation rate. DOE assumes that stress corrosion cracking is limited to the surface area defined by the closure-lid welds because the disposal containers will be solution annealed to eliminate the residual stresses associated with fabrication welds before waste loading and closure welding. Therefore, the approach adopted by DOE to mitigate or eliminate the possibility of crack growth is to reduce the residual stresses associated with closure welding. The current waste package design for the potential license application consists of two alternative processes of mechanical residual stress mitigation (i.e., laser peening or controlled plasticity burnishing) for the outer lid closure weld (Bechtel SAIC Company, LLC, 2003b). Laser peening is the baseline process for the potential license application and uses multiple passes of a high-power pulsed laser beam to introduce compressive stresses on the surface.

The DOE stress corrosion cracking models consider weld residual stress the only source of stresses significant to stress corrosion cracking (CRWMS M&O, 2000a; Bechtel SAIC Company, LLC, 2003b). 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 potential repository have not been assessed by the DOE, such as stresses generated at the line of contact of the waste package with the emplacement pallet. Residual stresses from waste package fabrication or applied stresses resulting from seismic and rockfall events combined with the necessary environmental conditions may be sufficient to cause stress corrosion cracking of the outer container. As a result, the waste package may experience localized plastic deformations in locations where it interacts with the drip shield and pallet and existing stress corrosion cracks in the closure lid-weld area may propagate at an increased rate. Furthermore, DOE proposed solution annealing and laser peening (or controlled plasticity burnishing) to eliminate any residual stresses created during the fabrication and the closure of the waste packages; thus, stress corrosion cracking testing of mitigated samples is not considered. Accordingly, the effects of welding and postweld treatments on the stress corrosion cracking susceptibility of Alloy 22 in the expected waste package environments have not been evaluated by the DOE.

Overall, the available information is sufficient to expect that the information necessary to assess the stress corrosion cracking of the waste package with respect to system description and model integration will be available at the time of a potential license application.

5.1.3.1.4.8.2 Data and Model Justification

For the stress corrosion cracking of Alloy 22, crack propagation rates ranging from 1.0×10^{-14} to 5.0×10^{-13} m/s [3.9×10^{-13} to 1.9×10^{-11} in/s] were measured in an air-saturated alkaline solution (pH 13.4) with a composition similar to basic saturated water (Table 5.1.3.1-4) at 110 °C [230 °F] (Bechtel SAIC Company, LLC, 2003b). The stress corrosion cracking tests were performed under constant load conditions using three Alloy 22 compact tension specimens loaded to K_I values of 30 and 45 MPa·m$^{1/2}$ [27.3 and 40.9 ksi·in$^{1/2}$]. These crack growth rates were used to determine the value of the repassivation parameter n. 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, the measured crack growth rates lead to a mean value of 1.304 for n with the lower and upper bounds of 0.984 and 1.624, using the two-time standard deviation value of the normal distribution. DOE recognizes that the variation of n, which is one of the most important parameters in the model, as a function of environmental factors, is not available because of a lack of experimental data. DOE also recognizes that the samples used to determine the n values were also used to validate the model. Thus, the stress corrosion cracking model has not been validated for Alloy 22 in the environments expected to contact the waste packages.

In the case of the threshold stress intensity factor for stress corrosion cracking, 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, 2000i). 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 four different initial K_I values ranging from 22 to 43 MPa·m$^{1/2}$ [20 to 39 ksi·in$^{1/2}$]. However, the experimentally measured K_{Iscc} values were not used in the technical basis document because plane strain conditions have not been satisfied in the test specimens (Bechtel SAIC Company, LLC, 2003b). Instead, K_{Iscc} is defined using a crack blunting criterion. It is assumed that crack blunting occurs as the crack growth rate approaches the general dissolution rate at the crack tip. Under such conditions, a stress corrosion crack will not grow. The revised n values and a mean general corrosion rate of 7.23 nm/yr [2.85×10^{-4} mpy] were used to determine the values of K_{Iscc}. The respective K_{Iscc} values range 2.65–28.50 MPa·m$^{1/2}$ [2.41–25.93 ksi·in$^{1/2}$] with a mean value of 11.38 MPa·m$^{1/2}$ [10.35 ksi·in$^{1/2}$]. It is claimed that this K_{Iscc} value is highly conservative considering the high stress corrosion cracking resistance of Alloy 22. Sufficient justification for using K_{Iscc} as a bounding parameter for performance was not provided by DOE.

The current DOE waste package design precludes stress corrosion cracking through mitigation of residual tensile stress in the closure weld. Both residual stress measurements and finite element stress analyses were conducted to assess the effectiveness of the stress mitigation processes (Bechtel SAIC Company, LLC, 2003b). Residual stress measurements were obtained from both laser peened and controlled plasticity burnished 25.4-mm [1-in]-thick Alloy 22 welded plates, indicating that the depth of the compressive layer achieved with either stress mitigation technique is greater than 5 mm [0.20 in]. The measured compressive residual stress distributions with depth are supported by the finite element calculations. As noted, DOE proposed postweld treatments to eliminate any tensile residual stresses. Only examples of residual stress measurements obtained from test coupons were reported, and verification of this assumption has not been demonstrated. Thus, it is necessary to verify what process controls are used to assure 100-percent equal coverage of the compressive layer on the closure welds.

Evaluation of the complete stress distribution including hoop, radial, and longitudinal stresses, as well as through-thickness residual stress, is also needed. Furthermore, given the current waste package design, the basis for the applicability of the stress distribution obtained from test coupons to the actual welded waste package containers should be justified.

The DOE model abstraction for stress corrosion cracking of waste packages considers the important contributions of flaw frequency, size distribution, and orientation, as well as the residual stress, stress profiles, and stress intensity factors. The model abstraction has many of the necessary components to assess susceptibility to stress corrosion cracking and to predict crack propagation rates. The model abstraction, however, is supported by many assumptions, parameters, and calculations that need to be verified, such as flaw frequency and distribution parameters, residual stresses after solution annealing and quenching of the disposal container, and both magnitude and variation in the residual stress profiles after laser peening or controlled plasticity burnishing. Effects of waste package fabrication, material composition and material property variations, and environmental variations are not accounted for in the DOE model abstraction. DOE agreed (Schlueter, 2000) to provide additional information on the stress corrosion cracking susceptibility of the waste packages for conditions that may exist in the potential repository as well as the effects of stress corrosion cracking on the release of radionuclides.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.1.5), is sufficient to expect that the information necessary to assess the stress corrosion cracking of the waste package with respect to data being sufficient for model justification will be available at the time of a potential license application.

5.1.3.1.4.8.3 Data Uncertainty

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. In the application of the slip dissolution/film rupture model to Alloy 22, DOE adopted values ranging from 0.984 to 1.624 for the repassivation slope, n (Bechtel SAIC Company, LLC, 2003b). This range of values for n was based on results from three Alloy 22 compact tension specimens loaded to K_I values of 30 and 45 $MPa{\cdot}m^{1/2}$ [27.3 and 40.9 $ksi{\cdot}in^{1/2}$] using one water chemistry and test temperature. Input for the model includes average crack growth rates ranging from 1.0×10^{-14} to 5.0×10^{-13} m/s [3.9×10^{-13} to 1.9×10^{-11} 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 require 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 based on 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.

For the effect of lead solution chemistry on stress corrosion cracking, 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 ppm). Tests were conducted at 250 °C [452 °F] using U-bend specimens. These test conditions were extremely severe in lead concentrations and temperature. In contrast to the results reported by Barkatt and Gorman (2000), Andresen, et al. (2004) did not observe an increase in stress corrosion cracking susceptibility when 1,000-ppm lead (as $PbNO_3$) was added to basic saturated water test solutions, however, the solubility of lead in basic saturated water is low. Csontos, et al.[4,5] have reported no stress corrosion cracking of mill-annealed and welded Alloy 22 U-bend specimens in saturated $PbCl_2$ and $PbNO_3$ solutions. The solubility of lead in ground waters that enter the emplacement drifts is likely to be low as a result of the presence of carbonate, bicarbonate, chloride, and sulfate. Concentration of ground water as a result of evaporation is unlikely to increase lead concentrations, because of the low solubility of lead salts (Bechtel SAIC Company, LLC, 2003a).

The effects of waste package fabrication processes (e.g., welding and heat treatments) on stress corrosion cracking of candidate container materials remain a concern. 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. 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 in the expected waste package environment have not been fully evaluated by the DOE. Additionally, the DOE stress corrosion cracking models consider weld residual stress the only source of stresses significant to stress corrosion cracking. Accordingly, the effects of other possible types of applied stresses in the potential repository have not been assessed.

Uncertainties in data used to analyze the effects of initial defects on the performance of the waste package outer barrier (CRWMS M&O, 2000o) have not been characterized or propagated through the model abstraction. The 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.

To address these concerns, DOE agreed (Schlueter, 2000) to provide information on stress corrosion cracking including mode 1 parameters justification, credible environmental conditions,

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[4]Csontos, A.A., Y.-M. Pan, D.S. Dunn, L. Yang, and G.A. Cragnolino. "Pb Assisted Stress Corrosion Cracking Susceptibility of Alloy C-22 Weldments." Proceedings of the Materials Science & Technology 2003—Effect of Processing on Materials Properties for Nuclear Waste Disposition, Chicago, Illinois, November 9–12, 2003. R. Rebak, ed. Metallurgical and Materials Transactions. Warrendale, Pennsylvania: The Minerals, Metals, and Materials Society. In press. 2004.

[5]Csontos, A.A., Y.-M Pan, D.S. Dunn, L. Yang, and G.A. Cragnolino. "The Effect of Environmental Chemistry on the Pb Assisted Stress Corrosion Cracking Susceptibility of Mill-Annealed Alloy 22 and GTAW Weldments." Materials Research Society Symposium CC: Scientific Basis for Nuclear Waste Management XXVIII, San Francisco, California, April 12–16, 2004. L. Browning and J. Hanchar, eds. Warrendale, Pennsylvania: Materials Research Society. In press. 2004.

as well as a full range of metallurgical conditions for stress corrosion cracking and its mitigation processes.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.1.5), is sufficient to expect that the information necessary to assess the stress corrosion cracking of the waste package with respect to data uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.1.4.8.4 Model Uncertainty

The DOE evaluation of the stress corrosion cracking susceptibility of Alloy 22 originally considers two alternative models, the stress corrosion cracking stress intensity threshold model and the slip dissolution/film rupture model (CRWMS M&O, 2000a). Because the experimentally measured K_{Iscc} values were considered invalid, K_{Iscc} is defined using a crack blunting criterion that is also based on the slip dissolution and film rupture theory in the technical basis document (Bechtel SAIC Company, LLC, 2003b). Therefore, DOE has not considered other alternative models for stress corrosion cracking. The slip dissolution/film rupture model for Alloy 22 used a limited amount of data obtained for Alloy 22. The DOE evaluation of the stress corrosion cracking susceptibility of Alloy 22 should consider the effects of variations in water chemistry, material properties, fabrication and welding, and long-term exposure to elevated temperatures. To address this concern, DOE agreed (Schlueter, 2000) to provide additional data on stress corrosion cracking.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.1.5), is sufficient to expect that the information necessary to assess the stress corrosion cracking of the waste package with respect to model uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.1.4.8.5 Model Support

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 a limited amount of data obtained from laboratory tests (CRWMS M&O, 2000i; Bechtel SAIC Company, LLC, 2003b). 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 amount of data obtained in boiling water reactor environments (Ford and Andresen, 1988; Ford, 1990). Similar data are not available for Alloy 22 in the expected waste package environments. In addition, the technical basis document (Bechtel SAIC Company, LLC, 2003b) replaces the experimentally measured K_{Iscc} values with the calculated ones based on the crack blunting criterion. Although the associated data and model uncertainties are not adequately addressed, the DOE model abstraction for stress corrosion cracking of the waste package appears conservative.

Overall, the available information is sufficient to expect that the information necessary to assess the stress corrosion cracking of the waste package with respect to model abstraction output being supported by objective comparisons will be available at the time of a potential license application.

5.1.3.1.5 Summary and Status of Key Technical Issue Subissues and Agreements

Table 5.1.3.1-2 provides the status of all key technical issue subissues referenced in Section 5.1.3.1.2 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 review methods discussed in Section 5.1.3.1.4. Note the status and the detailed agreements pertaining to all the key technical issue subissues are provided in Table 1.2-1 and Appendix A.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (e.g., specified testing or analyses), indicates that information necessary to begin a technical review will likely be available at the time of a potential license application. As noted in this section of the report, however, further information should be provided on the use of calculated passive corrosion rates over the entire temperature range of intended use (Section 5.1.3.1.4.5), and on the effect of alloy compositional variations and fabrication processes on localized corrosion rates of the waste package (Section 5.1.3.1.4.6).

Table 5.1.3.1-2. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreement*
Container Life and Source Term	Subissue 1—The Effects of Corrosion Processes on the Lifetime of the Containers	Closed-Pending	CLST.1.01 through CLST.1.17
	Subissue 2—The Effects of Phase Instability of Materials and Initial Defects on the Mechanical Failure and Lifetime of the 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-Hydrological-Chemical Processes on Waste Package Chemical Environment	Closed-Pending	ENFE.2.04 ENFE.2.14

Table 5.1.3.1-2. Related Key Technical Issue Subissues and Agreements (continued)			
Key Technical Issue	**Subissue**	**Status**	**Related Agreement***
Evolution of the Near-Field Environment	Subissue 3—Effects of Coupled Thermal-Hydrological-Chemical Processes on Chemical Environment for Radionuclide Release	Closed-Pending	ENFE.3.01
	Subissue 5—Effects of Coupled Thermal-Hydrological-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
	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 review methods. Note: Key Technical Issue Agreement GEN.1.01 pertains to multiple integrated subissues, as well as some specific issues related to this integrated subissue.			

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5.1.3.2 Mechanical Disruption of Engineered Barriers

5.1.3.2.1 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, multipurpose canister, waste package pallet, drip shield, spent nuclear fuel cladding, and drift invert system. Although engineered backfill is not presently included in the engineered barrier system design, it may be placed within the emplacement drifts of the potential geologic repository for commercial spent nuclear fuel and high-level waste. If used, engineered backfill also would be assessed to determine how its performance characteristics and interactions with other engineered barrier system 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 degradation, and criticality. The relationship between this integrated subissue to other integrated subissues is depicted in Figure 5.1.3.2-1. The overall organization and identification of all the integrated subissues are 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 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 system when subjected to mechanically disruptive events and (ii) screening arguments used to justify the exclusion of mechanical disruption of engineered barriers processes from consideration. To date, however, the only total system performance assessment abstractions pertaining to mechanical disruption of the engineered barrier system that have been provided for review are those addressing igneous intrusion. Total system performance assessment abstractions for seismicity and rockfall and drift degradation have yet to be provided for review.

Igneous effects accounted for in the mechanical disruption of engineered barriers model abstraction presently are 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 (Section 5.1.3.10). Key processes associated with the mechanical disruption of engineered barriers by igneous intrusion are (i) basaltic magma flows into potential repository drifts, (ii) engineered barrier component response to basaltic magma exposure, and (iii) basalt and engineered barrier system cooling (which allows reestablishment of long-term hydrologic transport processes).

According to Bechtel SAIC Company, LLC (2002a), specific information to be developed to support the DOE total system performance assessment of the seismic scenario includes consideration of the response of the drip shield, waste package, and spent nuclear fuel cladding as functions of ground motion levels, rockfall, and fault displacement for degraded

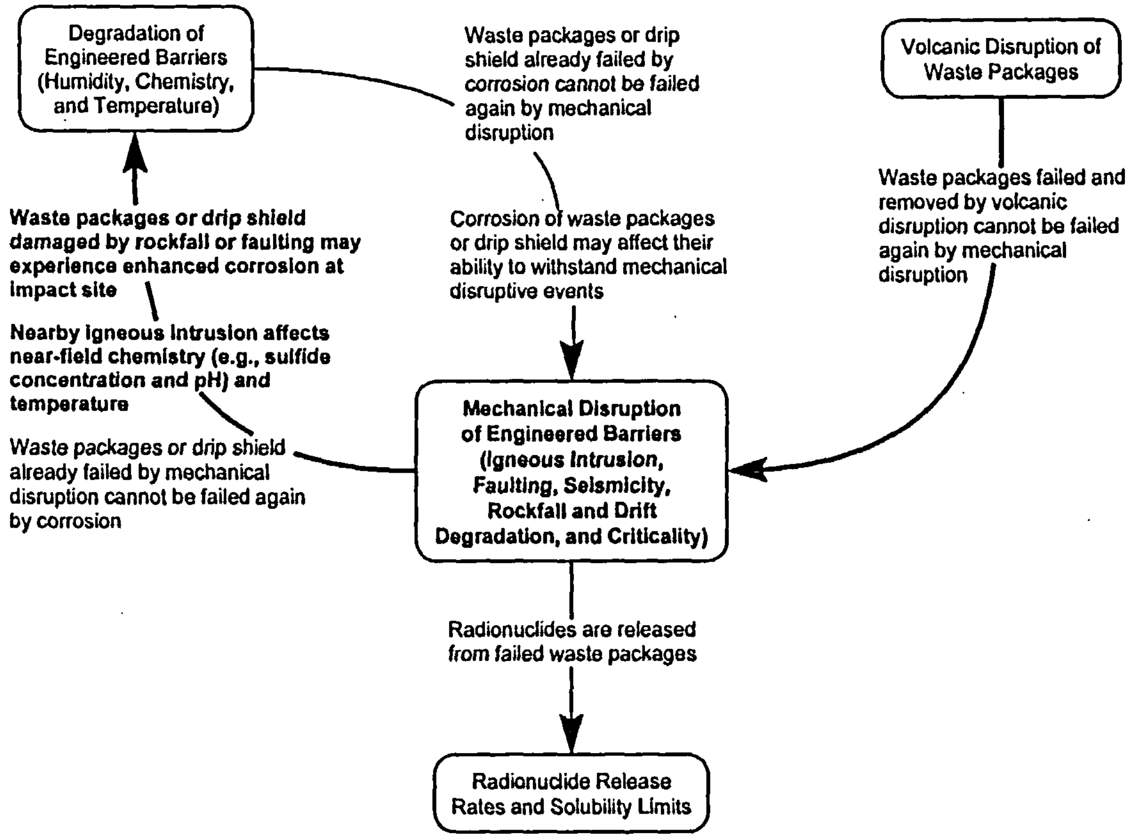

Figure 5.1.3.2-1. Diagram Illustrating the Relationship Between Mechanical Disruption of Engineered Barriers and Other Integrated Subissues. Material in Bold Is Identified in the Text.

component states that correspond to a 10,000-year period. The detailed DOE process-level modeling activities intended to support total system performance sensitivity studies addressing the effects of these disruptive events on repository performance are in various stages of completion. As a result, the scope of the staff review for these disruptive events is typically limited to the documentation of the process-level modeling efforts currently available in the public domain. The scope of the assessment presented here is limited to examining if data gathered and methodologies developed by DOE are likely to be adequately documented for the staff to undertake a detailed technical review of a license application if submitted. This assessment is not a regulatory compliance determination review of a potential license application.

5.1.3.2.2 Relationship to Key Technical Issue Subissues

The Mechanical Disruption of Engineered Barriers Integrated Subissue incorporates subject matter previously described in the following 17 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 Igneous 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 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 would provide to NRC. 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.

5.1.3.2.3 Importance to Postclosure Performance

One aspect of risk informing of the NRC understanding of postclosure repository performance is to determine how this integrated subissue is related to the DOE repository safety strategy (CRWMS M&O, 2000a). Specifically, the DOE repository safety strategy (CRWMS M&O, 2000a) acknowledges mechanical disruption of engineered barriers will affect the long-term risks of the potential repository to the public health and safety. 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).

Risk insights pertaining to the mechanical disruption of engineered barriers indicate that the effects of accumulated rockfall on engineered barriers, and the effects of seismic loading and igneous activity on engineered barriers are of medium significance to waste isolation. The dynamic effects of rockfall (i.e., dynamic impacts caused by discrete rock blocks that have been dislodged from the drift wall) on engineered barriers and the effects of faulting on engineered barriers are assigned low significance. The details of the risk insights ranking are provided in Appendix D of this report. This section also includes an evaluation of the effects of the number waste packages damaged by igneous intrusion, which is of medium significance (Appendix D).

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 potential geologic repository. Future seismotectonic and volcanic activities could affect both the natural and engineered barrier systems of the potential repository.

The DOE model results (Bechtel SAIC Company, LLC, 2002a,b; CRWMS M&O, 2000a) indicate igneous intrusion, faulting, seismicity, and rockfall and drift degradation are natural processes that could cause waste package failures and, thus, result in a dose to the receptor. Analyses used to demonstrate compliance with licensing requirements must factor into the performance calculations the likelihood of a potential disruptive event to determine a probability-weighted dose.

NRC risk insights (Appendix D of this report) based, in part, on total system performance sensitivity analyses (Mohanty, et al., 2002) indicate the disruption of engineered barriers by intrusive igneous activity has a medium significance to total system performance assessment results. A summary of the NRC risk informing process and sensitivity analyses can be found in Appendix D. The medium significance designation arises because the consequences from intrusive igneous activity are directly proportional to the number of waste packages damaged by direct magma flow into potentially intersected drifts. Typical igneous intrusions are on the order of 1–5 km [0.6–3.1 mi] long at potential repository depths (e.g., NRC, 1999a; Bechtel SAIC Company, LLC, 2003a). If drifts are spaced 81 m [266 ft] apart, a typical igneous intrusion could affect approximately 12–46 drifts. Damage to waste packages within each potentially intersected drift likely occurs from the high thermal and mechanical stresses created by basaltic magma (e.g., Bechtel SAIC Company, LLC, 2002b). Although detailed process-level models for these effects have not been developed, available information suggests the current waste package design may not provide the structural characteristics needed to ensure waste isolation after direct contact with basaltic magma (NRC, 1999a; Bechtel SAIC Company, LLC, 2003a).

Igneous intrusion has the potential to fail on the order of a thousand waste packages. The risk from this potential disruptive event is characterized as having medium significance to repository performance because of the low likelihood this event will occur within 10,000 years of permanent closure. Most DOE estimates for the annual probability of igneous disruption at the potential repository site range from 10^{-10} to 10^{-8} (e.g., Bechtel SAIC Company, LLC, 2003a). In contrast, alternative annual probability estimates generally range from 10^{-8} to 10^{-7} (e.g., NRC, 1999a) to values as high as 10^{-6} using Bayesian methods (e.g., Ho and Smith, 1997). None of these probability models, however, has considered current uncertainties in the number and age of past igneous events (e.g., Hill and Stamatakos, 2002). Using a range of alternative conceptual models, Hill and Stamatakos (2002) describe how these uncertainties may have negligible to order-of-magnitude effects on the igneous activity probability estimate. Because the risk from potential igneous intrusion is directly proportional to the probability of igneous activity, these unaccounted for uncertainties may result in negligible to order-of-magnitude effects on current risk estimates. The NRC staff is evaluating additional information provided in Ziegler (2003) to address current concerns regarding consideration of uncertainties in the DOE probability estimate.

Faulting and seismicity, unlike igneous intrusion, are potential disruptive events that have a relatively high likelihood to occur. Presently, faulting is considered to have a low significance on repository performance, however, because operational procedures preclude emplacement of the waste packages within proximity of known faults, and the number of waste packages that could be affected by faulting is relatively low (Section 5.1.3.2.4.2 contains additional discussion). Conversely, multiple seismic events of varying magnitudes are expected to occur.

These seismic events have the potential to affect the near-field environment by way of rockfall and cause damage to all components of the engineered barrier system. Because the magnitude of a seismic event and its corresponding annual frequency of occurrence, or return period, needed to cause sufficient damage to the engineered barrier system such that it will have an effect on repository performance has yet to be clearly established, seismicity has been characterized as having medium significance to repository performance (Section 5.1.3.2.4.3).

Rockfall and drift degradation have the potential to affect repository performance by changing the characteristics of the near-field environment and subjecting the drip shield to discrete rock block impacts, static loads arising from the accumulation of rockfall rubble, or both. As discussed in Section 5.1.3.2.4.4, the occurrence of falling rock blocks of sufficient size to cause appreciable damage to the drip shield is limited to the middle nonlithophysal rock unit of the potential repository. Current information indicates that this particular rock unit only represents approximately 15 percent of the potential repository footprint. As a result, the discrete rock block impact disruptive scenario has been characterized as having low significance to repository performance. The static loads created by rockfall and drift degradation may be sufficient to cause the drip shield to fail by buckling or creep (Section 5.1.3.2.4.4). Failure of the drip shield by either mechanism would cause the accumulated rockfall rubble loads to be transferred to the waste package. It should be noted, the potential failure of the drip shield under these conditions is strongly dependent on its design. In addition, accumulation of rockfall rubble in the drift also will increase the drip shield and waste package temperatures. High temperatures will adversely affect the load-bearing capacity of the drip shield and the waste package, increasing their failure potential. The increased temperature also may accelerate drip shield and waste package corrosion and wasteform dissolution. Because of the foregoing concerns and the current uncertainty associated with the accumulation of rockfall rubble, including its accumulation rate, spatial extent of occurrence, net load magnitudes (including seismic effects), and effects on the engineered barrier system, rockfall and drift degradation have been characterized as having medium significance to repository performance.

For two reasons, criticality also is included in the discussion about Mechanical Disruption of Engineered Barriers Integrated Subissue. The first reason is an in-package criticality event may cause significant mechanical degradation or outright failure of the wasteform 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). Because of its low probability of occurrence, criticality has been characterized as having low significance to repository performance.

In summary, the NRC staff risk insights (Appendix D) characterize the mechanical disruption of engineered barriers by way of igneous intrusion as having medium significance to potential repository performance (Section 5.1.3.2.4.1), faulting as having low significance (Section 5.1.3.2.4.2), seismicity as having medium significance (Section 5.1.3.2.4.3), rockfall and drift degradation as having medium significance (Section 5.1.3.2.4.4), and criticality as having low significance (Section 5.1.3.2.4.5). Assessment of the DOE characterizations and performance assessment abstractions of mechanically disruptive events is conducted at a level of detail commensurate with the assigned degree of significance.

5.1.3.2.4 Technical Basis

NRC developed a plan (NRC, 2003) consistent with the acceptance criteria and review methods found in previous issue resolution status reports. A status assessment of the DOE approaches for including mechanical disruption of engineered barriers in total system performance assessment abstractions is provided in the following subsections. For the sake of clarity, the technical bases for the staff comments will be presented within individual sections for the igneous intrusion (Section 5.1.3.2.4.1) and rockfall and drift degradation (Section 5.1.3.2.4.4) mechanically disruptive events. Each of these subsections, in turn, are organized according to the five review methods identified in Section 2.3: (i) Model Integration (including system description), (ii) Data and Model Justification, (iii) Data Uncertainty, (iv) Model Uncertainty, and (v) Model Support. For the faulting, seismicity, and criticality disruptive events, the technical bases for the staff comments are presented in Sections 5.1.2.2.4.2; 5.1.2.2.4.3 (and 7.4.3.2); and 5.1.2.2.4.4.

5.1.3.2.4.1 Igneous Intrusion

5.1.3.2.4.1.1 Model Integration

Risk insights pertaining to the mechanical disruption of engineered barriers indicate the model for effects of potential igneous intrusion on engineered barrier performance makes a significant contribution to risk calculations for possible radiological releases by hydrological processes. An important component of this model is the response of waste package materials to igneous magmatic conditions.

Engineered Barrier System Performance during Igneous Events: The DOE description of the igneous intrusion abstraction is documented in Bechtel SAIC Company, LLC (2003a). The technical basis for the engineered barrier model abstraction is contained in Bechtel SAIC Company, LLC (2003b), with characteristics of potential igneous events documented in Bechtel SAIC Company, LLC (2003c). In summary, the DOE approach to evaluating potential igneous disruption of waste packages involves several conceptual models. Models for magma ascent and initial interactions with potential repository drifts are discussed in Section 5.1.3.10 of this report.

For the mechanical disruption of engineered barriers abstraction, the DOE models begin with the assumption basaltic magma has filled all drifts directly intersected by an ascending igneous intrusion (i.e., dike) (Bechtel SAIC Company, LLC, 2003a). This approach is different from the model in CRWMS M&O (2000b), which assumed only a limited extent of magma flow into potentially intersected drifts. First-order models for magma flow into potentially intersected drifts (Woods, et al., 2002; Lejeune, et al., 2002) conclude magma likely would flow rapidly into and completely fill available voids in intersected drifts. Thus, the current DOE assumption appears reasonable for potentially intersected drifts rapidly filling with basaltic magma.

DOE currently concludes the combined thermal, mechanical, and chemical effects resulting from potential exposure to basaltic magma are sufficient to damage waste packages, drip shields, and cladding to the extent that no further protection to the wasteform is provided (Bechtel SAIC Company, LLC, 2003b). Thus, all waste packages and associated drip shields and cladding in a drift directly intersected by a potential igneous intrusion are presumed to fail during the igneous event (Bechtel SAIC Company, LLC, 2003b). Previously, DOE restricted this damage zone (i.e., Zone 1) in each potentially intersected drift to affect only three waste packages on either side of the igneous intrusion (CRWMS M&O, 2000b–e). DOE currently defines a second zone, Zone 2, to include only drifts not directly intersected by a potential igneous intrusion. Current Zone 2 was previously referred to as Zone 3 in CRWMS M&O (2000b,e), with the former Zone 2 representing a limited damage zone located more than three waste packages away from the point of intrusion intersection.

Coupled Thermal, Mechanical, and Chemical Effects of Igneous Intrusion: Staff agree the DOE models consider a sufficient range of interrelated processes to support the conclusion the drip shields and waste packages will fail when contacted by basaltic magma. Independent analyses performed by NRC 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, 2000b; Bechtel SAIC Company, LLC, 2003b). The yield stress of Alloy 22 decreases from 310.3 MPa [45 ksi] at room temperature to 189.6 MPa [27.5 ksi] at only 538 °C [1,000 °F] (ASME, 2001). Similarly, the ultimate tensile strength of the alloy decreases from 689.5 MPa [100 ksi] at room temperature to 588.8 MPa [85.4 ksi] at only 538 °C [1,000 °F]. At temperatures consistent with an igneous intrusive event {i.e., approximately 1,100 °C [2,012 °F]}, which is near the melting temperature of Alloy 22 {i.e., approximately 1,360 °C [2,480 °F]} (Haynes International, 1988), the yield stress and ultimate strength of the material are expected to decrease significantly. As a result, Alloy 22 is expected to respond to mechanical loads in a viscoplastic manner when subjected to an igneous intrusive event. The ductility of the alloy is not a function of temperature in the range 25–760 °C [77–1,400 °F] (Haynes International, 1988). A marked decrease in ductility for temperatures above 760 °C [1,400 °F] is not expected for this material. After exposure to temperatures of 760 °C [1,400 °F] for approximately 1,000 hours, 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 would likely increase the susceptibility of the material to mechanical failure as a result of seismic events after the intrusive event.

Additional information indicates Type 316 stainless steel, which is used to construct the waste package inner container, has approximately 30-percent greater thermal expansivity than materials analogous to Alloy 22 (ASME, 2001), which is used to construct the waste package outer container. For the current waste package design, which uses a narrow gap between the inner and outer containers, these differences in thermal expansion will create tensile stresses in the waste package outer container when subjected to magmatic temperatures. Exposure to magmatic temperatures also causes significant gas pressures within the confines of the waste package (CRWMS M&O, 2000b,d; Bechtel SAIC Company, LLC, 2003b). The combined effects of differential thermal expansion and internal gas pressurization likely contribute to waste package failure in basaltic magmatic conditions (Bechtel SAIC Company, LLC, 2003b).

After potential emplacement of basaltic magma in some drifts, dissolved gases will evolve from the cooling magma. Magmatic gasses, such as dilute sulfuric acid, are potentially corrosive to engineered materials (Bechtel SAIC Company, LLC, 2003c), thus, migration of these gasses could affect the performance of engineered materials in drifts not directly intersected by rising magma (i.e., Zone 2). Analyses in Bechtel SAIC Company, LLC (2003b) evaluate the potential migration of magmatic gasses in Zone 2 using first-order advection and diffusion models. These analyses conclude low permeability in the intruding basalt limits the advective flow of gas to within several meters of the intruded drift, and only minor diffusive transport of gasses will occur. Thus, DOE concludes degassing magma will have no significant effect on the performance of engineered materials in drifts not directly intersected by potential basaltic magma (Bechtel SAIC Company, LLC, 2003b). The NRC staff continue to evaluate information presented in Bechtel SAIC Company, LLC (2003a,b) to address concerns regarding potential degassing effects on engineered material performance (Reamer, 2001a).

The presence of natural or engineered backfill may affect the extent of potential magma flow into drifts. Limited intrusion into backfilled drifts, however, could still result in the rapid emplacement of some volume of basaltic magma. In this event, some waste packages may be separated from direct contact with the emplaced magma by backfill or rubble. Nevertheless, during the potential igneous event, basaltic magma will likely cool against this loose rubble and degas. The current DOE models do not consider the possible occurrence of natural or engineered backfill in models for potential magma flow into drifts and assume the only possible obstructions in the potentially intersected drifts are the waste packages and drip shields (Bechtel SAIC Company, LLC, 2003a–c). This approach appears conservative because the potential effects of direct magmatic contact on engineered materials are likely more deleterious than the possible effects of magma separated from the engineered materials by a zone of rock rubble (e.g., Bechtel SAIC Company, LLC, 2003b).

The current DOE model assumes much of the waste from potentially disrupted waste packages will be embedded in basalt (Bechtel SAIC Company, LLC, 2003b). Although few details are provided, DOE suggests that uranous oxide in the waste may alter to a uranyl silicate phase such as soddyite during a potential basaltic intrusive event (Bechtel SAIC Company, LLC, 2003b). DOE does not account for this potential waste alteration effect in the performance calculation, and instead adopts what it believes to be a conservative approach wherein the wasteform is unaffected during a potential igneous intrusive event (Bechtel SAIC Company, LLC, 2003a,b). Following a potential intrusive event, DOE assumes that any inflowing meteoric water alters the uranous oxide in spent nuclear fuel to uranyl oxide hydrates such as schoepite (Bechtel SAIC Company, LLC, 2003d). This assumption is the same as DOE has adopted in its basecase hydrologic release model (Bechtel SAIC Company, LLC, 2003b,d). Since DOE believes the basecase hydrologic release model is conservative, it maintains that this approach is also conservative for the intrusive scenario. Although the DOE analysis has not examined the specific physical conditions likely during basaltic intrusive events and the potential effects on wasteform alteration processes, the DOE basecase assumption for radionuclide solubilities may be reasonably conservative based on rapid schoepite formation with exposure to meteoric water (Bechtel SAIC Company, LLC, 2003b,d). However, it is possible that some radionuclides (e.g., technetium and iodine) may be released during a possible basaltic intrusion more easily than in the basecase, by the alteration or the physical pulverization of spent fuel matrices. The staff continues to evaluate information for the potential effects of igneous temperatures on the formation of transgranular fractures and radionuclide releases from in the wasteform.

In summary, DOE considers available information sufficient to conclude the coupled thermal, chemical, and mechanical effects from possible basaltic magmatism would render ineffective the barrier capabilities of all waste packages, drip shields, and cladding in drifts directly intersected by a potential igneous intrusion event. Although the DOE models do not directly account for physical processes likely to occur during basaltic igneous events, these models are based on an abstracted understanding of the coupled thermal, mechanical, and chemical effects likely to occur during potential intrusive igneous events. Based on this abstraction, the NRC staff views neutralization of engineered barriers on contact with basaltic magma as a reasonable conclusion given current information and model analyses. The staff continues to evaluate numerical models for processes related to degassing effects on engineered materials following potential igneous events. Although the current DOE approach of assuming rapid schoepite formation following a potential igneous event (i.e., nominal scenario model) appears conservative given available information, current DOE models do not explicitly evaluate potential wasteform alteration effects during a possible igneous event.

Overall, the available information is sufficient to expect that the information necessary to assess mechanical disruption of engineered barriers by potential igneous intrusive events, with respect to system description and model integration, will be available at the time of a potential license application.

5.1.3.2.4.1.2 Data and Model Justification

Risk insights pertaining to the mechanical disruption of engineered barriers by potential igneous intrusive events indicate the most important data and model justification needs are those used to support assumptions regarding likely deleterious effects of basaltic magma on engineered materials. Because there are few analogs for the effects of potential igneous events on engineered systems, abstraction of the performance assessment model necessarily will rely on indirect information.

Data Availability and Assumptions Pertaining to Igneous Events: Previous DOE models indicate many waste packages are resilient to damage if directly exposed to basaltic magma during potential igneous intrusion events (e.g., CRWMS M&O, 2000c,d). Currently, DOE concludes the combined thermal, mechanical, and chemical effects resulting from potential exposure to basaltic magma are sufficient to damage waste packages, drip shields, and cladding to the extent no further protection to the wasteform is provided (Bechtel SAIC Company, LLC, 2003b). Thus, all waste packages and associated drip shields and cladding in a drift directly intersected by a potential igneous intrusion are presumed to fail during the igneous event (Bechtel SAIC Company, LLC, 2003b). The NRC staff concludes this approach appears reasonably conservative given current information and uncertainties regarding engineered material response to conditions representative of basaltic igneous events in the Yucca Mountain region.

Limited data are available for engineered material properties at conditions representative of basaltic intrusive events. Basaltic magmas have temperatures approximately 1,100 °C [2,012 °F] (e.g., NRC, 1999a; Bechtel SAIC Company, LLC, 2003c). Magma intrusion can be accompanied by recurring pressure variations on the order of 0.1–10 MPa [14.5–1,450 psi] (e.g., Woods, et al., 2002). Available information for mechanical strength properties for waste package alloys under magmatic conditions indicates significant reductions in strength likely occur (Summers, et al., 1999; Rebak, et al., 2000; Haynes International, 2001; Bechtel SAIC Company, LLC, 2003b). In addition, internal gas pressurization and differential thermal expansion at beyond design temperatures, coupled with the dynamic load of the overlying magma and potential geochemical effects, appear sufficient to breach currently proposed waste packages. Currently available data and first-order models support the DOE conclusion that direct contact with basaltic magma will likely damage all exposed waste packages, drip shields, and cladding to the extent no further protection to the wasteform is provided (Bechtel SAIC Company, LLC, 2003b).

Data about the characteristics of basaltic igneous intrusions appear sufficient to support models that evaluate potential effects of igneous conditions on engineered barriers. Magma temperature and compositional information relevant to evaluating potential effects on engineered barriers appear consistently defined and used in Bechtel SAIC Company, LLC (2003b,c). Minor differences between this information and data presented in NRC (1999a) do not appear to affect risk calculations significantly. Basaltic igneous features have been characterized sufficiently to support the DOE evaluations of the potential effects of these features on engineered barriers.

Information on potential wasteform alteration effects in basaltic igneous environments is not readily available (Bechtel SAIC Company, LLC, 2003b). DOE has used generalized relationships to evaluate possible wasteform alteration during a basaltic intrusion, and suggests that potential alteration will not enhance solubility or radionuclide release. DOE's approach assumes potential igneous events do not alter the wasteform, and considers postevent solubility and transport to be the same as in the nominal scenario (Bechtel SAIC Company, LLC, 2003b).

In summary, it appears that sufficient data are available to support conceptual models of engineered barrier response to the physical, chemical, and thermal conditions representative of potential igneous events in the Yucca Mountain region. When direct information is not available to support the model abstraction, transparent assumptions are used to develop a reasonable approach in the evaluation of potential igneous effects on engineered barriers.

Overall, the available information is sufficient to expect that the information necessary to assess mechanical disruption of engineered barriers by potential igneous intrusive events, with respect to data being sufficient for model justification, will be available at the time of a potential license application.

5.1.3.2.4.1.3 Data Uncertainty

Risk insights pertaining to the mechanical disruption of engineered barriers by potential igneous intrusive events indicate the most important data uncertainty needs relate to support for assumptions regarding likely deleterious effects of basaltic magma on engineered materials.

Data Variability and Uncertainty Pertaining to Igneous Events: Previous DOE models indicate many waste packages are resilient to damage if directly exposed to basaltic magma during potential igneous intrusion events (e.g., CRWMS M&O, 2000c,d). Currently, DOE evaluates a range of information to consider the potential effects of basaltic magma on engineered barrier performance (Bechtel SAIC Company, LLC, 2003a–c). Although an explicit process model is not developed, DOE appears to consider an appropriate range of uncertainties in the characteristics of basaltic igneous events and the properties of engineered materials. DOE concludes the combined thermal, mechanical, and chemical effects resulting from potential exposure to basaltic magma are sufficient to damage waste packages, drip shields, and spent nuclear fuel cladding to the extent no further protection to the wasteform is provided (Bechtel SAIC Company, LLC, 2003b). Thus, all waste packages and associated drip shields and spent nuclear fuel cladding in a drift directly intersected by a potential igneous intrusion are presumed to fail during the igneous event (Bechtel SAIC Company, LLC, 2003b). This deterministic approach appears reasonable given current information and uncertainties regarding the response of the engineered barrier materials to conditions representative of basaltic igneous events in the Yucca Mountain region.

DOE uses an advective-diffusive process model to evaluate the potential migration of magmatic gasses from a drift intersected by magma to an adjacent, nonintersected drift (Bechtel SAIC Company, LLC, 2003b). The generalized advection model relies on a critical assumption regarding the extremely low effective permeability of the potentially intruded basalt. Although Bechtel SAIC Company, LLC (2003b) cites permeability information from analog basaltic intrusions, uncertainties in these data do not consider the likely effects on basalt permeability arising from interactions between flowing magma and engineered systems in a 5.5-m [18-ft]-diameter drift. Joints and fractures develop in cooling magmas in response to cooling rate and orientation to cooling surfaces (e.g., DeGraff and Aydin, 1993). Potential repository drifts containing waste packages, supports, and drip shields present multiple cooling surfaces for possible basaltic magma, relative to the simple cooling geometries in analog intrusions. Analog intrusion sites such as Paiute Ridge, Nevada, represent much larger volumes of magma than could potentially fill a drift (Bechtel SAIC Company, LLC, 2003c). Although a magma-filled drift is modeled as cooling to ambient temperatures within approximately 30 years (Bechtel SAIC Company, LLC, 2003b), the Paiute Ridge intrusion likely remained hundreds of degrees above ambient temperatures for at least 200–300 years following intrusion (Ratcliff, et al., 1994). These significant differences in cooling rate and surface orientations likely result in significant differences in fracture abundance between analog sites and potential intrusions in drifts. Based on these effects, intrusion permeabilities could be similar to host rock permeabilities {i.e., on order of 10^{-12} m^2 [10^{-11} ft^2] (Rosseau, et al., 1999)} rather than on order of 10^{-17} m^2 [10^{-16} ft^2] used in the DOE models (Bechtel SAIC Company, LLC, 2003b). A large increase in permeability could significantly affect results of the advective gas-flow model in Bechtel SAIC Company, LLC (2003b).

DOE uses generalized relationships to evaluate possible waste alteration reactions (Bechtel SAIC Company, LLC, 2003b). Nevertheless, DOE does not propagate the results of these relationships into the performance calculations. The NRC staff continues to evaluate whether DOE uses a reasonably conservative approach by assuming potential igneous events do not alter the wasteform and modeling postevent solubility and transport as in the nominal scenario (Bechtel SAIC Company, LLC, 2003b). Thus, the effects of uncertainties in the underlying data are adequately addressed through adoption of a reasonable approach.

In summary, most data ranges derived from basaltic igneous systems appear to adequately represent the uncertainty and variability in the characteristics of potential future igneous events in the Yucca Mountain region. Additional information should be provided, however, to consider a more realistic range of uncertainty in rock permeability for advective gas-flow calculations. This information need is identified in existing Igneous Activity Key Technical Issue agreements (Reamer, 2001a). Uncertainties in engineered materials appear adequately considered in the analyses of potential igneous effects on barrier capabilities.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.2.4.6), is sufficient to expect that the information necessary to assess mechanical disruption of engineered barriers by potential igneous intrusive events, with respect to data uncertainty being characterized and propagated through model abstraction, will be available at the time of a potential license application.

5.1.3.2.4.1.4 Model Uncertainty

Risk insights pertaining to the mechanical disruption of engineered barriers by potential igneous intrusive events indicate the most important model uncertainty needs relate to support for conclusions regarding likely deleterious effects of basaltic magma on engineered materials.

Consideration of Igneous Intrusion Model Uncertainty: Previous DOE models indicate many waste packages are resilient to damage if directly exposed to basaltic magma during potential igneous intrusion events (e.g., CRWMS M&O, 2000c,d). Currently, DOE evaluates a range of information to consider the potential effects of basaltic magma on engineered barrier performance (Bechtel SAIC Company, LLC, 2003a–c). Although an explicit process-level model is not developed, DOE appears to consider an appropriate range of conceptual models for evaluating the potential effects of basaltic igneous events on engineered barrier performance. DOE concludes the combined thermal, mechanical, and chemical effects resulting from potential exposure to basaltic magma are sufficient to damage waste packages, drip shields, and spent nuclear fuel cladding to the extent no further protection to the wasteform is provided (Bechtel SAIC Company, LLC, 2003b). Thus, all waste packages and associated drip shields and spent nuclear fuel cladding in a drift directly intersected by a potential igneous intrusion are presumed to fail during the igneous event (Bechtel SAIC Company, LLC, 2003b). This deterministic approach appears reasonable given current information and model uncertainties regarding engineered material response to conditions representative of basaltic igneous events in the Yucca Mountain region.

Models used to evaluate the migration of magmatic gasses from potentially intersected drifts used several reasonable assumptions regarding parallel gas flow and no reaction between magmatic gasses and surrounding wall rock (Bechtel SAIC Company, LLC, 2003b). DOE has not provided a traceable basis, however, to conclude uncertainty in the potential intrusion permeability is offset by other, similarly reasonable assumptions (Bechtel SAIC Company, LLC, 2003b). Thus, the model for migration of magmatic gasses may not account for uncertainty in the potential contributions from advective transport processes because of significant underestimation of host rock effective permeability.

Alternative Conceptual Models of Igneous Intrusion: DOE considered several alternative conceptual models in the evaluation of potential igneous intrusive effects on engineered barriers (Bechtel SAIC Company, LLC, 2003b). Additional analyses are conducted for potential basalt-wasteform alteration processes, reactions between basalt and corrosion products, effects of alteration in a localized zone around potentially intruded drifts, and fragmentation effects on the wasteform (Bechtel SAIC Company, LLC, 2003b). Each alternative conceptual model is judged to have large uncertainties, thus, the potentially conservative effects of these processes are not adopted in the performance model (Bechtel SAIC Company, LLC, 2003b). The NRC staff continues to evaluate this assessment.

In summary, uncertainty in the underlying conceptual models appears adequately considered in the evaluation of potential igneous intrusive effects on engineered barriers. Although the model uncertainties are not quantified, results of these uncertainties are used to justify reasonable assumptions regarding degradation of engineered barrier performance during basaltic igneous events. In addition, DOE appears to have considered an appropriate range of alternative conceptual models. Although these alternative conceptual models would likely enhance the performance characteristics of some engineered systems during potential igneous events, results of these alternative conceptual models are not used to reduce conservatism in the DOE performance assessment.

Overall, the available information is sufficient to expect that the information necessary to assess mechanical disruption of engineered barriers by potential igneous intrusive events, with respect to model uncertainty being characterized and propagated through model abstraction, will be available at the time of a potential license application.

5.1.3.2.4.1.5 Model Support

Risk insights pertaining to the mechanical disruption of engineered barriers by potential igneous intrusive events indicate the most important information needs for model support relate to conclusions regarding the number of waste packages potentially affected by an igneous intrusive event.

Consistency between Process-Level and Abstracted Igneous Intrusion Models: Rather than develop a series of detailed models to evaluate complex magma-engineered barrier interaction processes, DOE uses first-order models to support several apparently conservative conclusions regarding degradation of engineered barrier performance during potential igneous events. DOE concludes the combined thermal, mechanical, and chemical effects resulting from potential exposure to basaltic magma are sufficient to damage waste packages, drip shields, and spent nuclear fuel cladding to the extent no further protection to the wasteform is provided (Bechtel SAIC Company, LLC, 2003b). Thus, all waste packages and associated drip shields and spent nuclear fuel cladding in a drift directly intersected by a potential igneous intrusion are presumed to fail during the igneous event (Bechtel SAIC Company, LLC, 2003b). The NRC staff conclude this approach appears reasonably conservative given current information regarding engineered material response to conditions representative of basaltic igneous events in the Yucca Mountain region. Because DOE adopted a reasonable approach to evaluating the engineered barrier response to potential igneous events, additional model support is not warranted (i.e., Reamer, 2001a).

A similar approach is adopted by DOE to evaluate possible wasteform alteration effects during a potential igneous intrusive event. DOE concludes that, although wasteform alteration processes are possible, these processes would not result in a wasteform more soluble than currently is assumed to occur during nominal performance scenarios (Bechtel SAIC Company, LLC, 2003b). Thus, DOE neglects the potential effects of wasteform alteration during possible igneous intrusive events and currently assumes waste exposed to meteoric water following the igneous event alters to a relatively soluble form (i.e., schoepite) as in the nominal scenario calculations. The NRC staff continues to evaluate information in Bechtel SAIC Company, LLC (2003b) to support assumptions regarding rapid alteration of the wasteform to schoepite during the nominal scenario conditions.

Calculations involving potential migration of magmatic gasses use basic advection-diffusion relationships for ideal gasses. The NRC staff considers these relationships suitable to evaluate potential migration of magmatic gasses into wall rock and drifts adjacent to potentially intersected drifts. The conservatism in this approach, however, cannot be readily evaluated until additional calculations are performed using a more realistic range of host rock permeabilities.

In summary, the DOE models for the effects of potential igneous activity on engineered barriers provide results that appear to be reasonable. These model results appear consistent with empirical observations and simple extrapolations from available data.

Overall, the available information is sufficient to expect that the information necessary to assess mechanical disruption of engineered barriers by potential igneous intrusive events, with respect to model abstraction output being supportive by objective comparisons, will be available at the time of a potential license application.

5.1.3.2.4.2 Faulting

Details of the DOE approach to faulting hazard assessment along with staff evaluation of the DOE information are provided in Section 4.1.1 of this report. Staff evaluation of faulting effects on engineered barriers and postclosure performance is provided in Section 5.1.2.2.4.2 of this report. A review of the DOE analyses, coupled with risk insights gained from an independent consequence analysis of faulting (Stamatakos, et al., 2003), indicates that DOE has assembled sufficient information about direct faulting in the prelicensing period for NRC to conduct a review of a potential license application if submitted.

Overall, the available information is sufficient to expect that the information necessary to assess the probability of faulting affecting the repository system will be available at the time of a potential license application. Therefore, the staff considers the faulting subissue to be closed, as defined within the Structural Deformation and Seismicity Key Technical Issue.

5.1.3.2.4.3 Seismicity

Details of the DOE approach to seismic hazard assessment along with staff evaluation of the DOE information are provided in Section 4.1.1 of this report. Staff evaluation of seismicity is provided in Sections 5.1.2.2.4.3 and 7.4.4.2 of this report. DOE indicated (Section 5.1.2.2.4.3) it is currently changing its approach to the development of seismic inputs in total system performance assessment consequence modeling of seismicity for postclosure performance assessment. DOE has not yet provided documentation of these changes and thus, staff cannot evaluate the potential effects of seismicity on the engineered barrier system. Because DOE has indicated that these changes may be substantial, staff cannot determine at the present time whether sufficient information will be available at the time of a potential license application for the staff to begin its technical review.

5.1.3.2.4.4 Rockfall and Drift Degradation

5.1.3.2.4.4.1 Model Integration

The characterization of potential rockfall and drift degradation during the postclosure period is important for several reasons. First, individual rock blocks large enough to cause mechanical damage may strike the drip shield several times during the postclosure period. Second, sustained mechanical loading from accumulated rockfall rubble may cause mechanical damage to the drip shield and, possibly, the waste package if mechanical interactions with the drip shield were to occur. Third, if a sufficient amount of rubble accumulates in the openings early enough to affect heat flow, the insulating effect may cause an increase in the temperature of the engineered barrier system components. Such an increase in temperature may cause the load-bearing capacity of the drip shield to decrease and may affect the near-field environmental parameters relevant to the long-term performance of the engineered barrier system. Fourth, the presence of rockfall rubble in the openings may affect the potential for seepage water contacting the engineered barrier system components. The DOE analysis indicates only a small percentage of the emplacement drifts would intersect a rock type that is likely to produce discrete rock blocks of sufficient size to damage the drip shield through dynamic rock-block impact[1] (Bechtel SAIC Company, LLC, 2003e). Consequently, the effects of accumulated rockfall rubble are likely to be more important to repository performance than the effects of individual rock blocks striking the drip shield. Risk insights pertaining to the mechanical disruption of engineered barriers by rockfall and drift degradation indicate that the effects that these loads may have on the engineered barrier system are strongly dependent on the magnitude and time of occurrence of the accumulation of rockfall rubble. Important aspects of this disruptive scenario are (i) ability of the drip shield to protect the waste package from the accumulated rockfall rubble loads that arise under static and seismic conditions and (ii) potential changes to the near-field environment, including temperature, relative humidity, and seepage water chemistry. DOE has not provided any abstracted model to include the effects of accumulated rockfall rubble in performance assessment, but has presented information based on its drift degradation analysis (Bechtel SAIC Company, LLC, 2003e) indicating the amount of rockfall rubble that may occur under the anticipated repository conditions would be small.

Use of Results from Rockfall Modeling in Postclosure Performance Assessment: DOE identified nine rock blocks to be considered in assessing discrete rock-block impact on the drip shield (Bechtel SAIC Company, LLC, 2003e, p. 160 and Attachment IX). The mass of the blocks varies in the range 6.3–21.4 tonnes [2.8–9.7 kip], which DOE indicates is greater than a design-basis mass of 6 tonnes [2.7 kip] based on an earlier DOE study (Bechtel SAIC Company, LLC, 2003e, p. 160). DOE has not provided any discussion of the probability of a drip shield impact by the selected blocks, how such probability would relate to results from rockfall modeling, or a basis for not considering potential drip shield impacts from blocks smaller or larger than the nine selected blocks.

Degraded drift-perimeter profiles also are provided by DOE (Bechtel SAIC Company, LLC, 2003e, Attachment XVIII). The profiles are intended to be used to evaluate the potential effects of rockfall on seepage. Any accumulated rockfall rubble in the drifts is not included when defining the profiles, and the basis for not including such rubble also is not provided. Furthermore, the potential effects of rockfall rubble on the engineered barrier system temperatures (e.g., Fedors, et al., 2004) are not included in the discussion of potential uses for the degraded-drift profiles.

A characterization of the potential static loads that may be imposed on drip shields from accumulated rockfall rubble is provided in Bechtel SAIC Company, LLC (2003e, pp.180, 196, and Attachment XVI). The characterization consists of average pressures on the top and vertical surfaces of a rigid surface representing the drip shield and bar charts describing pressure distributions on the 3 surfaces, based on dividing each surface into 10 segments. It is not clear at this time how the DOE intends to use the calculated pressure distributions to establish the design basis for the drip shield. For example, the drip shield design basis could be

[1]Board, M.P. "Mechanical Drift Degradation Analysis." *Presentation to Advisory Committee on Nuclear Waste, November 19, 2003.* Las Vegas, Nevada. 2003.

the pressure distributions as calculated, the average or maximum pressures derived from these distributions, or some other characterization of the rockfall rubble pressure. Furthermore, DOE compared the drip-shield pressures calculated from the UDEC-Voronoi model, which is discussed in more detail later, with larger pressures calculated using an analytical approach based on the bulking behavior of broken rock (Bechtel SAIC Company, LLC, 2003e, p. 199). DOE argued the analytical approach is overly conservative because it does not account for parts of the rubble supporting their own weight through arching. The occurrence of arching in a rubble pile and the fraction of rubble weight that may be self-supported through arching depend on the distribution of particle sizes and shapes within the rubble. The sizes and shapes of particles in the UDEC-Voronoi model are not selected to match any structural features of the modeled rock mass (Bechtel SAIC Company, LLC, 2003e, p. 141). Consequently, the internal structure of a rock rubble pile generated in the model does not represent the internal structure of a rock rubble pile that may occur at Yucca Mountain. The model, therefore, would not be appropriate for calculating the value of a property or behavior that is controlled by the internal structure of rubble. In order to justify taking credit for arching to reduce the amount of rock-rubble load considered in the design or performance analysis of the drip shield, DOE should provide information on how the amount of arching calculated from the UDEC-Voronoi model is used to determine the amount of arching that may develop in a lithophysal-rock rubble pile.

Combined Effects of Seismicity and Accumulated Rockfall Rubble Loads: An assessment of the potential effects of accumulated rockfall rubble loads on the drip shield has not been provided by DOE. These effects include, but may not be limited to (i) changes to the dynamic response of the drip shield when subjected to seismic excitation, (ii) structural buckling of the drip shield, and (iii) creep failure of the drip shield materials (Section 5.1.3.2.4.4.2). Those drip shields that may not have buckled under static conditions, may do so during a seismic event. If the drip shield does not buckle outright, additional plastic deformations under combined seismic and accumulated rockfall rubble loads may be sufficient to transfer the rockfall rubble loads from the drip shield directly to the waste package (Gute, et al., 2003).

Corrosion and Material Degradation Processes: In addition to temperature effects, various corrosion and material degradation processes that could influence the ability of the drip shield to withstand seismic and rockfall and drift degradation loads are (i) wall thinning as a result of enhanced uniform corrosion of titanium in the presence of large amounts of fluoride, and (ii) hydrogen entry and concentration in the titanium metal alloys. To address the effects of uniform corrosion, some of the drip shield structural analyses have been performed using reduced thicknesses for those components with surfaces that may be exposed to fluoride. Because of the uncertainty, however, of the fluoride availability, supply, and concentrations, it is not clear whether the reduced thickness value used in the analyses sufficiently accounts for the potential effects of this corrosion process. The titanium drip shield may also be susceptible to hydrogen-induced cracking through absorption of hydrogen generated from galvanic coupling of titanium with carbon steel components of the invert or gantry rail (CRWMS M&O, 2000f). The proposed use of perforated stainless steel sheets and bolts for the ground support materials, such as steel mesh and steel rock bolts, will reduce significantly the possibility of galvanic coupling leading to hydrogen entry into the titanium drip shield. Because cathodic hydrogen entry is coupled to the uniform corrosion of the drip shield, any factor increasing the corrosion rate may increase the hydrogen uptake. Uptake of hydrogen above a critical concentration will result in a substantial decrease in the ductility of the titanium alloys. The occurrence of hydrogen uptake into the drip shield titanium alloys, however, is not expected to be spatially extensive throughout the potential repository.

As with the drip shield, some DOE analyses of the waste package subjected to mechanically disruptive events are performed using an Alloy 22 outer container that has been reduced in thickness to account for the potential occurrence of general corrosion. The calculated reductions in thickness are based on corrosion rates measured in immersion tests conducted for a period of up to five years. Because the corrosion rate decreases with time, the projected loss of thickness based on immersion tests is likely to be conservative. The only other corrosion related process that could affect the structural capabilities of the waste package that has been identified at this time is stress corrosion cracking, which is discussed in more detail in the following subsection and Section 5.1.3.1 of this report.

See Section 5.1.3.1 of this report for additional information pertaining to the various corrosion and material degradation processes relevant to the potential geologic repository.

According to Bechtel SAIC Company, LLC (2002a), specific information to be developed to support the DOE total system performance assessment of the seismic scenario includes consideration of the responses of the drip shield, waste package, and spent nuclear fuel cladding as functions of levels of ground motion, rockfall, and fault displacement for degraded component states that correspond to a 10,000-year period, based on detailed structural response calculations. In addition, staff continue to review the DOE efforts to assess the effects rockfall and drift degradation may have on the engineered barrier system, near-field environment, and on the quantity and chemistry of water.

While some information on the mechanical disruption of the engineered barriers (i.e., rockfall and drift degradation) with respect to the system description and model integration may be available at the time of a potential license application, the staff is currently of the view that DOE should provide additional information on the DOE calculations of the amount of arching that may occur in a lithophysal rubble pile, and on the potential effects of an accumulated rubble load on the drip shield.

5.1.3.2.4.4.2 Data and Model Justification

Risk insights pertaining to the mechanical disruption of engineered barriers by rockfall and drift degradation indicate the most important data and model justification needs pertain to the consistency of the engineered barrier system design criteria with process-level models used to demonstrate the seismic and rockfall and drift degradation design basis loads.

Representation of Repository Host Rock in Rockfall Modeling: DOE developed three modeling approaches for rockfall that differ on the basis of the rock-mass characteristics represented in the model and the type of output information expected from calculations performed using the model (Bechtel SAIC Company, LLC, 2003e). The repository host rock consists of two lithological zones: lithophysal and nonlithophysal. The lithophysal zone (i.e., lower and upper lithophysal stratigraphic units) is expected to constitute approximately 85 percent of the emplacement drifts. The nonlithophysal zone (i.e., middle and lower nonlithophysal stratigraphic units) makes up the remaining 15 percent[2] (Bechtel SAIC Company, LLC, 2003e).

The nonlithophysal rock generally consists of fractured hard rock. Mechanical behavior of the nonlithophysal rock is dominated by relative movements of discrete rock blocks on fracture surfaces and some fracturing of the blocks. DOE developed a three-dimensional model of the nonlithophysal rock focused on analyzing the motions of interconnected rock blocks (Bechtel SAIC Company, LLC, 2003e). The model was developed for characterizing potential discrete rock blocks that may strike the drip shield. The model is referred to hereafter as the 3DEC model because calculations based on the model were performed using the computer program 3DEC (Bechtel SAIC Company, LLC, 2003e). Fractures in the rock mass are represented in the model by taking statistical fracture-geometry samples from a stochastically generated three-dimensional fracture model representing the nonlithophysal rock mass. The distribution of discrete rock blocks in the 3DEC model, therefore, represents the distribution of discrete rock blocks in the nonlithophysal rock mass. The three-dimensional fracture model was developed using the computer code FracMan (Bechtel SAIC Company, LLC, 2003e).

The FracMan simulations are conditioned with a subset of the fracture data collected from detailed line surveys in the Exploratory Studies Facility and Enhanced Characterization of the Repository Block cross drift. DOE has not demonstrated the synthetic fracture distributions are statistically similar to the detailed line survey data or the synthetic distributions represent the fracture distribution in the nonlithophysal rock. The spread in fracture orientations in the rock mass is underrepresented in the DOE synthetic fracture populations because of the value of the *dispersion coefficient used in the FracMan calculation. DOE uses a dispersion value of 70 in* the FracMan simulations to describe the orientation variation of the steeply dipping fractures; however, a confirmatory analysis of the DOE fracture data performed by the staff indicates the dispersion values should be no higher than 30. The high dispersion value used by DOE leads to synthetic fracture populations with a low variation in fracture orientation and contributes to an unrealistic representation of fracture intersections in the 3DEC models. Also, the intensity of gently dipping (i.e., subhorizontal) fractures is underrepresented in the three-dimensional fracture model because DOE does not correct the fracture data for sampling bias. DOE uses an uncorrected spacing of 4.2 m [13.8 ft] for the low-angle fractures, whereas a confirmatory staff analysis of the DOE fracture data indicates a corrected spacing of approximately 0.5 m [1.6 ft] for the fractures. As a result, frequency of the subhorizontal fractures is underrepresented by a factor of eight in the FracMan-generated synthetic fracture population used as input for the 3DEC modeling described in Bechtel SAIC Company, LLC (2003e). This underrepresentation of the subhorizontal fractures results a in low occurrence of the potential release planes in the 3DEC rockfall models. Furthermore, the DOE detailed line survey data suggest that the majority of fractures in the nonlithophysal units terminate against other discontinuities (i.e., fractures or layer interfaces) rather than ending blindly within rock blocks as suggested by the DOE interpretations of the same data. Staff analyzed the detailed line survey data and found only 25 percent, approximately, of the more than 11,000 fractures in the middle nonlithophysal interval exhibit a blind terminator at one end. Furthermore, less than 3 percent of these fractures exhibited blind terminations at both ends. The 3DEC models include a large number of rock bridges representing blind terminations of fractures within rock blocks. These

[2]Board, M.P. "Mechanical Drift Degradation Analysis." *Presentation to Advisory Committee on Nuclear Waste, November 19, 2003.* Las Vegas, Nevada. 2003.

concerns regarding representation of the DOE fracture data in the FracMan analysis need to be addressed by DOE to provide information needed for staff assessment of the 3DEC rockfall models.

The lithophysal rock generally consists of an essentially unfractured upper lithophysal or intensely fractured lower lithophysal ground mass with lithophysal cavities of various sizes and shapes. The DOE analysis (Bechtel SAIC Company, LLC, 2003e) indicates fractures and cavities do not form planes of weakness that follow a preferred orientation. DOE concludes, therefore, mechanical behavior of the lithophysal rock mass can be represented as an isotropic continuum. DOE developed a model of the lithophysal rock mass that consists of an assemblage of randomly oriented polygonal blocks (Bechtel SAIC Company, LLC, 2003e). This model is referred to hereafter as the UDEC-Voronoi model because calculations based on the model were performed using the computer program UDEC with a Voronoi-tessellation algorithm for generating polygonal blocks (Bechtel SAIC Company, LLC, 2003e). The blocks can slide on or separate from each other, which allows the model mass to break up or deform in a way that potentially can simulate rock deformation, fracturing, and rockfall in an over-stressed rock mass. Geometry, strength, and stiffness of the blocks and block interfaces are not determined by matching any similar properties of the lithophysal rock. The block and block-interface properties, instead, are assigned values such that the macroscopic strength and stiffness of the assemblage match the measured strength and stiffness of the lithophysal rock mass. The UDEC-Voronoi model is used by DOE to determine the occurrence and magnitude of rockfall and drift degradation in lithophysal rocks (Bechtel SAIC Company, LLC, 2003e). A confirmatory analysis performed by the staff of the DOE fracture data for the lower lithophysal unit indicates the fracture orientations are not random, but, rather, are dominated by a northwest-southeast striking, subvertical set and a subhorizontal set. Two other subvertical fracture sets are present, although these are less well developed. The fracture spacing is generally small and highly variable, which indicates the rock can be expected to form small blocks. The occurrence of at least four fracture sets, variability of fracture orientations within each set, small average fracture spacing, and large variability of spacing imply the mechanical behavior of the rock mass at the scale of the emplacement drifts would be dominated by a high density of variously oriented weakness planes. Such a behavior can be represented reasonably as an isotropic continuum, as illustrated in Hoek and Brown (1980, pp. 132 and 165).

DOE also developed a UDEC-Voronoi model to calculate the magnitude of drift degradation and accumulated rockfall rubble in nonlithophysal rocks (Bechtel SAIC Company, LLC, 2003e, pp. 140–144). DOE explained this analysis is performed to complement the 3DEC model analysis of wedge-type rockfall. The 3DEC model calculations are used to characterize potential discrete rock blocks that may strike the drip shield, whereas the UDEC-Voronoi model is used to calculate the extent of drift degradation and amounts of rockfall rubble. Use of the UDEC-Voronoi model for this purpose is necessary because the DOE information indicates tensile and compressive stresses generated by seismic waves during an earthquake of 10^{-6} or lower probability would exceed the rock strength (Bechtel SAIC Company, LLC, 2003e, pp. 99 and 140). The UDEC-Voronoi model has the capability to represent potential internal fracturing of rock blocks, which is not represented in the 3DEC model. The UDEC-Voronoi model, however, does not account for preexisting fractures. Because both relative movements of rock blocks on preexisting fractures and internal fracturing of the blocks contribute to rockfall and drift degradation in nonlithophysal rocks, DOE should provide information to explain how results of the 3DEC and UDEC-Voronoi models would be used to characterize the amount of rockfall and extent of drift degradation in the nonlithophysal rock units.

DOE developed a model representing an intensely fractured zone within the nonlithophysal zone (Bechtel SAIC Company, LLC, 2003e, p. 148). Such an intensely fractured zone is encountered along an approximately 1,000-m [3,281-ft] length of the Exploratory Studies Facility tunnel and was observed to be dominated by closely spaced northwest-striking subvertical fractures. The DOE model for the intensely fractured zone consists of a three-dimensional continuum model with directional fracture properties blended with isotropic rock properties to obtain a transversely isotropic rock mass. The model, referred to hereafter as the ubiquitous joints model, is used to analyze the seismic response of the intensely fractured zone. The analysis, however, is not used to draw any conclusions regarding rockfall or drift degradation. Furthermore, an analysis of the thermal-mechanical response of the intensely fractured zone was not performed.

In summary, DOE has developed three model representations of the repository host rock for use in rockfall calculations. First, the 3DEC model was developed for characterizing discrete rock blocks that may strike the drip shield. This model could, in principle, be used to perform such calculations, however, several concerns need to be addressed (as discussed in this section) before staff can review the 3DEC model used for the calculations (Bechtel SAIC Company, LLC, 2003e). The 3DEC model would not be appropriate for calculating the extent of drift degradation or the amount of accumulated rockfall rubble because the potential fracturing of rock blocks, which has a strong effect on both quantities, is not represented in the model. This limitation of the 3DEC model is acknowledged by DOE (Bechtel SAIC Company, LLC, 2003e, pp. 140–141); however, the degraded-drift profiles provided by DOE to characterize drift degradation in nonlithophysal rock (Bechtel SAIC Company, LLC, 2003e, pp. 161–165) are calculated using the model. These drift profiles are inappropriate for characterizing potential drift degradation because the model used to calculate the profiles does not account for rock-failure processes that control the extent of drift degradation. Second, the UDEC-Voronoi model was developed for calculating the extent of drift degradation and the amount of accumulated rockfall rubble in lithophysal and nonlithophysal rocks. This model could, in principle, be used to perform such calculations; however, several concerns need to be addressed (as discussed subsequently in this section) before staff can review estimates of the extent of drift degradation and amounts of accumulated rockfall rubble provided by DOE (Bechtel SAIC Company, LLC, 2003e). Third, the ubiquitous joints model was developed for representing the intensely fractured zone, but the model is not used to support any evaluation of rockfall or drift degradation. DOE, therefore, has not provided any rockfall assessment for the intensely fractured zone. According to Mongano, et al. (1999), the intensely fractured zone observed in the Exploratory Studies Facility main drift is not expected to be present in the Enhanced Characterization of the Repository Block cross drift. Recent seismic tomography studies by Gritto, et al. (2004), however, suggest several sources of very high fracture density {i.e., 5–6 m^{-3} [0.1–0.2 ft^{-3}]} may be present in the current repository footprint. Gritto, et al. (2004) conclude the intensely fractured zone extends to the west-southwest from the Exploratory Studies Facility through the repository horizon (i.e., panel 2). A rockfall assessment should be provided for any drifts located in the intensely fractured zone.

Rockfall Model Geometry and Boundary and Initial Conditions: The 3DEC model consists of a cube approximately 25 m [82 ft] on each side with a 5.5— [18-ft]-diameter tunnel approximately centered (Bechtel SAIC Company, LLC, 2003e, pp. 93–97). The fractured-rock zone (i.e., the model zone representing the host rock) extends to two tunnel diameters above the tunnel and one tunnel diameter to the sides. The boundary conditions for static and thermal-mechanical analyses are zero boundary-normal displacement on all exterior boundaries. Different boundary conditions were used for dynamic analysis as explained in Bechtel SAIC Company, LLC (2003e, p. 96). The UDEC-Voronoi model consists of a rectangle 81 m [266 ft] wide and 35 m [115 ft] high with a centered 5.5— [18-ft]-diameter circular opening that represents the emplacement drift. The 81-m [266-ft] width is consistent with the center-to-center spacing between emplacement drifts in the DOE proposed design. The boundary conditions for static and thermal-mechanical analyses (Bechtel SAIC Company, LLC, 2003e, p. 180) are zero normal displacements on the lateral boundaries to represent symmetry, zero vertical displacement at the base {i.e., 17.5 m [57.4 ft] below the tunnel axis}, and fixed traction representing the initial *in-situ* stress at the top {i.e., 17.5 m [57.4 ft] above the tunnel axis}.

The 3DEC and UDEC-Voronoi models are inappropriate for calculating thermally induced rockfall. Because the temperature change zone and the thermal-mechanical perturbation are not encapsulated by either model, the zero-perturbation conditions specified at the model boundaries are inappropriate. The only exceptions are the lateral boundaries of the UDEC-Voronoi model where symmetry boundary conditions are specified consistent with the proposed drift layout. DOE performed an additional thermal-mechanical analysis using a three-dimensional continuum model, referred to hereafter as FLAC3D model (Bechtel SAIC Company, LLC, 2003e, p. III–19), which, for example, indicates displacements on the order of centimeters at depths of several hundred meters below the emplacement drifts. The zone of potential thermal-mechanical perturbation around an emplacement drift would grow with time as indicated by the FLAC3D analysis and other previous analyses (e.g., Ofoegbu, 2001, 2000). In the 3DEC and UDEC-Voronoi models used for DOE analyses, zero-perturbation conditions were applied at a small distance above and below the emplacement drift. DOE needs to justify using such a small model with zero-perturbation conditions specified at the boundary when other analyses (including another DOE analysis) indicate a relatively large perturbed zone above and below an emplacement drift. At a technical exchange and management meeting on repository design and thermal-mechanical effects, the NRC staff expressed a concern regarding the DOE representation of boundary conditions in thermal-mechanical modeling (Reamer, 2001b, Key Technical Issue Agreements RDTME.03.13 and RDTME.03.19). Staff raised the same concern during an evaluation of the drift degradation analysis and model report (NRC, 2004).

The UDEC-Voronoi model also may be inappropriate for calculating seismically induced rockfall because the model does not extend far enough to encapsulate the potential drift degradation zone. An alternative approach to calculating the extent of drift degradation (referred to hereafter as the analytical model) indicates the degradation zone could extend 2–8 drift diameters above the initial perimeter of the emplacement drift, depending on the value of the bulking factor used in the calculation (cf., Gute, et al., 2003; Bechtel SAIC Company, LLC, 2003e, pp. 189–192). In contrast, the UDEC-Voronoi model extends only 2.68 drift diameters above the initial drift perimeter, which implies any degradation zone calculated using the model cannot extend more than a fraction of this distance. DOE stated the analytical model is overly conservative (Bechtel SAIC Company, LLC, 2003e, p. 199). This statement, however, cannot be confirmed using the analysis provided by DOE because the UDEC-Voronoi model used for the DOE analysis does not extend far enough to permit a comparison of its results against the analytical model. If the potential degradation zone indicated by the analytical model were encapsulated by the UDEC-Voronoi model, the degradation zone calculated using the latter could be used to assess the DOE statement the analytical model is overly conservative. Instead, the analytical model results indicate the UDEC-Voronoi model used for the DOE calculations probably truncated the potential degradation zone, which would raise a concern about using results from such a model to characterize the extent of drift degradation from seismic effects.

The temperature distributions used for the DOE analysis of thermally induced rockfall were obtained from a thermal-hydrological analysis (Bechtel SAIC Company, LLC, 2003e, p. 83). Only 10 percent of the repository thermal load was applied in the thermal-hydrological analysis during the first 50 years, because of an assumption (Bechtel SAIC Company, LLC, 2003e, p. 43) that 90 percent of the thermal load will be removed by ventilation during this period. The DOE information presented to justify using a heat-removal fraction of 90 percent (Bechtel SAIC Company, LLC, 2003f) has not been reviewed by the NRC staff. In a letter to DOE documenting the NRC staff review of earlier DOE information on heat removal through ventilation, Schlueter (2002) stated uncertainties in the ventilation models would disallow using a value of heat-removal fraction greater than 70 percent without further clarification of the uncertainties.

Representation of Yucca Mountain Seismic Hazard in Rockfall Modeling: As discussed in Sections 5.1.2.2.4.3 and 7.4.4.2 of this report, DOE is currently changing its approach to the development of seismic inputs in total system performance assessment consequence modeling of seismicity for postclosure performance assessment. DOE has not provided the revised seismic hazard information and, thus, at this time, the staff cannot evaluate adequacy of the seismic hazard information used in rockfall modeling.

Representation of Relevant Physical or Chemical Phenomena That May Affect Rockfall: Potential physical and chemical phenomena that may affect rockfall and drift degradation, but not represented explicitly in the DOE calculations include the static fatigue phenomenon (Bechtel SAIC Company, LLC, 2003g, pp. 6–19) and the geochemical alteration of fracture-wall rock (e.g., Ofoegbu, 2000, p. 2-8). The static fatigue phenomenon accounts for the difference between the strength of rock subjected to sustained loading and the strength measured through laboratory compression testing at conventional (i.e., relatively rapid) loading rates (e.g., Lajtai and Schmidke, 1986). DOE expects to obtain laboratory data characterizing the static-fatigue behavior of Yucca Mountain tuff[3] and to update the drift degradation analysis using these data. Although specific mechanisms of time-dependent strength degradation are not included in the DOE rockfall model, the effects of strength degradation are investigated. First, the shear strength of fracture surfaces in the 3DEC model is reduced by decreasing the friction angle from 41 to 30 degrees. DOE concludes such strength reduction would have a negligible effect on rockfall, but this conclusion needs to be reevaluated considering concerns regarding representation of the DOE fracture data in the 3DEC model (see the discussion in *Representation of Repository Host Rock in Rockfall Modeling*). Second, the effect of rock-mass strength degradation on rockfall was examined using the UDEC-Voronoi model by decreasing the cohesion parameter to zero in five steps of 20-percent reduction (Bechtel SAIC Company, LLC, 2003e, pp. 197–198). The analysis indicates a strength reduction by a factor of 40 percent or smaller would not cause significant rockfall. DOE, however, has not provided information regarding the effects of such strength reduction occurring in combination with thermal or seismic loading.

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[3]Price R.H. "Time Dependent and Thermal Properties." *Presentation at NRC and DOE Technical Information Exchange on Repository Design and Thermal-Mechanical Effects, May 6–8, 2003.* Las Vegas, Nevada. 2003.

Engineered Barrier System Process-Level Models: The process-level models used by DOE 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 used to perform these analyses because it can 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 demonstrating the finite element models used to simulate the functionality of the drip shield and waste package are sufficient to capture highly localized phenomena. To address this issue, DOE has proposed in the response to key technical issue agreement PRE.07.02 (Bechtel SAIC Company, LLC, 2003h) that the waste package finite element model mesh discretization is sufficient if the relative difference in results generated by the initial mesh and the refined mesh is approximately one order of magnitude smaller than the relative difference in mesh size in the region of interest. As defined by DOE, the mesh size refers to the volume or the area of the representative element in the region of interest. Reasonable approximations of the stress are needed to assess susceptibility of the various engineered barrier system components to a potential breach by plastic collapse or mechanical crack initiation and propagation.

Although the general methodology for establishing adequacy of a finite element mesh discretization is satisfactory (Bechtel SAIC Company, LLC, 2003h, Section 3.1.1), additional information is required to support the proposed convergence criterion. As defined in the report, the proposed convergence criterion requires the difference in results between two mesh discretizations be less than 10 percent of the relative difference in mesh size in the region of interest. The report did not explicitly state, however, which particular analysis results are to be used in implementing this convergence criterion. Because displacements are continuous from one element to the next (i.e., displacements exhibit C^0 continuity), basing the convergence criterion on this variable may not provide sufficient continuity in the discontinuous displacement derivatives, which are used to calculate the strains and stresses. Moreover, additional justification is required for relating the finite element solution results to mesh size. As noted by Bathe (1996, Section 4.3),

> "The element stresses are calculated using derivatives of the displacements ..., and the stresses obtained at an element edge (or face) when calculated in adjacent elements may differ substantially if a coarse finite element mesh is used. The stress differences at the element boundaries decrease as the finite element mesh is refined, and the rate at which this decrease occurs is of course determined by the order of the elements in the discretization."

In other words, the appropriate percentage of the difference in results relative to the difference in mesh size needed to achieve a consistent measure of the allowable discontinuity of stress and strain between adjacent elements is dependent on the element formulations used in the model (e.g., single-integration point versus higher-order elements). As a result, the proposed mesh discretization convergence criterion should be calibrated for the specific element formulations being used so the allowable discontinuity of the stresses calculated at the node points in the region of interest is reasonably consistent from one model to the next. For example, establishing an allowable difference between the minimum and maximum element results of interest at a shared node could be a basis for calibrating the proposed mesh discretization convergence criterion.

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, 2000g) 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. Although it has been determined that the effect of dynamic rock block impacts on the drip shield has a low significance with regard to overall repository performance, the following concerns are relevant to how the drip shield is generally represented in other process-level, finite element models.

Even though the drip shield is intended to be a free-standing structure, the DOE finite element model of discrete rock block impacts with the drip shield uses fixed displacement boundary conditions at the base of the drip shield. In addition, the finite element model does not account for the (i) potential interaction between the drip shield and gantry rails, (ii) effect of the invert floor moving vertically upward as a result of the seismic excitation that may occur concurrently with rockfall, or (iii) degradation of the carbon steel structural framework of the invert. These boundary conditions may have a significant influence on the overall structural behavior of the drip shield when subjected to rock block impacts (Gute, et al., 2003). 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 assumes in these models the contact area between the impacting rock block and drip shield will encompass at least a 3 m [9.9 ft] length of the drip shield. Distributing the impact load throughout 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 were consistent with localized, point-type impacts.

The constitutive relationships used for the drip shield materials (i.e., Titanium Grades 7 and 24) within the DOE finite element models simulating the drip shield and discrete 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 (ASME, 2001), however, are strongly dependent on temperature. For example, the room temperature values of yield stress and ultimate tensile strength for Titanium Grade 7 are, respectively, reduced from 275.8 MPa [40.0 ksi] and 344.7 MPa [50 ksi] to 176.5 MPa [25.6 ksi] and 249.6 MPa [36.2 ksi] at 150 °C [300 °F]. 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 ASME (2001). Note 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 by the U.S. Department of Defense (1998) also are 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 seismic excitation, rock block impacts, and sustained loading created by the accumulation of rockfall rubble. For example, the room temperature values of yield stress and ultimate tensile strength for Titanium Grade 5 are reduced from 827.4 MPa [120.0 ksi] and 896.3 MPa [130 ksi] to 653.6 MPa [94.8 ksi] and 752.9 MPa [109.2 ksi] at 150 °C [300 °F].

DOE has not provided an assessment of the stresses generated in the waste package outer container created by discrete rock block impacts, seismic excitation, and potential mechanical interactions with the drip shield. In addition, when assessing the response of the waste package to these mechanically disruptive scenarios, DOE should address the effects of welding defects and waste package material degradation processes such as uniform corrosion, localized corrosion, stress corrosion cracking, and the possible decrease in ductility caused by container fabrication processes, long-term thermal aging, or high strain rates experienced during seismic loading conditions. Breaching of the waste package outer container by either mechanical or corrosion processes will result in the exposure and subsequent degradation of the inner stainless steel container.

Mechanical interactions between the drip shield and waste package could occur if the accumulation of rockfall rubble is sufficient to cause the drip shield to either buckle or creep. This interaction could also occur if the invert (i.e., the drip shield and waste package foundation) should experience sufficient degradation. The primary concern is that the accumulated rockfall rubble load acting on the drip shield would be transferred to the waste package by way of the drip shield bulkheads. The stresses generated in the waste package outer container for this scenario have the potential to exceed the ultimate strength of Alloy 22 because of the relatively small contact area associated with this type of interaction. Because the Alloy 22 waste package outer container is expected to experience significant plastic deformations under these conditions, the initial contact area can be expected to increase significantly as the outer container deforms under the applied load. This increase in contact area will, in turn, reduce the average contact stress acting on the waste package outer container and, potentially, the deformed system could reach a state of equilibrium where the plastic flow of the material is arrested. DOE has redesigned the drip shield and the potential drip shield–waste package contact area may change for the new design.

Assuming the wasteform will not be subjected to rockfall loads, either directly or indirectly, the only potential mechanical loading condition that could affect the structural integrity of the wasteform is seismicity. At the present time, DOE has not established the threshold seismic loads that the various wasteforms to be disposed of at the potential repository can withstand without structural degradation.

Mechanical Failure Criteria: The following discussion addresses the four potential failure mechanisms that can affect engineered barrier system performance as the result of seismic rockfall and drift degradation disruptive events. These failure mechanisms are plastic collapse, fracture, stress corrosion cracking, and creep. It should be noted, however, plastic collapse and fracture are mutually exclusive because one bounds the other depending on the (i) material in question, (ii) state of the applied stress, and (iii) size and orientation of existing defects or cracks.

To assess the plastic collapse mechanical failure mode, DOE proposes to use the failure criteria suggested by ASME (2001, Appendix F–1341.2) for plastic analysis to interpret the results obtained from the finite element analyses (Bechtel SAIC Company, LLC, 2003i, Section 3.1.2). DOE pointed out, however, if these criteria are not satisfied, integral measures will be examined on a case-by-case basis to determine if a less conservative failure criterion may be used. In addition, it should be noted the proposed failure criteria (ASME, 2001, Appendix F–1341.2) for Level D service limits are intended to assure violation of the pressure-retaining boundary will not occur, but it is not intended to assure operability of components either during or following the specified event (ASME, 2001, Appendix F–1200). Because the proposed Level D service limits are based on the ASME code (2001) definitions of primary stresses, the assessment of the results obtained from finite element analyses of the engineered barrier system components subjected to various design basis events should be characterized in terms of these stresses.

To establish the applicable failure criterion (i.e., plastic collapse or fracture) for assessing engineered barrier system component material failure under mechanical loading conditions, DOE proposed to use a failure assessment diagram approach in their response to key technical issue agreement CLST.02.03 (Ziegler, 2002). This approach to establishing the applicability of the plastic collapse and fracture mechanics failure criteria is acceptable to the staff (Schlueter, 2003). As noted earlier, based on currently available information, the staff has determined that plastic collapse is the governing mechanical failure mechanism for the Alloy 22 waste package outer container and Titanium Grade 7 drip shield plate[4] (Dunn, et al., 2004). For the Titanium Grade 24 drip shield bulk heads and support beams, however, the staff has determined that mechanical failure may be governed by fracture mechanics or mixed mode (i.e., fracture and plastic collapse) behavior.[5] From a fracture mechanics point of view, failure is assessed in terms of the material fracture toughness, the applied stress, and the flaw size and geometry (Anderson, 1995). For example, the stress magnitude at the location of a flaw is dependent on the stress distribution within the drip shield in reaction to the applied load (e.g., seismic and accumulated rockfall rubble). The stress distribution, in turn, depends on the drip shield design, and any residual stresses created during the fabrication process. DOE has yet to establish the allowable applied stress for the Titanium Grade 24 drip shield components. In addition, DOE has not identified the drip shield fabrication methods and nondestructive evaluation testing procedures that will be employed to control flaw sizes, geometries, densities, and distribution within acceptable limits.

Basically, there are two methods for using the finite element method to assess the potential for the propagation of existing defects (i.e., fracture) or the stresses that would promote stress corrosion cracking. The first is to explicitly include the presence of a defect or crack in the finite element model. Using this approach allows the model to explicitly calculate the stresses generated at the crack tip when subjected to the design basis loads and, in some cases, calculate the stress intensity directly from the results of the analysis (i.e., from a fracture mechanics perspective). The second approach, which appears to be the DOE method of choice, is to model the drip shield and waste package without any defects and then assess the potential for stress corrosion cracking based solely on the calculated stresses. This may, or may not, be adequate, depending on the magnitudes and nature of the principal stresses (i.e., tensile or compressive) and the flaw size. Felbeck and Atkins (1996, p. 337) note,

> "A criterion for fracture based on the attainment of some characteristic maximum tensile stress has been used with a certain degree of success, particularly in very brittle solids. It should be clear, however, that quantitative criteria for fracture based merely on some maximum tensile stress (or combination of stresses) address only part of the necessary

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[4]Csontos, A.A., D.S. Dunn, Y. Pan, and G.A. Cragnolino. "The Effect of Fabrication Processes on the Governing Mechanical Failure of Alloy 22, Ti-Grade 2/7, and Ti-Grade 5/24 Alloys." Scientific Basis For Nuclear Waste Management XXVIII. L. Browning and J. Hanchar, eds. Warrendale, Pennsylvania: Materials Research Society. In press. 2004.

[5]Ibid.

and sufficient conditions for fracture. No consideration is given to the energetics of crack propagation. Although maximum stress theories may be intuitively appealing and *may* work reasonably well under the circumstances, they should not be expected to apply to large structures containing critical flaws."

The particular concern of the staff is the combined effect of a tensile first principal stress, which is perpendicularly oriented to the crack surface, and a compressive third principal stress oriented in the plane of the crack surface. If the crack geometry is not explicitly included in the model, the contribution of the additional tensile stress created at the crack tip by the in-plane compressive stress is not accounted for, resulting in an underestimation of the potential for crack propagation or stress corrosion cracking.

Based on the foregoing observations, DOE has not justified using a single component of stress to assess the potential for stress corrosion cracking occurring in the engineered barrier system component materials. The DOE response to the DOE and NRC key technical issue agreement RDTME.3.18 (Bechtel SAIC Company, LLC; 2003i, Appendix F) does not specifically identify what stress measure would be used as the equivalent uniaxial stress to assess the potential for stress corrosion cracking when a multiaxial stress state exists. Moreover, only one component of stress (calculated using the finite element method) appears to be used for assessing the potential of stress corrosion cracking of the drip shield and waste package in Bechtel SAIC Company, LLC (2003i, Appendixes B.3 and C.4). Specifically, with regard to the waste package, it is stated in Bechtel SAIC Company, LLC (2003i, Appendix B.3), "... the hoop stress, which promotes radially oriented crack growth, is the dominant component of stress in the waste package outer shell closure lid weld regions and, therefore, only the hoop stress profiles were considered in the integrated waste package degradation model ... ." It is not clear whether this conclusion is limited to the as-welded condition or also is true after stress mitigation by way of laser peening or plasticity burnishing. In addition, the statement "... the use of all stress components to determine principal stress for comparison against allowable stresses (from a uniaxial test) may be conservative, because it appears that initiation and failure is [sic] a function of the maximum stress only in a specific direction." (Bechtel SAIC Company, LLC; 2003i, Appendix F) must be justified. It must be recognized, however, that the potential for stress corrosion cracking is not based solely on the stress state of the material. The occurrence of stress corrosion cracking is also dependent on the water chemistry coming into contact with the material and the microstructure of the material (see Section 5.1.3.1 for additional discussions pertaining to the potential occurrence of stress corrosion cracking).

The staff recognizes that after stress mitigation, the waste package closure weld residual stresses are predominantly hydrostatic in nature (i.e., $\sigma_1 \approx \sigma_2 \approx \sigma_3$, where σ_i are the principal stresses) and compressive. As a result, the occurrence of stress corrosion cracking is unlikely in this region of the waste package. It still needs to be demonstrated, however, this stress state is not adversely affected by various mechanical loading scenarios after being placed into service (e.g., seismic loads, rockfall loads, or both). Furthermore, a more thorough and rigorous assessment of the stresses incurred by the drip shield when subjected to static rockfall rubble loads may be required to assess the potential occurrence of stress corrosion cracking.

In addition to plastic collapse, fracture, and stress corrosion cracking, creep is a fourth potential mechanical failure mechanism that may affect the engineered barrier system performance. Creep is the time-dependent strain observed during a constant stress test. The following sections discuss the potential repository conditions that may cause creep to occur of the drip shield and waste package outer container materials.

Because the reductions are significant in yield stress and ultimate tensile strength for Titanium Grades 7 and 24 resulting from elevated emplacement drift temperatures, staff are concerned these materials also will be susceptible to creep-related failures arising from the support of dead loads created by the accumulation of rockfall rubble. 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) reports 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 show 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).

In DOE and NRC key technical issue agreement TSPAI.02.02 (Reamer, 2001c), DOE agreed to provide the technical basis for screening out "Creeping of metallic materials in the EBS [Engineered Barrier System]" (TSPAI.02.02, Comment 37) from consideration in the total system performance assessment of the potential geologic repository for spent nuclear fuel and high-level waste. The information provided in Bechtel SAIC Company, LLC (2003i, Appendix C) addressing key technical issue agreement CLST.01.14 does not include an adequate technical basis for excluding creep as a potential failure mechanism for the titanium drip shield. Moreover, as reported by Neuberger, et al. (2002), information available in the literature indicates titanium and some of its alloys may creep at temperatures as low as 50 °C [122 °F] when subjected to stress levels as low as 60 percent of the material yield strength.

DOE has neither referenced specific creep data for Titanium Grades 7 and 24 nor provided adequate analyses demonstrating dead loads caused by accumulated rockfall rubble will not occur. Creeping of the drip shields subjected to rockfall rubble loads can reduce the clearance between the drip shield bulkhead and the waste package. Given time, the rockfall rubble loads ultimately may be supported by the waste package directly, or, during a seismic event, the clearance may have been sufficiently reduced to the point the drip shield will repeatedly impact the waste package, resulting in damage presently not accounted for.

If the waste package outer container is not breached directly from mechanical interactions with the drip shield under accumulated rockfall rubble load conditions, the potential for failure by creep may be a concern, given the magnitude of the applied stress and the recognition this stress will be present once manifested. At the present time, no definitive material data are available to support the inclusion or exclusion of creep as a potential failure mechanism for Alloy 22 under expected repository temperatures and the aforementioned loading conditions.

Finally, DOE has not developed a methodology for assessing the accumulation of damage that could occur to the drip shield and waste package by rockfall and drift degradation, along with multiple seismic events over time.

Consistency of Engineered Barrier System Design Bases and Design Criteria: To date, DOE has not provided the relationship between the engineered barrier system component (e.g., the drip shield, waste package, drift invert system, and so on) design criteria and the design basis loads for seismicity and rockfall and drift degradation. As a result, the magnitude of the seismic and rockfall and drift degradation loads, including appropriate combinations of these loads, that could compromise the structural integrity of the engineered barrier system components is not clear. In the DOE response to Structural Deformation and Seismicity Key Technical Issue agreements SDS.01.01 and SDS.02.03 (Stepp and Cornell, 2001, p. 1), however, DOE committed to performing a probabilistic risk assessment for the engineered barrier system "... implementing standard seismic risk assessment methods and guidelines established by the NRC .... The EBS [engineered barrier system] components that will be analyzed will include the waste package, waste emplacement pallet, and drip shield and will include the affects of rockfall."

While some information on the mechanical disruption of the engineered barriers (i.e., rockfall and drift degradation) with respect to data being sufficient for model justification may be available at the time of a potential license application, the staff is currently of the view that DOE should provide additional information on the DOE determination of the thermo-mechanical response of the intensely fractured zone of the nonlithophysal unit, on the combined effects of static fatigue and seismic loading in rockfall, on the interactions of the drip shield and outer waste package due to rockfall, on effects from the drip shield fabrication process, on creep failure data for the specific titanium alloys to be used in the drip shield, and on the potential accumulation of damage to the engineered barrier system from rockfall, drift degradation, and multiple seismic events.

5.1.3.2.4.4.3 Data Uncertainty

Risk insights pertaining to the mechanical disruption of engineered barriers by rockfall and drift degradation indicate the most important data uncertainties pertain to the variability and corresponding correlation of those parameters that affect the temporal and spatial occurrences of rockfall and drift degradation.

Data Uncertainty in the Analysis of Thermally Induced Rockfall: The occurrence of rockfall and drift degradation owing to repository thermal loading will be controlled by the relationship between thermally induced rock stress and the available rock strength. Thermal stress development in a heated rock mass is controlled by the induced temperature distribution, the rock stiffness (typically described in terms of the Young's modulus, E), and thermal expansivity, α. The induced temperature at Yucca Mountain will be determined by the thermal loading (including any heat removal through ventilation) and the thermal and moisture-flow properties of the rock. The DOE treatment of data uncertainties in the calculation of temperature distributions is reviewed in Section 5.1.3.3, Quantity and Chemistry of Water Contacting Engineered Barriers and Waste Forms. The potential for the induced thermal stress to cause rockfall and drift degradation is controlled by the rock strength. The strength of a rock mass is typically expressed in terms of the Mohr-Coulomb failure criterion (e.g., Jaeger and Cook, 1979, pp. 95–101), defined in terms of the friction angle, ϕ, and a second parameter such as the unconfined compressive strength, q_u. DOE provided values of α, ϕ, q_u, and E for calculating thermally induced rockfall and drift degradation in the lithophysal rock units at Yucca Mountain (Bechtel SAIC Company, LLC, 2003e, Attachments V.4.1 and V.5).

The DOE data provide α as a function of temperature but do not include variation of the parameter spatially or variation with respect to another rock property. Information presented by DOE during a technical exchange[6] indicates potential variability of α with respect to factors other than temperature. The α data in the drift degradation report (Bechtel SAIC Company, LLC, 2003e, p. V–26), however, do not include information sufficient to permit a reliable estimate of the uncertainty band. Furthermore, data in the drift degradation report provide values of α for temperatures up to 125 °C [257 °F], whereas the Price[7] data extend to a temperature of approximately 180 °C [356 °F] and indicates an increase in the gradient of the α-versus-temperature curve at approximately 150 °C [300 °F]. Because the magnitude of thermal stress is generally proportional to αE, uncertainties in either α or E can affect the calculated stresses.

DOE has not obtained any measurements of ϕ for the lithophysal rocks because of practical limitations on laboratory testing caused by the occurrence of relatively large void spaces (lithophysae) in the rock. DOE, therefore, relies on experience-based estimates indicating values of ϕ in the range 40–50 degrees (Bechtel SAIC Company, LLC, 2003e, p. 280 and V–14). Rockfall, however, depends on the strength of rock near the boundary of the drift openings. The effect of ϕ on rock strength is negligible near the boundary of the openings because the value of confining stress at such locations is approximately zero. Therefore, if rockfall assessment is based on analyzing the stress conditions near the boundary of the openings (e.g., Ofoegbu, et al., 2004), results of the assessment would not be sensitive to ϕ. If, on the other hand, the assessment includes calculating rock strength at locations far enough from the openings such that the influence of confining pressure on strength is not negligible, then the result of the assessment will likely be sensitive to uncertainties in the value of ϕ. The UDEC-Voronoi model used for the DOE calculations is likely sensitive to variations in ϕ because it includes calculations of rock-failure initiation at locations far from the openings (e.g., Bechtel SAIC Company, LLC, 2003e; Figures on pp. 173, 187, and 188). DOE, however, does not provide an assessment of the sensitivity of the rockfall calculations to ϕ variations (cf., Bechtel SAIC Company, LLC, 2003e, Tables on p. 168).

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[6]Price R.H. "Time Dependent and Thermal Properties." Presentation at NRC and DOE Technical Information Exchange on Repository Design and Thermal-Mechanical Effects, May 6–8. Las Vegas, Nevada. 2003.

[7]Ibid.

The DOE data for the strength and stiffness of the lithophysal rocks indicate the value of E varies from approximately 5 to 20 GPa [725 to 2,900 ksi], and q_u increases approximately linearly with E within the range of the measured data (Bechtel SAIC Company, LLC, 2003e, Figure V–3). Data are obtained from laboratory unconfined compression testing of 29 specimens, each approximately 30 cm [12 in] in diameter with a length-to-diameter ratio of 1.1–2.0 (Bechtel SAIC Company, LLC, 2003e, Table V–8). The specimens are obtained from upper lithophysal (Tptpul) and lower lithophysal (Tptpll) rock units along the Exploratory Studies Facility tunnel, the Enhanced Characterization of Repository Block cross drift, and at Busted Butte (Bechtel SAIC Company, LLC, 2003e, p. 223). Measurements of E and q_u are obtained from 19 of the 29 tests. Only q_u is measured from the remaining 10 tests. Only 14 of the 19 q_u-E data pairs are usable because the other five pairs are from specimens with a length-to-diameter ratio of 1.5 or smaller. Ten of the usable 14 data pairs are obtained from Tptpul specimens, and 4 pairs are obtained from Tptpll specimens. The DOE information[8] indicates approximately 80 percent of the emplacement drifts would be located in Tptpll rock and approximately 4 percent in Tptpul.

The 14 q_u-E data pairs are treated as representing the same data population in the DOE analysis (Bechtel SAIC Company, LLC, 2003e, Figure V–3), which implies treating the Tptpul and Tptpll rocks as belonging to a generally homogeneous continuum with statistically varying mechanical properties. Significant variations in physical characteristics among the two rock types (Bechtel SAIC Company, LLC, 2003e, p. 220) indicate significant variability in the mechanical properties of such a generally homogeneous continuum. The Tptpll is intensely fractured with short length, interlithophysal, and, predominantly, vertical fractures having a spacing on the order of inches. The Tptpul, in contrast, is typically unfractured. The lithophysal openings in the Tptpul have a diameter in the range approximately 1–10 cm [0.39–3.9 in]. Lithophysal openings in the Tptpll, in contrast, vary from smooth and spherical to irregular with sharp boundaries and have a size in the range approximately 1–180 cm [0.39–71 in]. The large variation in the occurrence and characteristics of physical features that affect strength and stiffness implies a similar variation in the strength and stiffness of a generally homogeneous continuum representing the lithophysal rocks. The variability implies a large uncertainty in using the 14 q_u-E data pairs to represent the strength and stiffness of the lithophysal rocks.

To represent the data uncertainty, DOE divides the range of the 14 q_u-E data pairs into 5 intervals referred to as rock-mass categories (Bechtel SAIC Company, LLC, 2003e, pp. 227–228 and V–13 through V–14). Each rock-mass category is assigned the values of q_u and E corresponding to the midpoint of the data interval represented by the category. In other words, DOE determines the values of q_u and E using the best-fit line to the 14 q_u-E data pairs (Bechtel SAIC Company, LLC, 2003e, Figures 149 and V–3). Even if the DOE approach were to represent variability within the 14 data points, there would be considerable concern regarding using the approach to represent uncertainty in the strength and stiffness of the lithophysal rock mass. The uncertainty arises from several factors, such as sparsity of data; low spatial coverage of sampling; wide variability of fracture intensity and size, shape, and volume fraction of lithophysal cavities; and potential variability of rock-matrix properties. DOE does not account for uncertainty in the interpretation of the measured data to obtain values of q_u and E for modeling rockfall and drift degradation.

[8]Board, M.P. "Mechanical Drift Degradation Analysis." *Presentation to Advisory Committee on Nuclear Waste, November 19, 2003.* Las Vegas, Nevada. 2003.

At a technical exchange and management meeting about repository design and thermal-mechanical effects, DOE agreed to address the NRC concerns regarding the DOE representation of data uncertainty in thermal-mechanical modeling (Reamer, 2001b). The staff raised the same concern during an evaluation of the drift degradation analysis and model report (NRC, 2004).

While some information on the mechanical disruption of the engineered barriers (i.e., rockfall and drift degradation) with respect to data uncertainty may be available at the time of a potential license application, the staff is currently of the view that DOE should provide additional information on the DOE determination of the sensitivity of the models to possible variation in the friction angle, and the coupled uncertainties in thermal expansivity, unconfined compressive strength, and Young's modulus for the rock units of the potential repository.

5.1.3.2.4.4.4 Model Uncertainty

Risk insights pertaining to the mechanical disruption of engineered barriers by rockfall and drift degradation indicate the most important model uncertainties pertain to the interpretations of alternative rockfall and drift degradation process-level models.

Alternative Conceptual Models for Rockfall and Drift Degradation: DOE considered three potential alternatives to the approaches it uses for modeling the mechanical behavior of a fractured rock mass and concluded that none of the three alternatives is applicable to rockfall modeling because none would result in a direct calculation of the amount of rockfall (Bechtel SAIC Company, LLC, 2003e, p. 209). One alternative approach considered by DOE consists of representing the lithophysal rock as an elastic-plastic continuum. The potential failure and postfailure behaviors of the continuum would be described using a yield criterion [e.g., Mohr-Coulomb failure criterion (cf., Jaeger and Cook, 1979, pp. 95–101)] and a flow rule.

An alternative to the elastic-plastic continuum modeling approach consists of calculating stresses using a linear-elastic model of the rock and comparing the stresses against the rock strength to determine potential occurrence of over-stress conditions. This approach is widely used to assess the stability of underground openings. Although the approach would not result in a direct calculation of rockfall magnitudes, it can be used in combination with an analytical model (e.g., Gute, et al., 2003; Bechtel SAIC Company, LLC, 2003e, pp. 189–192) to quantify rockfall and drift degradation. The linear-elastic model can be used to identify potential persistent occurrence of over stress (e.g., Ofoegbu, et al., 2004). Persistent occurrence of over stress near the boundary of an emplacement drift in such a model implies potential occurrence of progressive rockfall. The progressive rockfall would begin with the breakup of any over-stressed rock near the roof of the opening. Such rock would breakup and fall; expose new surfaces that would, in turn, be subjected to over stress, breakup, and fall; thereby causing progressive rockfall and drift degradation. An application of this analysis approach to the DOE design for emplacement drifts in lithophysal rocks at Yucca Mountain (Ofoegbu, et al., 2004) indicates stress conditions that would favor progressive rockfall and drift degradation will exist around the emplacement-drift openings for more than 1,000 years. Because the over-stress condition will persist for a long time, rockfall and drift degradation will begin after the ground-support system becomes ineffective, following repository closure. Progressive rockfall and drift degradation will cease after the openings develop into a stable shape or become filled with rubble. The amount of rockfall rubble and the extent of drift degradation resulting from the process can be calculated using an analytical model based on the bulking behavior of broken rock (e.g., Gute, et al., 2003).

A fundamental difference between the linear-elastic model and the UDEC-Voronoi model may result in the two models producing differing estimates of the occurrences and magnitudes of rockfall. Because a linear-elastic model does not include mechanism for stress relief, any suppressed thermal expansion of the heated rock mass would cause a stress change as specified by the stress-strain relationship. The Voronoi model, on the other hand, includes several interparticle contacts that may slide or separate to relieve stress. As a result, the thermally induced stress change calculated in the Voronoi model may differ from the stress change specified by the stress-strain relationship for the equivalent linear-elastic solid. A rock mass naturally would have several cracks of various orientations and sizes that may serve as stress-relief mechanisms, thereby limiting thermally induced stress change to smaller than calculated, using the linear-elastic model. Although the interparticle contacts in a Voronoi model may be similar to cracks in that they provide stress-relief mechanisms; the orientations, sizes, and mechanical characteristics of the interparticle contacts in a Voronoi model are not selected based on any characterization of fracture features in the modeled rock mass. The Voronoi model is calibrated to represent the overall mechanical behavior, not the internal structure, of the rock mass. The occurrence of stress-relief mechanisms in the Voronoi model, therefore, is not calibrated to represent the potential occurrence of such mechanisms in the modeled rock mass. Therefore, whereas the linear-elastic model may overestimate thermally induced stress by not including any stress-relief mechanisms, the Voronoi model may underestimate the stress by including an inappropriate amount of stress relief. Both models are calibrated to match the overall stress-strain behavior of the rock, but tend to represent opposite extremes of the aspects of rock behavior controlled by the occurrence of stress-relief mechanisms. The two models, therefore, are admissible as alternative conceptual models of the mechanical behavior of a rock mass subjected to nonuniform heating.

The rockfall assessment obtained by combining a linear-elastic analysis (e.g., Ofoegbu, et al., 2004) with the analytical drift-degradation model (e.g., Gute, et al., 2003) represents an alternative to the DOE approach (Bechtel SAIC Company, LLC, 2003e) is consistent with the DOE data for lithophysal rocks and with the current scientific understanding of the mechanical behavior of a rock mass subjected to nonuniform heating. DOE has not evaluated the effects of such rockfall assessment on the performance of a potential geologic repository at Yucca Mountain as is required according to 10 CFR 63.114(c).

At the time this report was prepared, the effects of rockfall and drift degradation were not yet implemented within the total system performance assessment model abstraction for disruptive events. DOE agreed to address the NRC staff concerns, as discussed in Sections 5.1.3.2.4.4.1, 5.1.3.2.4.4.2, and 5.1.3.2.4.4.4. Depending on resolution of these concerns, the effects of rockfall and drift degradation will be included or excluded from the total system performance assessment model abstraction for disruptive events.

While some information on the mechanical disruption of the engineered barriers (i.e., rockfall and drift degradation) with respect to data uncertainty may be available at the time of a potential license application, the staff is currently of the view that DOE should provide additional information on the DOE determination of how the potential effects of rockfall and drift degradation will be implemented in the total system performance assessment model abstraction for disruptive events.

5.1.3.2.4.4.5 Model Support

Risk insights pertaining to the mechanical disruption of engineered barriers by rockfall and drift degradation as they pertain to model support cannot be established at this time because the total system performance assessment model abstractions have yet to be implemented and provided for review.

At the time this report was prepared, the effects of rockfall and drift degradation were not yet implemented within the total system performance assessment model abstraction for disruptive events. DOE agreed to address the NRC concerns, as discussed in Sections 5.1.3.2.4.4.1, 5.1.3.2.4.4.2, and 5.1.3.2.4.4.4. Depending on the resolution of these concerns, the effects of rockfall and drift degradation will be included or excluded from the total system performance assessment model abstraction for disruptive events.

Overall, the current information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.2.4.6), is sufficient to expect that the information necessary to assess mechanical disruption of engineered barriers (i.e., rockfall and drift degradation) with respect to model abstraction output being supported by objective comparisons will be available at the time of a potential license application.

5.1.3.2.4.5 Criticality

DOE has not included nuclear criticality as part of the mechanical disruption of engineered barriers model abstraction. DOE indicated it intends to exclude nuclear criticality events from the performance assessment, based on low probability. The DOE evaluation of nuclear criticality is assessed in Section 5.1.2.2.4.4 of this report.

5.1.3.2.4.6 Summary and Status of Key Technical Issue Subissues and Agreements

Table 5.1.3.2-1 provides the status of all key technical issue subissues, referenced in Section 5.1.3.2, 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 review methods discussed in Section 5.1.3.2.4. Note the status and the detailed agreements pertaining to all the key technical issue subissues are provided in Table 1.2-1 and Appendix A.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (e.g., specified testing or analyses), indicates that information necessary to begin a technical review will likely be available at the time of a potential license application. As noted in this section of the report, however, further information should be provided on the amount of arching that may occur in a lithophysal rubble pile, and on the potential effects of an accumulated rubble load on the drip shield (Section 5.1.3.1.4.4.1); on the thermo-mechanical response of the intensely fractured zone of the nonlithophysal unit, on the combined effects of static fatigue and seismic loading in rockfall, on the interactions of the drip shield and outer waste package because of rockfall, on effects from the drip shield fabrication process, on creep failure data for the specific titanium alloys to be used in the drip shield, and on the potential accumulation of damage to the engineered barrier system from rockfall, drift degradation, and multiple seismic events (Section 5.1.3.2.4.4.2); on the sensitivity of the models to possible variation in the friction angel, and on the overall effect of data uncertainty in the thermo-mechanical modeling (Section 5.1.3.2.4.4.3); and on how the potential effects of rockfall and drift degradation will be implemented in the total system performance assessment model abstraction for disruptive events (Section 5.1.3.2.4.4.4).

Table 5.1.3.2-1. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreement*
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 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	IA.1.01 IA.1.02
	Subissue 2—Consequences of Igneous Activity	Closed	IA.2.10
		Closed-Pending	IA.2.18 IA.2.19 IA.2.20
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 Geologic 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.3.03 RDTME.3.15 to RDTME.3.19
Structural Deformation and Seismicity	Subissue 1—Faulting	Closed-Pending	SDS.1.02

Table 5.1.3.2-2. Related Key Technical Issue Subissues and Agreements (continued)			
Key Technical Issue	**Subissue**	**Status**	**Related Agreement***
Structural Deformation and Seismicity	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
	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 review methods

Note: Key Technical Issue Agreement GEN.1.01 pertains to multiple integrated subissues, as well as some specific issues related to this integrated subissue.

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5.1.3.3 Quantity and Chemistry of Water Contacting Engineered Barriers and Waste Forms

5.1.3.3.1 Description of Issue

The Quantity and Chemistry of Water Contacting Engineered Barriers and Waste Forms Integrated Subissue addresses features, events, and processes in the engineered barrier system that may alter the chemical composition and volume of water contacting 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 (Section 5.1.3.6), and quantity and chemistry of water inside breached waste packages are addressed by the Radionuclide Release Rates and Solubility Limits Integrated Subissue (Section 5.1.3.4). The relationship of this integrated subissue to other subissues is depicted in Figure 5.1.3.3-1. The figure shows the relationship between this integrated subissue and the Flow Paths in the Unsaturated Zone (Section 5.1.3.6), Radionuclide Transport in the Unsaturated Zone (Section 5.1.3.7), Degradation of Engineered Barriers (Section 5.1.3.1), Radionuclide Release Rates and Solubility Limits (Section 5.1.3.4), and Mechanical Disruption of Engineered Barriers (Section 5.1.3.2) subissues. The overall organization and identification of all the integrated subissues are depicted in Figure 1.1-2.

DOE documented its approach to modeling the quantity and chemistry of water contacting engineered barriers and wasteforms in various reports prepared in support of the 2002 site recommendation and in anticipation of a license application (CRWMS M&O, 2000a–f, 2001a; Bechtel SAIC Company, LLC, 2003a,b). Key elements of the abstraction have changed since the previous version of this status report (NRC, 2002). At the time of preparation of this report, relevant reports supporting a potential license application had not yet been released.

This section documents the current NRC staff understanding of the model abstractions developed by DOE to incorporate the quantity and chemistry of water contacting engineered barriers and wasteforms into its total system performance assessment. The assessment focuses on those aspects most important to waste isolation based on the risk insights gained to date (Appendix D). The scope of the assessment presented examines whether data gathered and methodologies developed by DOE are likely to be adequately documented for the staff to undertake a detailed technical review if a license application were submitted. This assessment is not a regulatory compliance determination review of a license application.

5.1.3.3.2 Relationship to Key Technical Issue Subissues

The Quantity and Chemistry of Water Contacting Engineered Barriers and Waste Forms Integrated Subissue incorporates subject matter previously captured in the following 20 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)

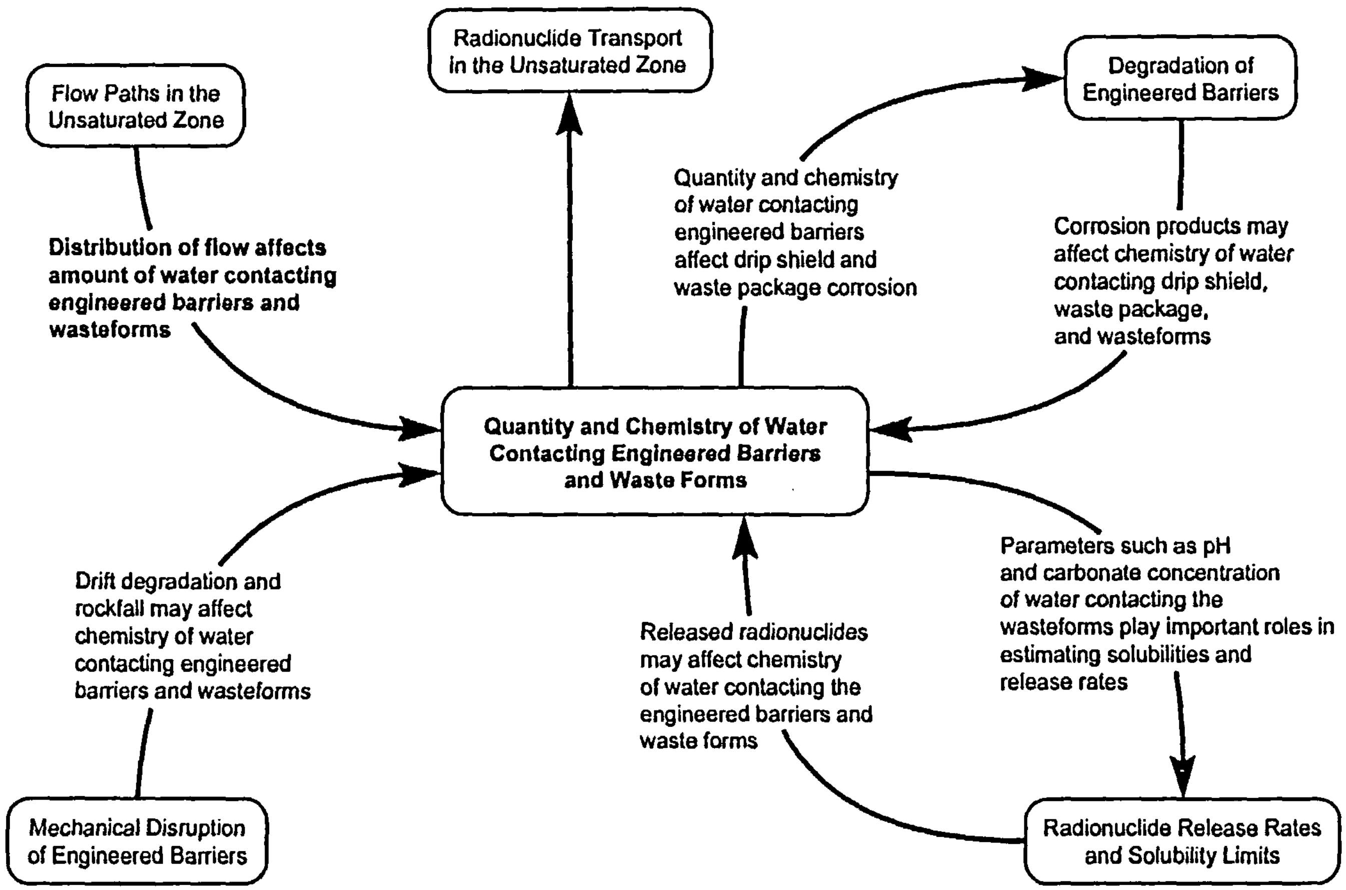

Figure 5.1.3.3-1. Diagram Illustrating the Relationship Between the Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms Integrated Subissues and Other Integrated Subissues. Material in Bold Is Identified in the Text.

- 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 Coupled 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)

- 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 Effects 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)

- Radionuclide Transport: Subissue 4—Nuclear Criticality in the Far Field (NRC, 2000b)

- 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 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.

5.1.3.3.3 Importance to Postclosure Performance

Five of the DOE eight principal factors in the repository safety strategy can be related to the Quantity and Chemistry of Water Contacting Engineered Barriers and Waste Forms Integrated Subissue (CRWMS M&O, 2000g). These five 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.

The details of the risk insights ranking are provided in Appendix D. This risk insight has been rated as having high significance to waste isolation, because evaluating the range in chemistry of water seeping into the drift and contacting the drip shield and waste package is important for determining corrosion modes and rates of the engineered materials. These risk insights are based on NRC analyses using the TPA Version 4.1 code, that provides information on the timing and extent of seepage, which affects the timing and quantity of water contacting engineered barriers and wasteforms (Mohanty, et al., 2002, Chapter 4). The same report also documents analyses that show the importance to waste isolation of the quantity and chemistry of water contacting engineered barriers and wasteforms. The risk insights report (Appendix D) assesses the quantity and chemistry of water contacting engineered barriers and wasteforms as having high significance to waste isolation. In particular, the chemistry of the seepage water is the most important issue. The chemistry of the water in the emplacement drifts depends on three basic water sources: seepage, condensation, and deliquescence. The expected lifetimes of the drip shield and waste package depend strongly on the quantity and chemistry of water contacting their surfaces. Low pH and elevated concentrations of certain dissolved anionic species would enhance aqueous corrosion—promoting specific corrosion modes, either uniform or localized, the latter of which has orders-of-magnitude higher rates than the former. These corrosion-promoting conditions can arise either from evaporation of seepage waters on contact with the heated surfaces of the engineered barriers or by brine formation caused by deliquescence of salts on the surfaces.

The quantity and chemistry of water that drips onto the drip shield and waste package surfaces largely determine the composition of the salts that can precipitate on those surfaces. The composition of a salt determines the composition of its associated brine and the relative humidity conditions under which that salt, once formed, will deliquesce to form a brine. The composition of the salt is largely determined by the chemistry of the evaporating water, whereas the timing and extent of salt formation is largely determined by thermal-hydrological conditions in and above the repository. In addition to increasing the concentration of deleterious anions promoting corrosion, evaporation of seepage water and deliquescence can also concentrate corrosion inhibiting species such a nitrate.

Analyses by Browning, et al. (2004) suggest the in-drift environments of most concern to drift shield and waste package corrosion for the period of performance are those arising when the dominant aqueous processes are (i) seepage and evaporation and (ii) seepage, evaporation, and condensation. Salt deliquescence is a factor in both environments. Significant condensation may, however, dilute solutions and reduce the likelihood of enhanced corrosion rates. The analysis concluded that the duration, conditions, and corrosion-related consequences of these two environments are of particular significance.

Uncertainty in abstracting the quantity and chemistry of water contacting engineered barriers and wasteforms arises from several sources. Bounding the range of in-drift water compositions requires characterization of diverse in-drift features, events, and processes that may alter the chemical composition of seepage waters. The timing, quantity, and temperature of seepage water entering the drift depends on diverse factors such as climate, flow pathways in the unsaturated zone, heat load, drift design, preclosure ventilation, and local controls on flow pathways. The most significant source of uncertainty in determining the chemical environment for corrosion of the engineered barriers is the range of in-drift water compositions that may result from spatial and temporal variations in seepage water composition, the composition and amount of condensed water formed by cold-trap processes, and the extent of chemical interactions between these waters and engineered and natural materials. Coupled thermal-hydrological-chemical processes occurring in the rocks that overlie the potential repository largely will determine the quantity and chemistry of water seeping into the drifts. Prediction of these processes with reactive transport simulations is limited by model and parameter uncertainties. Some uncertainty also remains with respect to the composition of dust that may be deposited on the drip shield and waste package surfaces during the ventilation period and the extent to which chemical interactions between dust and in-drift waters may form aqueous layers containing aggressive anionic species.

5.1.3.3.4 Technical Basis

NRC developed a review plan (NRC, 2003) consistent with the acceptance criteria and review methods found in the previous issue resolution status reports. An assessment of the DOE approaches for including the quantity and chemistry of water contacting engineered barriers and wasteforms in total system performance assessment abstractions is provided in the following subsections. This assessment is organized according to the five review methods identified in Section 2.3: (i) Model Integration (including system description), (ii) Data and Model Justification, (iii) Data Uncertainty, (iv) Model Uncertainty, and (v) Model Support.

5.1.3.3.4.1 Model Integration

The DOE performance assessment abstraction of the quantity and chemistry of water contacting engineered barriers and wasteforms is described in the technical basis documents on in-drift chemical environment and seepage (Bechtel SAIC Company, LLC, 2003a,c). DOE significantly updated the abstraction since the last NRC assessment (2002). The quantities and rates of water supplied to the engineered barriers are modeled on the basis of repository-specific, thermal-hydrological conditions. The unsaturated zone flow abstraction (CRWMS M&O, 2000h; Bechtel SAIC Company, LLC, 2003c) 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 boiling period, DOE evaluates the possibility that coupled effects on flow would alter significantly flow pathways (CRWMS M&O, 2001a,b; Bechtel SAIC Company, LLC, 2003a,c). These reports conclude mineral precipitation would alter fracture permeability in a thin region and would have the beneficial effect of reducing seepage. Hence, seepage fluxes under ambient and thermally perturbed conditions are taken directly from thermal-hydrological models without considering chemical processes and water chemistry.

Based on the thermal seepage model, DOE considers when the drift temperature is above a threshold value of 100 °C [212 °F], no ground water will reach the waste package or drip shield by seepage (Bechtel SAIC Company, LLC, 2003a). Above this threshold temperature, the chemical environment on the surfaces of the waste package and drip shield will be dominated by the chemistry of the dusts and any concentrated aqueous solution that can form by deliquescence. Below this temperature, ground water may reach the waste package and drip shield by seepage; thus, the chemistry of the ground water will dominate the chemical environment on the surfaces. Therefore, aqueous corrosion of the waste package and drip shield is considered by DOE in two scenarios: the dust deliquescence scenario and the crown seepage scenario (Bechtel SAIC Company, LLC, 2003b; Section 5.1.3.1.4 of this report).

The basic approach to establishing the range of credible water chemistries is to select a set of starting dilute waters and model the evolution of the waters as they are subjected to repository-relevant conditions and processes (Bechtel SAIC Company, LLC, 2003a, especially Appendixes A, E, and G). The water chemistries selected as representative of seepage waters entering the drift through time are based on 5 realizations of a detailed process-level model, the results of which are abstracted into 11 groups (bins) of like water types on the basis of the chemical divide phenomenon. Lookup tables of solution compositions are constructed by simulating the evaporation of the 11 seepage bin chemistries at different temperatures and PCO_2 values, and time-integrated probabilities of occurrence are calculated for each bin or water type. For the dust deliquescence scenario, DOE models evaporation of experimental dust leachates to produce six bins, or water types, and constructs lookup tables. In the DOE total system performance assessment, key chemical and physical parameters are read from the lookup tables. If aqueous conditions are found to exist, chemical composition indicators "interpolated from the lookup tables are then applied as input to other models that represent different modes of corrosion, radionuclide solubility, colloid mobility, and in-package chemistry" (Bechtel SAIC Company, LLC, 2003a, Page 4-34).

Staff find reasonable the binning approach to predicting the chemical composition of water contacting the drip shield and waste packages, including both evaporation of seepage and deliquescence. The approach appears to take into account important chemical processes and components, including introduced and engineered materials, and the DOE container corrosion tests include a sufficient range of environmental conditions to adequately reflect potential in-drift environments (Section 5.1.3.1). The available information is sufficient to conclude that the information necessary to assess integration of the abstraction of the quantity and chemistry of water contacting engineered barriers and wasteforms with other components of the DOE performance assessment will be available at the time of a potential license application.

The DOE Multiscale Thermohydrological Model (CRWMS M&O, 2001b) output includes in-drift temperature and relative humidity, which are fed to other process models (e.g., corrosion and near-field chemistry), and percolation at 5 m [16 ft] above the drift, which is fed to the seepage model in the total system performance assessment model. Temperature and relative humidity across the drift and percolation above the drift are important to performance because they influence the five principal factors in the repository safety strategy discussed in Section 5.1.3.3.3. The following few paragraphs discuss the integration of the thermohydrological model with the seepage model, the in-drift mass and heat transfer conceptual model, and the thermal-mechanical model.

There should be consistency of results from the DOE Multiscale Thermohydrological Model with the ambient seepage model (Bechtel SAIC Company, LLC, 2003d; CRWMS M&O, 2002h) during periods when there is no significant thermal perturbation of the system. There are marked differences, however, in seepage rates between the two models when no heat load is applied to the thermohydrological model (Bechtel SAIC Company, LLC, 2003c) suggesting the seepage representations differ markedly. Features, such as fracture heterogeneity, that have been discounted as important because of the metric of seepage rate, may have been incorrectly discounted.

The DOE Multiscale Thermohydrologic Model approach does not represent natural convection and the cold-trap effects (i.e., heat and mass movement along the length of drift), nor has DOE provided a basis for excluding natural convection and the cold-trap process from performance assessments. Redistribution of heat and moisture across the engineered barrier system and along the drift between the center and edge of the repository affect the quantity and chemistry of water contacting the engineered barrier components and the transport of potentially released radionuclides to the natural environment. An evaluation of three-dimensional effects is important for understanding axial and local natural convection airflow patterns caused by thermal gradients that may alter thermal profiles along the drift and cause vapor redistribution leading to condensation on cooler surfaces. The hottest part of the drift wall will control the vapor pressure in the drift. Condensation theoretically will occur on surfaces that are cooler than the source area of the water for the vapor, which combined with the non-uniform distribution of drift wall temperatures caused by natural convection and radiation, leads to the possibility of condensation on components of the engineered barrier besides the drift wall. Also, local temperature variations in the engineered barrier system also may lead to local elevation of relative humidity or condensation (e.g., beneath the drip shield). At the Thermal Effects on Flow Technical Exchange and Management meeting (Reamer, 2001a), DOE agreed to consider the cold-trap effect and incorporate important effects in the thermohydrological model for performance assessment. DOE provided a summary of the current approach in the Multiscale Thermohydrological Model approach in Appendix L of Bechtel SAIC Company, LLC (2003a) that neglects the three-dimensional effects of convection on heat and mass transfer in the drift. The DOE noted, however, that process-level modeling of three-dimensional heat and mass transfer in drifts was expected to be completed prior to license application.

There is sparse data to support simulations of heat and mass transfer in drifts, though DOE maintains the Passive Test will provide support for model validation. Prior to the elimination of the heat source at the western end of the drift, natural convection and the cold-trap process may have occurred in the Passive Test in the Enhanced Characterization of the Repository Block drift when it was isolated from the ventilation system by several bulkheads. The environment in the Passive Test has not yet returned to ambient conditions, however, large amounts of water appeared to have been redistributed along the drift by temperature gradients. Convective air flow at higher temperatures and larger temperature gradients than shown in the Passive Test will occur in the postclosure heated drifts. Hence, there will be some unquantified uncertainty associated with simulating heat transfer processes in the markedly different temperature regimes because of the transition from laminar to turbulent air flow between the regimes. CNWRA simulations based on independent laboratory experiments and related analyses (Fedors, et al., 2004) showed axial flow patterns would not be impeded by the strong cross-sectional flow patterns imparted by the heat rising directly off the waste package, implying axial convection and the cold-trap process will not be limited to the extreme ends of each drift. Modeling by Danko and Bahrami (2004) estimates significant portions of drifts would experience condensation at rates larger than estimated by seepage.

Thermal-mechanical processes may significantly affect hydrological properties (Section 5.1.3.2). DOE initially evaluated the thermal-mechanical effects on hydrological properties based on analyses of localized thermally induced rock responses near a heated drift (CRWMS M&O, 2000i; Bodvarsson, et al., 2001). The case of fracture-aperture changes in the pillars between two heated drifts was not considered in the DOE analyses. NRC (2002) suggested an increase is possible in aperture of subhorizontal fractures in the pillars between drifts because of thermal-mechanical effects and could be important to cross-repository flow caused by the potential diversion of water flux from the pillar to one of the adjacent drifts. DOE responded to this important concern by developing a fully coupled thermal-hydrological-mechanical methodology, based on the thermohydrological simulator TOUGH2 Version 1.6 and the fracture mechanics simulator FLAC3D Version 2.0 (Bechtel SAIC Company, LLC, 2003e). To assess the impact of thermal-hydrological-mechanical processes on the flow field, DOE calculated changes in the mean value of the permeability for a conservative case of strong thermally, hydrologically, and mechanically induced changes. Model results show a significant increase in water saturations for the Topopah Spring Tuff middle nonlithophysal unit and a lesser (and more uncertain) increase for the Topopah Spring Tuff lower lithophysal unit (Bechtel SAIC Company, LLC, 2003d,e). The main impact of the effect is to decrease the size and duration of the dryout zone around the repository drift. This approach to modeling coupled thermal-hydrological-mechanical effects on flow is described in sufficient detail to allow evaluation.

The DOE technical basis for selecting, including, and excluding specific coupling relationships from the total system performance assessment abstraction is not transparent and traceable in all cases. It is not clear that near-field processes or chemistries omitted from testing or modeling could not lead to significantly shorter waste package lifetimes. One major assumption 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 provided sufficient analysis of the role of chemical reactions in drip shield and waste package corrosion that addresses the criteria used to distinguish between included and excluded couplings or that provides an adequate technical basis for modeling decisions based on those criteria (Schlueter, 2003a). DOE has agreed to address this issue in the context of providing the technical basis for consolidation or establishment of the 11 geochemical bins (Bechtel SAIC Company, LLC, 2003a). DOE has agreed to address the couples considered, the range of chemistries considered, the rationale for including or excluding couples and chemistries, and the limitations of any codes used to develop the bins (Reamer, 2001b).

Although not all these features, events, and processes are fully addressed in currently available reports, DOE has agreed to do so in future reports. It appears the information necessary to assess exclusion and inclusion of features, events, and processes in the abstraction of the quantity and chemistry of water contacting engineered barriers and wasteforms will be available at the time of a potential license application.

The current DOE total system performance assessment submodels describing the in-drift geochemical environment do not consider the effects of drift degradation because current DOE models predict drift degradation is unlikely to occur within 10,000 years. The potential that drift degradation may occur is discussed in Gute, et al. (2003) and Section 5.1.3.2 of this report. Drift degradation, if it occurs, may significantly affect seepage rates, as well as temperatures and relative humidities on waste package surfaces through time (Fedors, et al., 2004), introducing uncertainties in the initiation time, duration, and spatial distribution of aqueous corrosion caused by deleterious brine compositions associated with some deliquescent salts (Pabalan, et al., 2002; Browning, et al., 2004). Estimated temperatures estimates at the waste package and drip shield may increase from 160 °C [320 °F] to the range 230 to 360 °C [446 to 680 °F] if drift degradation results in a rubble pile covering the drip shield (Fedors, et al., 2004). Chemical interactions involving failed in-drift structural materials on drip shield and waste package surfaces will alter the chemical environment relative to scenarios without drift degradation. DOE has not provided sufficient technical bases for screening out the effects of drift degradation on in-drift geochemical environments for corrosion (including an analysis of data and model uncertainties).

Although the effect of drift degradation on seepage has been evaluated, DOE has not evaluated the effect on in-drift temperature and moisture redistribution. DOE evaluates the effects of drift degradation on seepage based on the drift degradation models for lithophysal and nonlithophysal rock (Bechtel SAIC Company, LLC, 2003f, 2001). Flow and seepage calculations are performed for selected representative drift profiles to examine the impact of changes in drift shape on seepage (Bechtel SAIC Company, LLC, 2003c). The drift geometry is modified based on discrete region key-block analysis and uniform enlargement of the drift for the nonlithophysal and lithophysal units. Using 20 realizations of fracture heterogeneity of permeability, most of the simulation results indicated a decrease in seepage with the degraded drift ceilings modified by key-block failure, which is counter-intuitive and needs to be clarified. For the uniformly degraded drift walls for the lithophysal unit, most percolation flux is still diverted around the perfectly cylindrical enlarged drift for most of the considered parameter range; actual seepage fluxes, however, are increased because of the larger footprint of the collapsed drift. Staff conclude the DOE abstraction of the quantity and chemistry of water contacting engineered barriers and wasteforms only partially address the effects of drift degradation. DOE will need to address this model integration issue if an adequate basis for excluding drift degradation is not provided (Section 5.1.3.2).

In summary, DOE has provided information on integration of the abstraction of the quantity and chemistry of water contacting engineered barriers and wasteforms into total system performance assessment. DOE has not yet provided sufficient information on how that abstraction accounts for processes such as thermal-mechanical effects on seepage, thermal-hydrological-chemical coupling, the cold-trap effect, and drift degradation, which could directly affect the volume and nature of water potentially affecting engineered barrier performance. DOE has, however, agreed to provide the additional information.

Overall, the available information, along with agreements between DOE and NRC (Section 5.1.3.3.5), is sufficient to expect that the information necessary to assess model integration of the abstraction of the quantity and chemistry of water contacting engineered barriers and wasteforms with respect to system description and model integration will be available at the time of a potential license application.

5.1.3.3.4.2 Data and Model Justification

In the technical basis document (Bechtel SAIC Company, LLC, 2003a, Appendix A), DOE describes how water data are used in models for evaporation of seepage entering drifts. The range of environments projected to form within drifts as a result of seepage and evaporation is categorized into 11 bins, or water types, characterized by dominant ionic species. Table A–3 of Bechtel SAIC Company, LLC (2003a) shows the bins and corresponding laboratory corrosion test solutions. Although model details were not available at the time this assessment was conducted, it appears DOE has sufficient data on the range of starting seepage water compositions to support the evaporation model.

DOE also projects environments forming in a dust deliquescence scenario, categorized into six bins, based on composition at 98-percent relative humidity. Resulting brine types compared with laboratory corrosion test solutions are shown in Table A–5 of Bechtel SAIC Company, LLC (2003a). DOE, however, has not provided a sufficient characterization of the dust expected to settle on engineered materials in the potential repository drift environment or an analysis how dust could affect the chemistry of water contacting the waste packages and drip shields. DOE characterizes dust samples collected from the Yucca Mountain Exploratory Studies Facility by leaching the dust in water and analyzing the leachates for soluble salt composition. In a similar manner as described for the analysis of seepage solutions, DOE conducted evaporation modeling of 52 leachates using EQ3/6. According to the final compositions of the evaporated solutions, DOE groups 52 leachates into 6 bins and use one water from each bin to encompass the range of leachate compositions (Bechtel SAIC Company, LLC, 2003a, Table 4-14, 2003b). Based on the number of leachates each selected water represents and the total number of leachates, each of the six bins is given a probability of occurrence.

Because the dust samples were collected in the Exploratory Studies Facility in a relatively short time, the majority of dust sampled is rock dust produced during the construction of the tunnel (Peterman, et al., 2003). During the ventilation period of the drift, a large portion of the dust on the surfaces of the waste package and the drip shield is expected to consist of atmospheric dusts. Therefore, the dusts that will be deposited onto the surfaces of the waste package and drip shield during the much longer period of waste emplacement likely would be different from those collected during repository construction. Relevant regional data on atmospheric dust compositions are available in the form of soluble constituents in wet precipitation samples collected in recent years at Death Valley, California (Illinois State Water Survey, 2004). Because of proximity and the wind direction near Yucca Mountain, the atmospheric deposition collected at Death Valley may be considered representative of the atmospheric dusts at Yucca Mountain. Compared with the six median waters selected to represent the dust leachate bins (Bechtel SAIC Company, LLC, 2003a, Table 4-8), the atmospheric precipitation near Yucca Mountain contains higher Ca^{2+} and Mg^{2+} contents relative to other cations. The presence of soluble Ca^{2+} and Mg^{2+} may cause corrosion to initiate at low relative humidities when the temperature is still high, and Alloy 22 has been shown to be subject to corrosion at temperatures between 140 and 160 °C [284 and 320 °F] $CaCl_2$-containing solutions. Therefore, the composition of brines that could evolve on Yucca Mountain atmospheric dusts may not be conservatively bound by the composition of the brines calculated for the six bins of leachates. Although the wet precipitation (rainwater) may not be directly deposited on the surfaces of the potential waste package and drip shield, the constituents in the rainwater are from the atmospheric dusts that may enter the drift and deposited on the surfaces. Therefore, a basis is needed for the assumption that the six bins sufficiently bounded the composition of brines that could deliquesce on dusts that may be deposited onto the surfaces of the drip shield and waste package during the emplacement period.

As mentioned previously, the dust impact analysis provided in the technical basis document (Bechtel SAIC Company, LLC, 2003a) is based on salt compositions derived by evaporation modeling (EQ3/6) of measured water-soluble leachate compositions. The salt compositions derived by EQ3/6 evaporation simulation may be different from those present in the original dusts and could have a different effect on the corrosion of the waste packages and drip shields. A basis is needed for the assertion the evaporative salts represented by the six dust leachate brine evaporation bins sufficiently bound the salts mixed with the dusts that may form on the drip shield and waste package surfaces.

The endpoint brine solutions calculated by DOE for dust deliquescence (Bechtel SAIC Company, LLC, 2003a, Table 4-14) may be divided into two main types: those that contain calcium (Bin 1) and those that do not (Bins 2–6). The experimentally measured deliquescence relative humidity for pure $CaCl_2$ and mutual deliquescence relative humidities for $CaCl_2$-KCl and $CaCl_2$-NaCl mixtures are approximately 15 percent and are independent of temperature between 50 and 70 °C [122 and 158 °F] (Yang, et al., 2004). The data of Yang, et al. also indicate mutual deliquescence relative humidities of the two salt mixtures are only slightly lower than that of the pure $CaCl_2$ salt. Therefore, the mutual deliquescence relative humidity of the salt mixture evaporated from the Bin 1 water is expected to be approximately 15 percent at temperatures near 70 °C [158 °F]. Based on the conductivity data observed at temperatures between 25 and 70 °C [77 and 158 °F] (Yang, et al., 2004), the mutual deliquescence relative humidity will probably remain at approximately 15 percent at higher temperatures for the calcium-containing salt mixtures.

The binned brines that do not contain calcium are dominated by $NaNO_3$ and KNO_3 (Bechtel SAIC Company, LLC, 2003a, Table 4-14). Because NaCl also is a dominant constituent in atmospheric dust (Illinois State Water Survey, 2004), the deliquescence behavior of salts associated with these brines may be bounded by the mutual deliquescence relative humidity of the $NaNO_3$-KNO_3-NaCl system. The experimentally measured mutual deliquescence relative humidity for this system is approximately 70 percent at 25 °C [77 °F] and 43 percent at 86 °C

[187 °F] (Yang et al., 2004, 2002). Based on the temperature-dependence trend, the mutual deliquescence relative humidity for this three salt mixture is as low as 29 percent at 130 °C [266 °F]. These values are in agreement with the DOE results obtained using the EQ3/6 model (Bechtel SAIC Company, LLC, 2003a, Figure I–6); DOE model results for the $NaNO_3$-KNO_3-NaCl-KBr system yielded similar mutual deliquescence relative humidities above 100 °C [212 °F].

Solutions containing calcium and magnesium salts are not stable near their boiling temperatures at ambient pressure because they undergo hydrolysis and produce acidic gases (Pulvirenti, et al., 2003). Thermogravimetric measurements by Hailey and Gdowski (2003) also showed pure $CaCl_2$ solution saturated at 22.5-percent relative humidity decomposes in approximately 20 hours at 150 °C [302 °F]. The $CaCl_2$ solution also appears to decompose slowly in the first 120 hours at 125 °C [257 °F]. Thus, there is a threshold temperature for the calcium containing salt to decompose, below 150 °C [302 °F] and probably close to 125 °C [257 °F].

Based on the previous analysis, the bounding minimum deliquescence relative humidity for Yucca Mountain dust salts may be the mutual deliquescence relative humidity of calcium-containing salts for temperatures below approximately 125 °C [257 °F] (i.e., their threshold decomposition temperature) and the mutual deliquescence relative humidity of $NaNO_3$-KNO_3-NaCl salts for temperatures above approximately 125 °C [257 °F]. Consequently, no aqueous solution will be formed by deliquescence at relative humidities lower than approximately 15 percent below approximately 125 °C [257 °F] or at relative humidities lower than approximately 30 percent above 125 °C [257 °F].

Assuming the prevailing pressure at the potential repository horizon is 0.89 bar (CRWMS M&O, 2001b), the maximum achievable relative humidities range from 100 percent at 96 °C [205 °F], to 38 percent at 125 °C [257 °F], to 30 percent at 133 °C [271 °F], and to 12 percent at 167 °C [333 °F] (U.S. Department of Commerce, 1996, NIST Steam Table Version 2.2). When the in-drift relative humidity is at the mutual deliquescence relative humidity, the brine formed will have the eutectic composition. When the in-drift relative humidity increases and passes the mutual deliquescence relative humidity of the salt mixture, however, the composition of the brine will change as a function of the in-drift relative humidity.

It has been shown nitrate is an effective inhibitor for localized corrosion of Alloy 22 in chloride-containing environments (Section 5.1.3.1). DOE concludes all six binned model brines will have a low molar chloride-to-nitrate ratio (Cl^-/NO_3^- = 0.002 to 2) for relative humidities within the drift between 26 and 60 percent, temperatures between 40 and 140 °C [104 and 284 °F], and pCO_2 between 10^{-2} and 10^{-4} bar (Bechtel SAIC Company, LLC, 2003a, Table 4-10, Figures 4-17 and 4-18). Compiled atmospheric dust data from the Illinois State Water Survey (2004) also yield low average molar chloride-to-nitrate ratios: 0.29 for the reported concentration measurements and 0.24 for the reported deposition amounts.

It appears the conditions used by DOE to model the chemical compositions of the six bins of waters for relative humidities between 26 and 60 percent (Bechtel SAIC Company, LLC, 2003a, Table 4-10, Figures 4-17 and 4-18) do not include the conditions at the mutual deliquescence relative humidities of the salt mixtures. The model-calculated composition for the NaCl-KNO_3 salt mixture (Bechtel SAIC Company, LLC, 2003a, Figure I–7) indicates the Cl^-/NO_3^- ratio also is low for the NaCl-KNO_3 system at the mutual deliquescence relative humidity conditions (eutectic composition of the NaCl-KNO_3 system) in the temperature range 25–125 °C [77–257 °F]. It appears, however, the Cl^-/NO_3^- ratio is still not known for the calcium-containing salts under mutual deliquescence relative humidity conditions. The brine solutions formed by the calcium-containing salts at the mutual deliquescence relative humidity may have lower nitrate-to-chloride ratios than the values predicted by the EQ3/6 model. The chemistry of the brines formed directly from the dust deposits, including the atmospheric dusts, may be different from the chemistry of the concentrated brines DOE modeled from evaporation of the leachates. Staff conclude DOE has not bounded the range of brine chemistries that may form by deliquescence, but has agreed to do so (Reamer, 2001b).

The previous version of this report (NRC, 2002) discussed numerous issues of model justification DOE agreed to address. As discussed in the following text, in some cases, staff found sufficient information in the in-drift chemistry technical basis document (Bechtel SAIC Company, LLC, 2003a) or other reports as cited. In other cases, the DOE reports were not available at the time of this assessment.

DOE has provided a technical basis for assumptions in its in-drift chemical models that do not explicitly treat chemical kinetics (Bechtel SAIC Company, LLC, 2003a). The basis for this approach includes the applicability of equilibrium models caused by long timescales and rapid reactions, the approximate treatment of kinetics by suppression of precipitation of certain phases, and consideration of rates of corrosion. In addition, arguments are offered that time-independent (i.e., not kinetic) abstractions of chemical conditions are required for use in performance assessment modeling. Furthermore, in the technical basis document, DOE provides clear recognition of the limitations of models that neglect kinetics of in-drift chemical processes. There is sufficient information available to evaluate DOE approach to accounting for chemical kinetics.

DOE has considered changes in local water and gas chemistries resulting from interactions with introduced and engineered materials, such as steel components, along preferential flow pathways. DOE concludes (Bechtel SAIC Company, LLC, 2003a, Appendix J) these materials will have only minor effects on seepage water chemistry. The technical basis document (Bechtel SAIC Company, LLC, 2003a) provides sufficient information to evaluate capillary pressure effects, low-relative humidity salt deliquescence behavior, use of mixed salts in deliquescence models, and comparison of model outputs with drift-scale test results.

Heat and mass loss out of the bulkhead of the Drift-Scale Heater Test reduced the usefulness of the test results to support models estimating water distribution in and near heated drifts. The heat and mass losses through the bulkhead might (i) mask preferential flow along fractures breaching the dryout zone and (ii) create additional data uncertainty should Drift Scale Heater Test model-derived parameters be used in other seepage process models or abstractions for the performance assessment. DOE provided a discussion (Brocum, 2002) on the technical basis for understanding heat and mass losses through the bulkhead. Schlueter (2003b) stated the Drift-Scale Heater Test results could not be used to support models evaluating the possibility of preferential flow breaching the dryout zone, nor to support the estimation of parameter values used in thermohydrological models used for performance assessment. Furthermore, DOE stated (Brocum, 2002) the effects of heterogeneity on condensate drainage and heat and mass losses through the bulkhead would be addressed in the design of the Cross-Drift Thermal Test. At the time of this assessment, the current DOE approach to interpreting Drift-Scale Heater Test results was not available.

Efficacy of preclosure ventilation is important for estimating the initial temperature conditions for postclosure thermohydrological modeling and for the design objective of maintaining pillar temperatures below boiling to allow for condensate drainage between emplacement drifts. In Schlueter (2002), NRC notes a uniform reduction factor of 0.70 is supported by results from the NRC ventilation model (Painter, et al., 2001). Furthermore, Schlueter (2002) states more detailed model support would be needed if the reduction factor were increased. The NRC ventilation model is similar to the MULTIFLUX model used by DOE to support the simplified calculations of the ANSYS model described in Blink (2002). Recent documents (Bechtel SAIC Company, LLC, 2003e,f) use a reduction factor of 0.9 and DOE no longer uses the MULTIFLUX model to support the ANSYS calculations, thus NRC is reviewing the estimated reduction factors and model support discussed in the ventilation report (Bechtel SAIC Company, LLC, 2003g). At the time of this assessment, the DOE license application design for the ventilation system is not known to NRC, but DOE has agreed to provide information on the ventilation model (Reamer, 2001a).

The DOE neglect of mineral precipitation in the vicinity of the emplacement drifts is based on the results of thermal-hydrological-chemical simulations described in the technical basis document (Bechtel SAIC Company, LLC, 2003a). These simulations indicate silica precipitation is likely to occur in a narrow zone, however, the resulting change in hydrological properties will have a benign or beneficial effect on seepage. 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. DOE agreed to provide additional documentation on the simulations pertaining to quantity of unreacted solute mass trapped in the dry computational cells in TOUGHREACT simulations, as well as on how this mass would affect precipitation and the resulting change in hydrological properties (Reamer, 2001b). Bechtel SAIC Company, LLC (2003a) indicates the TOUGHREACT software has been changed to improve handling of mineral precipitation in the boiling zone, but information required to determine if the simulations adequately represent mineral precipitation at the boiling front is in documents not available at the time of this assessment.

More generally, the technical basis document (Bechtel SAIC Company, LLC, 2003a) does not provide a detailed description for Revision 2 of the thermal-hydrological-chemical drift-scale simulation model. Some sensitivity analyses are conducted with Revision 1 of the model, whereas others use Revision 2. The technical basis document indicates significant changes were made to the TOUGHREACT code and to the model input from Revision 1 to Revision 2 of the drift-scale coupled-process model, but the report does not provide a description of the changes in the model or the bases for them. DOE has agreed to provide adequate information on the sources of uncertainty and variability in near-field environment models (Reamer, 2001b,c).

In currently available DOE reports, the range of inputs used in the latest version of the drift-scale coupled process models is not sufficiently transparent to assess the approach and basis for including the edge effects as predicted by the three-dimensional, mountain-scale, thermal-hydrological model. At the time of this assessment, the report containing the detailed information was not yet available.

In summary, DOE has provided information that will allow evaluation of data supporting the binning of water chemistries modeled to contact, and potentially corrode, engineered barriers. DOE, however, has not provided sufficient characterization of the dusts that may settle or form on surfaces and deliquesce to form brines. At the time this assessment was conducted, detailed descriptions of quantitative models used to calculate water chemistry evolution as a result of coupled thermal-hydrological-chemical processes on the drift scale, needed for evaluation of model results and use in the abstraction were not available. DOE has not provided updated information on heat and mass loss in the Drift-Scale Heater Test, and the ventilation model cannot be evaluated until a final design is available. DOE has agreed to provide the information needed on each of these topics.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.3.5), is sufficient to expect that the information necessary to assess quantity and chemistry of water contacting engineered barriers and wasteforms with respect to data being sufficient for model justification will be available at the time of a potential license application.

5.1.3.3.4.3 Data Uncertainty

The binning approach taken by DOE appears reasonable to address the uncertainty in total system performance assessment predictions caused by the propagation of uncertainties from process-level models. DOE estimation of engineered barrier performance relies directly on the calculated probability of corrosive waters reaching the barriers. The DOE assessment of the probability of seepage water compositions relies mainly on the results of parameter value sensitivity studies performed on coupled thermal-hydrological-chemical models. The results of the binning process, however, may be quite sensitive to the composition of the median representative water selected from each group. Uncertainty includes composition of the starting unsaturated zone water. Even for ambient conditions (Browning, et al., 2000), water compositions in the unsaturated zone will vary, depending on the types of materials encountered along a particular flow pathway and the duration of those interactions. In current DOE reports, the impact of the composition of the median representative water on the probability of forming conditions suitable for localized corrosion is not transparent. DOE has agreed to provide more detailed technical bases of this approach in future reports. DOE also has agreed to provide details concerning its treatment of uncertainties related to the quantity and chemistry of in-drift waters and to provide additional technical bases for the assumption that the probability of seepage water compositions can be reasonably determined from sensitivity studies of specific coupled thermal-hydrological-chemical models.

Appendix K, of the technical basis document (Bechtel SAIC Company, LLC, 2003a, Page K–5), states small changes in temperature may lead to differences of several orders of magnitude in water and gas chemistries. Water samples collected during the test, however, were obtained from zones that were hotter than the temperatures given for the samples (Bechtel SAIC Company, LLC, 2003a, Page K–12). Therefore, the data on water chemistry, measured at laboratory temperature conditions, may not reflect the water chemistry of the field conditions. DOE needs to provide information on temperature corrections made to the water chemistry data or provide technical justification for neglecting temperature effects on the water chemistry.

DOE addressed possible causes of data uncertainties in Section K.4 of the technical basis document (Bechtel SAIC Company, LLC, 2003a); however, no statistical measures of parameter uncertainties were provided, except for pH and temperature. DOE has agreed to provide confidence intervals where comparisons between predicted values and field measurements are made (Reamer, 2001b). Examples of such comparisons are shown in the technical basis document (Bechtel SAIC Company, LLC, 2003a, Figures 3-8 and 3-9).

The current DOE thermohydrological models used to support seepage fluxes do not account for measurement error, bias, and scale dependence in the saturation, water potential, and pneumatic pressure data used to calibrate the drift and mountain-scale hydrological property sets. With complex flow processes in fractured rock and little data to support model results, the ensemble effect of uncertainty in these topics may be important to model results. Because the thermohydrological models use the ambient hydrological property sets, it is important to evaluate the uncertainties described in Section 5.1.3.6.4.3 for thermohydrological models. Effects of fracture heterogeneity on thermal seepage and on in-drift temperature and relative humidity were specifically assessed using thermohydrological models and are discussed in more detail next.

The effects of fracture permeability heterogeneity on seepage for thermally perturbed conditions were examined in Bechtel SAIC Company, LLC (2003c). A two-dimensional, dual-permeability thermal seepage model was run with realizations of fracture permeability generated using statistical data from air injection testing in the Exploratory Studies Facility and Enhanced Characterization of the Repository Block drift. High-permeability zones such as the intensely fractured zone and faults were excluded from analyses because a standoff distance was expected by DOE to be included in the design criteria. The thermal seepage model incorporates the effect of the vaporization and capillary barriers. The inclusion of fracture heterogeneity, based on measurements, in the thermal seepage model addresses concerns of one DOE and NRC key technical issue agreement. There are inconsistencies in seepage results between the thermal seepage and ambient seepage models when considering only the capillary barrier. Grid refinement and parameterization, particularly in the zone immediately above the drift, are likely the causes for the inconsistencies between the ambient seepage model results and those of the thermal seepage model when run without a heat load. NRC staff will continue to review this issue as more information becomes available in the future.

The effects of heterogeneity of fracture permeability also are examined in the drift-scale thermohydrological model of the Multiscale Thermohydrologic Model (CRWMS M&O, 2001b). The DOE study on the effect of three-dimensional fracture heterogeneity indicates the thermohydrological conditions calculated by the model for the no-backfill case would not be changed during the boiling or postboiling period by adding the influence of drift-scale fracture heterogeneity. The effect of including fracture heterogeneity in the model is to increase slightly the relative humidity and evaporation rate on the drip shield in the postboiling period, because of increased dripping on the drip shield. Without fracture heterogeneity, dripping occurs only below the zones of highest net infiltration in the upper bound case of the glacial-transition climate. The fracture realizations in CRWMS M&O (2001b) are stated as extreme cases presented to illustrate the effect of heterogeneity; therefore, it should not be concluded that seepage would increase in the basecase. The staff review indicates the effect of abundant (approximately 25 percent by volume) and large lithophysae on diversion around drifts and seepage has not yet been sufficiently evaluated by DOE. From a geospatial perspective, the bounds that should be used are not clear for statistical parameters or models in the generation of heterogeneous fracture permeabilities. It also is not clear whether the in-drift conditions will be significantly affected by the thermohydrological conditions at the drift ceiling as estimated using the statistical parameters, grid resolution, and conceptualization of seepage and flow presented in CRWMS M&O (2001b). Because the available drift-scale thermohydrological model (CRWMS M&O, 2001b) produced seepage results that are inconsistent with ambient seepage model results, the adequacy of integration between the thermohydrological and ambient seepage models will need to be reviewed when the report describing the updated thermohydrological model is released, as planned by DOE.

In summary, DOE has not provided sufficiently detailed information how uncertainties in data supporting coupled thermal-hydrological-chemical models were propagated in the total system performance assessment abstraction of the probabilities of different water compositions contacting engineered barriers. In addition, uncertainty in fracture characteristics above the drift as it affects seepage calculations was not addressed in sufficient detail in available reports. DOE has agreed to provide this information, for use in evaluating DOE simulations of conditions that could potentially lead to engineered barrier corrosion.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.3.5), is sufficient to expect that the information necessary to assess the DOE abstraction of the quantity and chemistry of water contacting engineered barriers and wasteforms with respect to data uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.3.4.4 Model Uncertainty

Information is needed about the DOE treatment of model uncertainties related to (i) timing and extent of drift degradation, (ii) effects of the cold-trap process and condensation on chemistry of water contacting waste packages, and (iii) evolution of in-drift water chemistry resulting from reactions with introduced materials. The composition of water contacting drip shields and waste packages, and therefore directly affecting engineered barrier performance, may vary significantly through time and with drift location as a result of these processes. For example, seepage can be affected strongly by drift degradation, which has not been considered in the DOE thermal-hydrological and thermal-hydrological-mechanical models simulating drift seepage.

Condensed water is expected to have a composition quite different from seepage water compositions (Pulvirenti, et al., 2003; Browning, et al., 2004) and may mix with seepage waters or interact chemically with natural and introduced materials to varying extents through time in different repository locations. DOE has not sufficiently documented its expectations about either the volumetric contribution of condensed water in different drift locations through time or the effects of condensate on in-drift water compositions and repository performance. Current DOE evaluations of the uncertainties in the in-drift geochemical environment resulting from the cold-trap and condensation processes are not sufficient. DOE has agreed to provide information on model uncertainties (Reamer, 2001b).

Model uncertainties with respect to the range of local chemistry conditions at the drip shield and waste package surfaces are addressed by DOE using a probability model. As discussed in Section 5.1.3.3.4.1, the water chemistries selected by DOE to be representative of seepage waters entering the drift through time are based on five realizations of the drift-scale coupled processes models; the results are abstracted into 11 groups (bins) of like water types. Lookup tables of solution compositions are constructed by simulating the evaporation of the

11 seepage bin chemistries at different temperatures and pCO_2 values. DOE used the time-integrated probability of the occurrence of these 11 bins or water types as seepage from the crown of the drift to claim calcium-chloride type brines (bins 1 and 2), which are potentially corrosive to Alloy 22, have either zero or low probability of occurrence as seepage water and, therefore, are not expected to contact the drip shield or waste package surfaces. A study by Rosenberg, et al. (2001), however, shows evaporative concentrations of simulated Yucca Mountain pore water, with a composition based on values reported by Sonnenthal, et al. (1998), result in a calcium-chloride-type brine. Summary descriptions of the DOE approach are provided in Appendixes E and G of the technical basis document (Bechtel SAIC Company, LLC, 2003a). The detailed thermal-hydrological-chemical modeling results that support the DOE binning approach and its assertion that the occurrence of calcium-chloride brines have low probabilities are presented in reports not available at the time of this assessment. DOE has agreed to provide its detailed analyses of uncertainty and variability in the binning approach to the chemistry of water contacting the drip shields and waste packages, including justification for the choice of a 20,000-year time interval used to define the time-integrated probability of occurrence of water of each bin (Reamer, 2001b).

With respect to uncertainty in seepage models, alternative models for water movement in the thermally perturbed zone above the drifts include preferential flow along fracture planes breaching the dryout zone (Phillips, 1996; Birkholzer, 2003) and the ponded model based on heterogeneity of fracture properties. Bechtel SAIC Company, LLC (2003g) summarizes analyses using the alternative models and considers these to reflect the upper bound uncertainty for seepage (i.e., not representative of the basecase). The detailed technical bases needed to support these models are to be provided by DOE in documents not available at the time of this evaluation.

In summary, at the time of this assessment, detailed information was not available about the propagation of model uncertainties in the DOE abstraction of water chemistry bin probabilities and seepage rates. The probabilities calculated for potentially corrosive waters directly affect DOE simulations of engineered barrier performance. Detailed model uncertainty analyses are expected in future DOE reports.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.3.5), is sufficient to expect that the information necessary to assess the abstraction of the quantity and chemistry of water contacting engineered barriers and wasteforms with respect to model uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.3.4.5 Model Support

DOE used the EQ3/6 code to calculate the deliquescence behavior of the salt mixtures assumed to form on the drip shield and waste package surfaces. DOE has validated EQ3/6 using limited experimental data on selected simple systems. For example, Table 4-3 of the technical basis document (Bechtel SAIC Company, LLC, 2003a) compares calculated and experimental values for pure salts only. Table 4-11 of the same report compares calculated and experimental values for salt mixtures, but $NaCl+NaNO_3+KNO_3$ is the most complex mixture in the table. In contrast, the In-Drift Precipitates/Salts Model (Bechtel SAIC Company, LLC, 2003b) is used for more complex systems containing K^+, Na^+, and Ca^{2+} cations and Cl^-, NO_3^-, and CO_3^- anions. Seawater evaporation data also are used to validate the model. Data used in the validation, however, extended only to an ionic strength of 10 molal (Bechtel SAIC Company, LLC, 2003a, Figure 4-5), whereas the model was used to predict chemical compositions of brine solutions at low relative humidities that correspond to ionic strengths close to 30 molal (Bechtel SAIC Company, LLC, 2003a, Figures 4-12, 4-15, and E–9). The modeled mutual deliquescence relative humidity for the eutectic $Ca(NO_3)_2$-NaCl-$NaNO_3$-KNO_3-KBr system at 25 °C [77 °F] was 39 percent (Bechtel SAIC Company, LLC, 2003a, Table I–2 and Figure I–5). Recently measured deliquescence relative humidity for the $CaCl_2$, the $CaCl_2$-KCl, and the $CaCl_2$-NaCl systems is approximately 15 percent and is independent of temperature in the range 50–70 °C [122–158 °F] (Yang, et al., 2004). The conductivity data shown by Yang, et al. (2004) also indicate the mutual deliquescence relative humidities for both the $CaCl_2$-NaCl and the $CaCl_2$-KCl systems are lower than 17 percent at 25 °C [77 °F]. Because the mutual deliquescence relative humidity is always lower than the deliquescence relative humidity of the individual solutes (Ge, et al., 1998), the model calculated mutual deliquescence relative humidity is significantly higher than the experimentally measured mutual deliquescence relative humidity for the calcium-containing salt mixture. DOE has agreed to provide additional model validation in future reports (Bechtel SAIC Company, LLC, 2003a).

The composition of seepage water is likely to be influenced by the phases in the unsaturated fractured rock with which it reacts. Geochemical modeling is used to predict the composition of water seeping into the drifts. Sources of uncertainty include choices regarding 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 provided some evidence to support the model of fracture/matrix interaction by overcoring in the Single Heater Test, but has not provided relevant Drift-Scale Heater Test results at the time of this assessment was conducted. Comparison of pre- and post-test mineral assemblages, evidence of mineral alteration, and redistribution can be used to support predictive models.

The effects of the cold-trap process and of natural convection on temperature distribution in drifts have not been incorporated into performance analyses of in-drift moisture conditions (Bechtel SAIC Company, LLC, 2003a). Evidence suggests condensation is occurring behind the bulkhead of the Enhanced Characterization of the Repository Block, where conditions are unventilated and relative humidity is high (Bechtel SAIC Company, LLC, 2003h). DOE postponed the experiments pertaining to the distribution of condensation in drifts and it is not clear if sufficient data were obtained by DOE from the closed portion of the Enhanced Characterization of the Repository Block drift. Bechtel SAIC Company, LLC (2003a) notes additional information will be included in a future analysis and model report on natural convection and condensation.

DOE model abstractions on flow and seepage neglect any effects of mineral precipitation near emplacement drifts, using numerical simulations to justify the abstraction. Simulations summarized in the report Drift Scale Coupled Processes (DST and THC) Models Revision 1 using the TOUGHREACT Version 2.0 code show negligible mineral precipitation in the vicinity of emplacement drifts (CRWMS M&O, 2001a). More recent simulations summarized in the in-drift chemistry technical basis document (Bechtel SAIC Company, LLC, 2003a) and obtained with TOUGHREACT Version 3.0 show significant silica deposition, which results in one to two orders of magnitude reduction in permeability. The zone of silica deposition, however, is thin and in a location where it will not adversely affect drift seepage. To support these model abstractions, DOE uses the same numerical models to simulate the Drift-Scale Heater Test. Once the models have been shown to reproduce the results of the Drift-Scale Heater Test, the models are then used to simulate repository conditions to support the model abstraction. DOE plans to provide additional information on the model comparisons in future reports.

In summary, at the time this assessment was conduced, sufficient information was not available for evaluation of support for models of deliquescence and thermal-hydrologic-chemical effects on seepage, but this information is expected to be provided in the future.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.3.5), is sufficient to expect that the information necessary to assess model support for the DOE abstraction of the quantity and chemistry of water contacting engineered barriers and wasteforms with respect to model abstraction output being supported by objective comparisons will be available at the time of a potential license application.

5.1.3.3.5 Summary and Status of Key Technical Issue Subissues and Agreements

Table 5.1.3.3-1 provides the status of all key technical issue subissues, referenced in Section 5.1.3.3.2, for the Quantity and Chemistry of Water Contacting Engineered Barriers and Waste Forms Integrated Subissue. The table also provides the related DOE and NRC agreements pertaining to the Quantity and Chemistry of Water Contacting Engineered Barriers and Waste Forms Integrated Subissue. The agreements listed in the table are associated with one or all five review methods discussed in Section 5.1.3.3.4. Note the status and the detailed agreements pertaining to all the key technical issue subissues are provided in Table 1.2-1 and Appendix A.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (e.g., specified testing or analyses), indicates that information necessary to begin a technical review will likely be available at the time of a potential license application.

Table 5.1.3.3-1. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreement*
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

Table 5.1.3.3-1. Related Key Technical Issue Subissues and Agreements (continued)			
Key Technical Issue	**Subissue**	**Status**	**Related Agreement***
Evolution of the Near-Field Environment	Subissue 4—Effects of Coupled Thermal-Hydrological-Chemical Processes on Radionuclide Transport through Engineered and Natural Barriers	Closed-Pending	ENFE.4.01 through 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	TEF.1.01
	Subissue 2—Thermal Effects on Temperature, Humidity, Saturation, and Flux	Closed-Pending	TEF.2.01 TEF.2.02 TEF.2.04 through TEF.2.08 TEF.2.10 TEF.2.11
Container Life and Source Term	Subissue 1—The Effects of Corrosion Processes on the Lifetime of the Containers	Closed-Pending	CLST.1.01
	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 Engineered Barrier Subsystem	Closed	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

Table 5.1.3.3-1. Related Key Technical Issue Subissues and Agreements (continued)			
Key Technical Issue	**Subissue**	**Status**	**Related Agreement***
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
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 review methods.

Note: Key Technical Issue Agreement GEN.1.01 pertains to multiple integrated subissues, as well as some specific issues related to this integrated subissue.

5.1.3.3.6 References

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5.1.3.4 Radionuclide Release Rates and Solubility Limits

5.1.3.4.1 Description of Issue

The Radionuclide Release Rates and Solubility Limits Integrated Subissue addresses the release of radionuclides from the engineered barrier system to the geosphere. The relationship of this integrated subissue to other subissues is depicted in Figure 5.1.3.4-1. The overall organization and identification of all the integrated subissues are depicted in Figure 1.1-2. The DOE description and technical bases for abstraction of radionuclide release rates and solubility limits were documented previously in the total system performance assessment site recommendation (CRWMS M&O, 2000a) and several supporting analysis and model reports. Revisions to some of these analysis and model reports recently were published (CRWMS M&O, 2003a–d). This section documents the current NRC understanding of the abstractions DOE developed to incorporate radionuclide release and solubility limits into its total system performance assessment. The evaluation is focused on those aspects most important to repository safety based on the risk insights gained to date, including Appendix D. The scope of the assessment presented here is limited to examining whether data gathered and methodology developed by DOE are likely to be adequately documented for the staff to undertake a detailed technical review. This assessment is not a regulatory compliance determination review of a potential license application.

5.1.3.4.2 Relationship to Key Technical Issue Subissues

The Radionuclide Release Rates and Solubility Limits Integrated Subissue incorporates subject matter previously described in the following 10 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)

- 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 Coupled 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)

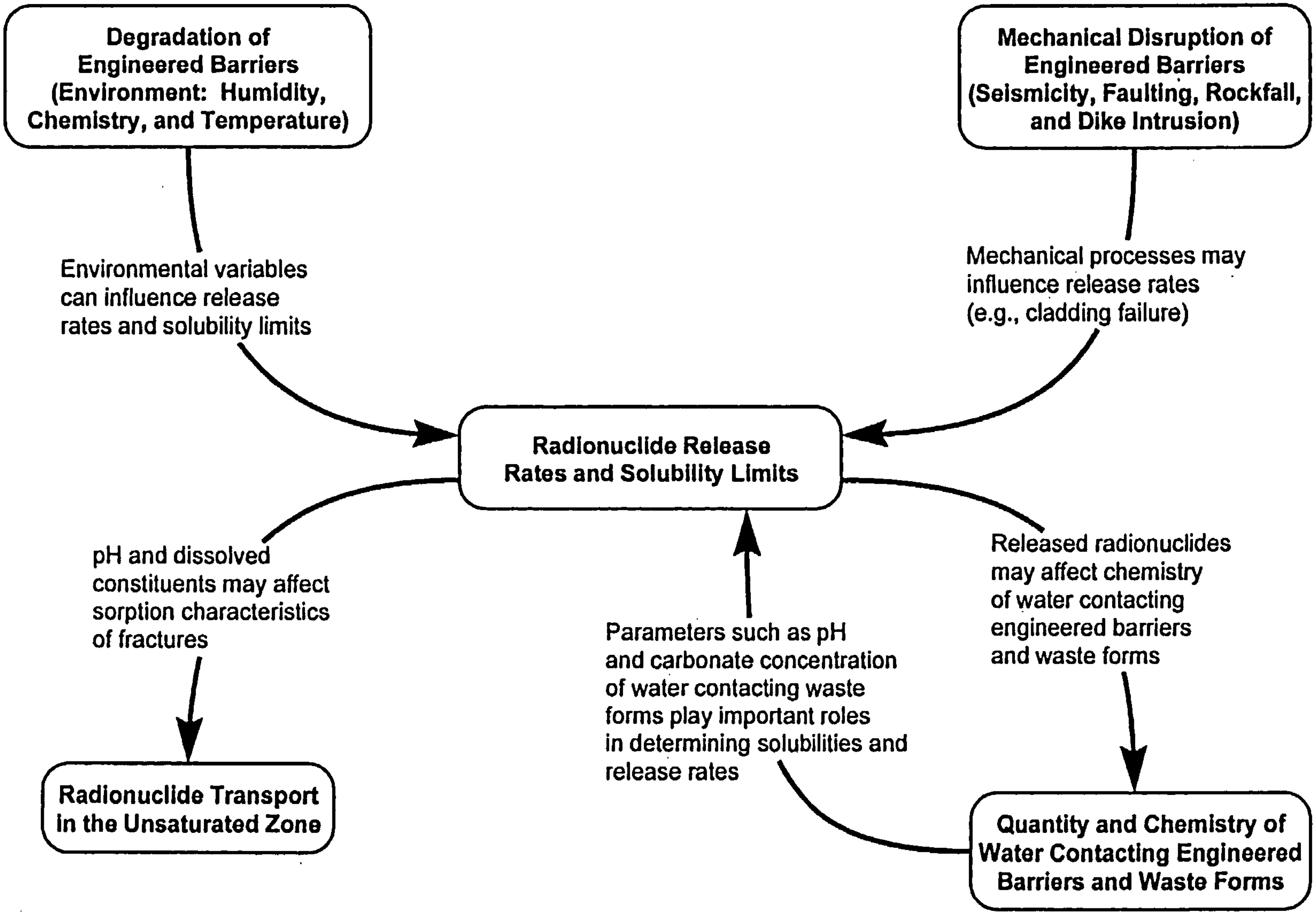

Figure 5.1.3.4-1. Diagram Illustrating the Relationship Between Radionuclide Release Rates and Solubility Limits and Other Integrated Subissues. Material in Bold Is Identified in the Text.

- 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 previous versions of issue resolution status reports and also were the bases for technical exchanges with DOE where agreements were reached on the additional information DOE would provide to NRC. The resolution status of this 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 Section 5.1.2.2 and are not repeated here. The subsequent sections incorporate applicable portions of these key technical issue subissues.

5.1.3.4.3 Importance to Postclosure Performance

One aspect of risk informing of the NRC understanding of postclosure repository performance is to determine how this integrated subissue is related to the DOE repository safety strategy. Risk insights pertaining to radionuclide release rates and solubility limits indicate the wasteform degradation rate, cladding degradation, solubility limits, and the effect of colloids on waste package releases are of medium significance to waste isolation. The mode of release from the waste package, flow and transport through the invert, and nuclear criticality are assigned low significance. The details of the risk insights ranking are provided in Appendix D.

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 wasteform 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 wasteform 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 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 system.

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 wasteform performance. Cladding performance pertains to the role of cladding in limiting water contact and subsequent dissolution of the spent nuclear fuel wasteform. Wasteform performance relates to the rate of mobilization of radionuclides caused by degradation of the wasteform itself (e.g., the irradiated uranium oxide matrix or high-level waste glass wasteform).

5.1.3.4.4 Technical Basis

NRC developed a review plan (NRC, 2003) consistent with the acceptance criteria and review methods found in the previous issue resolution status reports. A status assessment of the DOE approaches for including abstractions of radionuclide release rates and solubility limits in its total system performance assessment is provided in the following subsections. This assessment used the review methods identified in Section 2.2.1.3.4, Radionuclide Release Rates and Solubility Limits, of the review plan (NRC, 2003) and is risk informed based on insights documented in Appendix D. Several DOE abstractions pertain to the Radionuclide Release Rates and Solubility Limits Integrated Subissue. 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 System Flow and Transport. Staff comments for each topic are organized according to the five review methods identified in Section 2.3: (i) Model Integration (including system description), (ii) Data and Model Justification, (iii) Data Uncertainty, (iv) Model Uncertainty, and (v) Model Support.

5.1.3.4.4.1 Radionuclide Inventory

5.1.3.4.4.1.1 Model Integration

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, the 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 the 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 to the reasonably maximally exposed individual at the point of compliance 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, 2000a) 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 approach appears to account for all waste types that will be emplaced in the repository, with bases for the radionuclide source term in the various fuel types, and seems complete in this regard. Projections of radionuclide inventory include consideration of the greater-than-10-year trend in the nuclear industry to increase burnup of commercial fuel.

Overall, the available information is sufficient to expect that the information necessary to assess the effects of radionuclide inventory on radionuclide release rates and solubility limits with respect to system description and model integration will be available at the time of a potential license application.

5.1.3.4.4.1.2 Data and Model Justification

Sufficient data are available on the inventory of radionuclides in the waste to evaluate 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 (CRWMS M&O, 1999a,b), using SAS2H computer code sequence of the SCALE Version 4.3 code system (Oak Ridge National Laboratory, 1995). Inventory projections for the DOE spent nuclear fuel were derived for representative fuel types using the ORIGEN2 code (Croff, 1980). The spent nuclear fuel characterization information for all the DOE-owned spent nuclear fuel is reported in DOE (2003). Inventory projections for high-level waste are taken from the best available information for each vitrification site (DOE, 2002). With respect to sufficient data for model justification, no information (beyond that currently available) likely will be required for regulatory decisionmaking at the time of a potential license application.

Overall, the available information is sufficient to expect that the information necessary to assess the radionuclide inventory abstraction with respect to data being sufficient for model justification will be available at the time of a potential license application.

5.1.3.4.4.1.3 Data Uncertainty

DOE uses values for radionuclide inventories that appear to account for uncertainty and variability. No additional information likely is to be needed regarding the characterization and propagation of data uncertainty through the abstraction of waste inventory (CRWMS M&O, 1999a,b).

Overall, the available information is sufficient to expect that the information necessary to assess the radionuclide inventory abstraction with respect to data uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.4.4.1.4 Model Uncertainty

DOE provided information on the models used to generate radionuclide inventories (CRWMS M&O, 1999a,b). One model uncertainty in the DOE approach is that, in the radionuclide screening process, seven radionuclides with low solubilities were assigned to an insoluble group of radionuclides. Currently, NRC is evaluating some inconsistencies in data on radionuclide solubilities reported by DOE (Bechtel SAIC Company, LLC, 2003a) and the list of low solubility radionuclides (CRWMS M&O, 2001a) and their potential effect on receptor dose. No additional information is likely to be needed regarding the characterization and propagation of model uncertainty through the abstraction of the waste inventory.

Overall, the available information is sufficient to expect that the information necessary to assess the radionuclide inventory abstraction with respect to model uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.4.4.1.5 Model Support

DOE provided information on its total system performance assessment analyses for each type of waste stream, including its use of the data on reactor configuration, reactor history of the fuel, initial fuel enrichment, burnup, and age of the waste to make projections of radionuclide inventory for commercial spent nuclear fuel, DOE-owned spent nuclear fuel, and high-level waste glass (CRWMS M&O, 2001a, 1999a,b). No additional information is likely to be needed regarding model abstraction output.

Overall, the available information is sufficient to expect that the information necessary to assess the radionuclide inventory abstraction with respect to model abstraction being supported by objective comparisons will be available at the time of a potential license application.

5.1.3.4.4.2 In-Package Chemistry

5.1.3.4.4.2.1 Model Integration

Estimation of the in-package chemical environment is integral to the DOE calculations of wasteform degradation rate, radionuclide solubility, and colloid formation and stability. Risk insights pertaining to radionuclide release rates and solubility limits indicate wasteform degradation rate, solubility limits, and effect of colloids on waste package releases are of medium significance to waste isolation. The in-package chemistry model and the in-package chemistry model abstraction described in Bechtel SAIC Company, LLC (2003b) consider chemical interactions of water with the waste package materials and the wasteform for commercial spent nuclear fuel, codisposed high-level waste glass, and N-Reactor spent nuclear fuel. The interactions of water with waste package materials and wasteforms are simulated as a function of time using the EQ3/6 code by assigning kinetic rates to the reactants. Waste package materials included in the EQ3/6 simulations are the steel and aluminum alloys present in commercial spent nuclear fuel and N-Reactor spent nuclear fuel, such as Types 304L and 316 stainless steels, A516 carbon steel, and aluminum Alloy-1100. The waste package materials were assigned fixed values of corrosion rates and the high-level waste glass was given a dissolution rate dependent on pH and temperature. The equation for the dissolution rate for commercial spent nuclear fuel is a function of pH, temperature, O_2 partial pressure, and aqueous carbonate concentration, whereas the equation for the degradation rate of N-Reactor spent nuclear fuel is dependent only on temperature.

Two different water ingress models were used: (i) water vapor ingress and subsequent condensation with film formation (i.e., the water condensation model) and (ii) seepage dripping where seepage water enters the waste package, forms a film, reacts with the components inside the waste package, and exits the waste package. For the seepage dripping model, three water compositions were used as the initial inputs to the EQ3/6 simulations: the composition of J–13 Well water and the compositions of two porewater samples, referred to as Ca-porewater and Na-porewater, obtained from core specimens proximal to the repository. All three waters are dilute, and the latter two are quite close in composition except in the concentration of sodium and magnesium ions. Although DOE asserted that the two porewaters bounded the porewater compositions (Bechtel SAIC Company, LLC, 2003b), the compositions do not bound the range in chemistry of water that potentially can enter the waste package. For example, the evaporation of seepage water that contacts the hot waste package or the deliquescence of salts present on the waste package surface could result in waters with high concentrations of dissolved species (Bechtel SAIC Company, LLC, 2003c). The evolution of in-package chemistry resulting from the interaction of waste package components with these high ionic strength waters likely would be different from that because of interaction with dilute waters considered in the DOE abstraction of in-package chemistry. If corrosion is the only mechanism for degradation of the waste packages, breach of waste packages during the thermal period may 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 the probability of these other mechanisms is not high enough to warrant evaluating the consequences of these other processes. On the other hand, the effect of high-ionic strength input waters on in-package chemistry may not be large enough to have a significant effect on radionuclide release. DOE agreed (Reamer, 2001a) to update the in-package chemistry model to account for scenarios, their associated uncertainties, and implementation in the total system performance assessment model.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.4.5), is sufficient to expect that the information necessary to assess in-package chemistry with respect to system description and model integration will be available at the time of a potential license application.

5.1.3.4.4.2.2 Data and Model Justification

The in-package chemistry model represents the metal alloy waste package components as special reactants in the EQ6 input files. The amount of metal alloy that EQ6 adds to the reaction during a run is the product of the corrosion rate, the duration of the EQ6 timestep, and the surface area of the reactant. The surface areas used in the simulation remained fixed for the duration of the reactants' existence. Single values of corrosion rates are used and are stated to be supported by data for a range of temperatures and corroding water compositions provided in a reference. The reference with this support was not publicly available at the time of the NRC staff review, and DOE has agreed to provide it. In Bechtel SAIC Company, LLC

(2003b), insufficient technical justification was provided by DOE for the assumed corrosion rates of waste package components, and the likely modes of corrosion that account for the rates were not identified. DOE agreed (Schlueter, 2000) to address concerns regarding the effect of corrosion rates on in-package chemistry.

The dissolution rate equation for commercial spent nuclear fuel used in the in-package chemistry abstraction is the same equation recommended in CRWMS M&O (2000c,d). The DOE abstraction of commercial spent nuclear fuel degradation is reviewed in Section 5.1.3.4.4.4 of this report. It is stated DOE did not provide sufficient data to justify the abstracted model of spent nuclear fuel dissolution in the acid range of the model. Further, the abstracted model eliminated the term related to burnup of fuel, without considering results from high burnup fuels. For N-Reactor fuel, a constant reaction rate, based on a value five times the U-metal rate listed in DOE (2000), is used to describe the dissolution of the N-Reactor fuel. For high-level waste glass, a dissolution rate based on the transition state theory is used. DOE used conservative dissolution rates for commercial spent nuclear fuel, N-Reactor fuel, and high-level waste glass for calculations of radionuclide release from the wasteforms; however, variations in these rates may affect the calculated in-package chemistry. DOE agreed (Schlueter, 2000) to address concerns regarding the effect of corrosion rates on in-package chemistry.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.4.5), is sufficient to expect that the information necessary to assess in-package chemistry with respect to data being sufficient for model justification will be available at the time of a potential license application.

5.1.3.4.4.2.3 Data Uncertainty

For waste package components, such as Types 304L and 316 stainless steels, A516 carbon steel, and aluminum Alloy-1100, single values of corrosion rates are used (Bechtel SAIC Company, LLC, 2003b). These values are supported by data for a range of temperatures and corroding water compositions provided in a reference. DOE assessed the magnitude of the response of the in-package chemistry model to variability in metal alloy corrosion rates and determined the model was sensitive to a factor of five decrease in the metal alloy corrosion rates, which had the effect of delaying the pH response compared with the reference case (Bechtel SAIC Company, LLC, 2003b).

DOE evaluated the effect of variations in the waste package design configuration on the in-package chemistry by increasing the volume of the A516 carbon steel component. The results showed increasing the mass and surface area of A516 by a factor of approximately 10 had little influence on the pH profile.

Overall, the available information is sufficient to expect that the information necessary to assess in-package chemistry with respect to data uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.4.4.2.4 Model Uncertainty

In NRC (2001, 2000a), staff commented the DOE assumption that 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. Because the internal geometry of the 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. The revised in-package chemistry model presented in Bechtel SAIC Company, LLC (2003b) differs from the model used previously. The revised model is a film model in which the void space inside a failed waste package is partially occupied by liquid water in thermodynamic equilibrium with atmospheric gases both explicitly interacting in the solid-water-gas chemical system inside of a waste package. The film model uses a surface-area-based scaling technique, in contrast to the previous bathtub model that uses a volume-based scaling technique.

Overall, the available information is sufficient to expect that the information necessary to assess in-package chemistry with respect to model uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.4.4.2.5 Model Support

The following were considered by DOE to be salient essential features of the in-package chemistry model

- Production of alkaline waters (8 <pH <10) by interaction of dilute solutions with wasteform glass components

- Production of mildly acidic (3 <pH <5) waters by interaction of incoming solutions with internal components of the waste package (primarily A516 carbon, steel and Type 316 stainless steel)

- Production of high ionic strength solutions (>1 M) by reaction with wasteform compounds

DOE evaluated support for the in-package chemistry model by comparing the broad ranges of pH and ionic strengths derived from the model with values observed in natural systems (Bechtel SAIC Company, LLC, 2003b). The high-end pH values predicted by the in-package chemistry model were corroborated by natural observations documented in peer-reviewed literature, such as large-scale weathering of alkali-bearing silicates that lead to high pH values in alkali lakes and the high pH waters (up to 12) observed in deep ground waters in contact with dissolving ultramafic rocks isolated from atmospheric CO_2 gas. The acid production by long-term steel degradation was compared with alteration of pyrite, FeS_2, under oxidizing conditions, which generates pH values from 2 to 5, such as in areas with acidic mine drainage. The accumulation of dissolved salts during prolonged wasteform degradation was compared with alkali lakes that have high pH, such as Alkali Valley, Oregon, which has a pH of 10.1 and an ionic strength that exceeds 4 M.

Overall, the available information is sufficient to expect that the information necessary to assess in-package chemistry with respect to model abstraction output being supported by objective comparisons will be available at the time of a potential license application.

5.1.3.4.4.3 Degradation of Cladding on Commercial Spent Nuclear Fuel

5.1.3.4.4.3.1 Model Integration

Risk insights pertaining to radionuclide release rates and solubility limits indicate cladding degradation is of medium significance to waste isolation. Zircaloy cladding exhibits extremely low uniform corrosion rates in aqueous environments and could delay substantially the release of radionuclides from commercial spent nuclear fuel if it remains intact. Performance assessments show a high correlation between dose and fraction of failed cladding. Cladding failure can occur as a result of localized corrosion, stress corrosion cracking, and hydride reorientation and embrittlement, under a combination of adverse environmental and stress conditions. Cladding also may fail as a result of creep, caused by hoop stresses arising from internal fuel rod pressure, or by mechanical failure when subjected to loads associated with seismic events and rockfall.

DOE has considered that cladding can be an effective metallic barrier against the release of radionuclides from commercial spent nuclear fuel. Little experimental evidence has been provided, however, to support such an assessment nor have solid technical bases been developed for all the assumptions included in the model abstraction. This is the case, in particular, for the modeling of localized corrosion and stress corrosion cracking, as well as for the lack of consideration of hydride reorientation and embrittlement as a potential failure process that may be faster and, hence, more detrimental than unzipping alone.

Recently, DOE provided performance assessment calculations showing that 95th percentile cladding degradation, or even complete neutralization, increases the mean dose by one order of magnitude; and the dose is more than four orders of magnitude lower than that specified in the regulations for the reasonably maximally exposed individual (Bechtel SAIC Company, LLC, 2002). In the nominal case, however, it is assumed the fraction of failed cladding perforated before unzipping remains constant at 0.08, to approximately 50,000 years, and reached 0.2 only after 100,000 years (CRWMS M&O, 2000a). Note, however, these estimates of cladding protection do not consider the full range of possible failure mechanisms nor their probabilities and, therefore, may overestimate the effectiveness of cladding as a barrier.

In the TPA Version 4.1 code sensitivity report (Mohanty, et al., 2004; Appendix D, Figure 4.3.4-3), it is apparent that the introduction of cladding protection decreases the dose at 10,000 years with respect to that for the basecase from 2×10^{-4} to 3×10^{-6} mSv/yr [2×10^{-2} to 3×10^{-4} mrem/yr). Release rates of highly soluble and mobile radionuclides like Tc-99 and I-129 account for most of the 10,000-year predicted dose and are approximately proportional to the amount of spent nuclear fuel exposed. Other hazardous but less mobile radioelements like plutonium and americium may not be affected by the amount of spent nuclear fuel exposed, because the release is likely to be controlled by solubility limits.

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 wasteform 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. The degradation of the commercial spent nuclear fuel cladding was assumed to occur in two stages (CRWMS M&O, 2000b,e). The first stage of degradation corresponds to fuel rod failure as a result of cladding perforation by small cracks and holes.

The second stage involves progressive exposure of the spent nuclear fuel matrix as a result of axial 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 at the time of disposal and the percentage of rods perforated, taking into account data obtained from reactor operation, pool storage, dry storage, and transportation, including fuel handling (CRWMS M&O, 2000f). 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 1.15 percent of the total) was assumed initially perforated (CRWMS M&O, 2000f).

With the purpose of defining the creep damage of the Zircaloy cladding, which is considered the predominant potential failure mode prior to disposal, DOE used an empirical creep model developed by Matsuo (1987). 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, 2000f). DOE concluded little creep occurs for rod stresses less than 80 MPa [11.6 ksi]. It is assumed 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 during dry storage and transportation, compared with an actual failure rate of 0.045 percent (CRWMS M&O, 2000f).

Cladding perforation after waste package emplacement was assumed caused by creep, localized corrosion, stress corrosion cracking, and mechanical failure as a result of seismic events (CRWMS M&O, 2000e). To evaluate the possibility of creep and stress corrosion cracking for 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 the distribution of internal fuel rod pressure and corresponding hoop stress (CRWMS M&O, 2000e,f).

Unzipping of the cladding under dry conditions is excluded from the model abstraction, assuming the integrity of containers is maintained during the performance period (CRWMS M&O, 2000g). Only wet unzipping is assumed to occur. The time to unzip a fuel rod under wet conditions is estimated as a function of waste package temperature and 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 conservative, and included the consideration of uncertainties, DOE states that these criteria are less conservative than in previous total system performance assessments.

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, 2000g). 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 by hydride embrittlement would be unlikely, even if hydride reorientation did occur.

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, 2000f). 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 indicated 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. It was concluded delayed hydride cracking can be ruled out as a possible mechanism for cladding failure of spent nuclear fuel in the potential repository because the computed mean K_I value of 0.47 MPa•m$^{1/2}$ [0.43 ksi•in$^{1/2}$] was too low. As discussed in the next section (Section 5.1.3.4.4.3.2), the distribution of cladding stresses and temperatures and the evolution following waste package emplacement should also be considered. DOE has agreed (Schlueter, 2000) to address issues regarding hydrogen embrittlement as a mode of cladding degradation.

The Murty's creep-versus-strain correlation was selected to evaluate creep rupture after waste package emplacement on the basis of experimental data for unirradiated cladding (CRWMS M&O, 2000e). It is claimed 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 as a result of irradiation hardening. 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. In general, model integration for creep is adequate, however, specific details need further evaluation.

Localized corrosion also is considered a process leading to perforation of the commercial spent nuclear fuel cladding (CRWMS M&O, 2000e). 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 a bounding analysis because it is implicitly assumed 100-percent efficiency in the chemical reaction of fluoride with Zircaloy.

In the process model report (CRWMS M&O, 2000b), the role of fluoride is emphasized as a species promoting accelerated corrosion in local areas, however, insufficient technical basis is offered in CRWMS M&O (2000h). In addition, analyses of flow and volume of water contacting the fuel rods to evaluate the local attack by fluoride are limited and require additional justification.

Stress corrosion cracking also is considered a possible process leading to the perforation of cladding by cracks, 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, 2000e). This mechanism as such has been postulated as the cause of pellet-cladding interaction failure in reactors following steep power ramps, but seems unlikely for potential repository conditions. The possibility of stress corrosion cracking induced by iodine is discussed in the process model report (CRWMS M&O, 2000b). The iodine concentration is estimated to be above the 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.

The system description and model integration used in the abstraction of localized corrosion and stress corrosion cracking needs to consider the range of chemical conditions that may prevail in the in-package aqueous environment. As noted in the section on in-package chemistry (Section 5.1.3.4.4.2.1), compositions of the pore waters used in the process model and model abstraction (Bechtel SAIC Company, LLC, 2003b) do not bound the range of water chemistries that potentially can enter the breached waste packages, and, therefore, the evolution of the in-package water chemistry is not fully captured in the abstraction. Water with higher concentrations of anionic species could be present as a result of evaporation and concentration of seepage waters (Bechtel SAIC Company, LLC, 2003c). As noted in Section 5.1.3.4.4.2.1, DOE has agreed to update the in-package chemistry model to account for this scenario.

Although localized corrosion, in the form of pitting corrosion promoted by chloride anions, is a possible failure process (NRC, 2001), DOE excluded this detrimental effect of chloride (CRWMS M&O, 2000g) by assuming (i) chloride concentrations are 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, 2000e).

The chloride concentration inside breached waste packages, however, has not been properly bounded in the in-package chemistry abstraction (Bechtel SAIC Company, LLC, 2003b), and the presence of Fe^{3+} ions cannot be considered an absolute requirement because corrosion potentials higher than the pitting potential could 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, 2000h). In the discussion, the occurrence of pitting corrosion induced by chloride during repository conditions is questioned. 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, 2000g) 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. A low pH is assumed for the attack by fluoride, whereas this low pH is not taken into account when estimating the concentration of Fe^{3+} ions that may promote the oxidizing conditions required for pitting corrosion in chloride solutions. The lowest value of pH estimated for the inflow of pore water is nearly 3.0, however, a pH close to 1.0 is calculated in the case of water vapor condensation (Bechtel SAIC Company, LLC, 2003b). DOE agreed (Schlueter, 2000) to address concerns of the effects of in-package chemistry on localized corrosion of cladding.

Stress corrosion cracking of Zircaloy cladding may occur in the presence of hoop stresses of sufficient magnitude under 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, 2000e). The technical bases to support modeling of cladding degradation as a result of internal stress corrosion cracking by iodine are limited (NRC, 2001). DOE agreed (Schlueter, 2000) to address concerns of the effects of in-package chemistry on cladding degradation as a result of external stress corrosion cracking.

The remaining process that, according to DOE, 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}/yr. 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, 2000g) using the screening argument the waste package will remain intact for more than 10,000 years.

DOE has provided adequate information on the system description and model integration for creep and mechanical failure. For mechanical failure, however, the abstraction is related to the evaluation of seismic events (Section 5.1.2.2), and the exclusion of rockfall effects is related to the integrity of the waste package.

Potential processes of spent nuclear fuel cladding degradation that have significance to waste isolation are considered by DOE and incorporated in the model abstraction with the exception of localized corrosion and stress corrosion cracking. Additional information should be provided by DOE to dismiss the possibility of hydride reorientation and embrittlement, particularly for high burnup fuel, which has a significantly higher hydrogen content than average burnup fuel. The initial conditions of the cladding appear to be properly characterized by DOE including the distribution of initially perforated fuel rods. However, updated information is needed for high burnup fuel. There is insufficient technical basis, in terms of empirical observations or mechanistic understanding, supporting the two stages of cladding degradation used by DOE in the model abstraction. The screening of features, events and processes related to the degradation of cladding is adequately performed by DOE, with the exception of the lack of consideration of hydride reorientation and embrittlement. The full effects of in-package water chemistry need to be incorporated by DOE in the modeling of localized and stress corrosion cracking. DOE has stated that it intends to present a significantly different model abstraction for the cladding degradation, which will have more detail on the issues raised in this section (including cladding failure mechanisms, effects of water flow in the waste package, and high burnup fuel) in a technical basis document that was not available at the time of this review.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.4.5), is sufficient to expect that the information necessary to assess degradation of cladding on commercial spent nuclear fuel with respect to system description and model integration will be available at the time of a potential license application.

5.1.3.4.4.3.2 Data and Model Justification

Insufficient data have been presented to justify that accelerated corrosion by fluoride and 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 (Schlueter, 2000) to address concerns regarding the effects of in-package chemistry on localized corrosion and stress corrosion cracking of Zircaloy cladding.

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, 2000h) 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, the behavior of commercial purity zirconium (containing hafnium and lacking the Zircaloy alloying elements) is comparable to that of Zircaloy. Although a reasonable statement in general terms, no specific data are provided for aqueous 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 above 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, 2000h). The equation is not used in the model abstraction, however. In the analysis and model report (CRWMS M&O, 2000e), corrosion by fluoride to stoichiometrically form ZrF_4 is conservatively assumed to be determined by its concentration in 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.

As noted in the analysis and model report (CRWMS M&O, 2000e), 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 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 could be caused by other species present in the modified ground water, such as chloride. In addition, an adequate technical basis should be provided for selecting the critical stress relevant to the environment in which external stress corrosion cracking may occur. DOE agreed (Schlueter, 2000) to address concerns of the effects of in-package chemistry and stress on stress corrosion cracking of Zircaloy cladding.

In the assessment of hydride reorientation and delayed hydride cracking (CRWMS M&O, 2000i), 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. It is not clear that the proper cladding hoop stress, which mainly depends on the internal fuel rod pressure, was 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, 2000i). The hydrogen solubility in Zircaloy-2 and -4 at this temperature is approximately 90 ppm. Consequently, some 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. 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 the evolution following waste package emplacement in the repository. The consideration of cladding stresses is particularly important for high burnup fuel in which the hydrogen content could be significantly higher. DOE agreed (Schlueter, 2000) to address concerns regarding hydrogen embrittlement as a mode of cladding degradation.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.4.5), is sufficient to expect that the information necessary to assess degradation of cladding on commercial spent nuclear fuel with respect to data being sufficient for model justification will be available at the time of a potential license application.

5.1.3.4.4.3.3 Data Uncertainty

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 stresses and temperatures of the cladding are too low for hydride reorientation to occur and the cladding material would maintain sufficient strength even if hydride reorientation occurred, hence, cladding failure would be unlikely (CRWMS M&O, 2000b,g). The DOE arguments are not consistent, however, with the cladding temperatures and stresses documented in the analysis and model report (CRWMS M&O, 2000e). According to the 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, 2000g), 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, 2000i). 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 [8.0 and 17.4 ksi] (CRWMS M&O, 2000e). 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 evolutions 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 (Schlueter, 2000) to address concerns regarding cladding temperature and stress related to hydride embrittlement.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.4.5), is sufficient to expect that the information necessary to assess degradation of cladding on commercial spent nuclear fuel with respect to data uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.4.4.3.4 Model Uncertainty

The DOE model uncertainty characterization and use of alternative models are insufficient for certain aspects of commercial spent nuclear fuel cladding degradation. Alternative models or model uncertainties are not fully evaluated by DOE for certain aspects of localized corrosion and stress corrosion cracking of cladding.

The DOE abstraction considered most forms of degradation that may affect integrity of the 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).

After comparing the results of various alternative creep models to define the creep damage of zirconium cladding prior to 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, 2000f). After an evaluation of six creep models against five sets of experimental data, DOE elected Murty's creep model for disposal (CRWMS M&O, 2000e). DOE claimed 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, 2000e).] 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, 2000f). The Murty's model and the creep strain criteria both lead to conservative failure estimates.

In excluding hydride reorientation, DOE also argued the fracture strength of zirconium cladding with reoriented hydrides remains high. Concern is 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 zirconium-2.5 wt% niobium 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 decreased only 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 has not included hydride reorientation in its analyses of cladding failure or considered the possibility that hydride reorientation might lower the upper limit of the failure strain (11.7 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. 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, 2000e). 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, 2000f). The possible failure rate of these high burnup fuel rods has not been considered. DOE agreed (Schlueter, 2000) to address concerns of hydrogen embrittlement.

Finally, no alternative models have been included by DOE for localized corrosion and external stress corrosion cracking. DOE needs to demonstrate that 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, they 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 (Schlueter, 2000) to address concerns regarding the effects of in-package chemistry on localized corrosion and stress corrosion cracking of cladding.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.4.5), is sufficient to expect that the information necessary to assess degradation of cladding on commercial spent nuclear fuel with respect to model uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.4.4.3.5 Model Support

To date, adequate verification of the model abstraction for cladding degradation is not available. 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 to bound the rate at which other corrosion processes may perforate the cladding. The model abstraction of stress corrosion cracking, in which only iodine is considered the causative species for stress corrosion cracking, has not been verified for the conditions expected in the repository. DOE agreed (Schlueter, 2000) to provide a technical basis for the various modes of cladding degradation.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.4.5), is sufficient to expect that the information necessary to assess degradation of cladding on commercial spent nuclear fuel with respect to model abstraction output being supported by objective comparisons will be available at the time of a potential license application.

5.1.3.4.4.4 Commercial Spent Nuclear Fuel Dissolution

5.1.3.4.4.4.1 Model Integration

Risk insights pertaining to radionuclide release rates and solubility limits indicate wasteform degradation rate is of medium significance to waste isolation. The dissolution rate of the wasteform in an aqueous environment is important for all radionuclides. The dissolution rate uncertainty is considerable such that the time required to release radionuclides from the spent nuclear fuel matrix or vitrified wasteforms can vary from hundreds of years to hundreds of thousands of years. Water chemistry and temperature within the waste package could affect the degradation rate of the spent nuclear fuel. Corrosion of the internal metallic components of the waste package (e.g., fuel assembly baskets) could reduce the pH, leading to higher dissolution rates from spent nuclear fuel.

Several studies demonstrated the sensitivity of dose to the dissolution rate of the spent nuclear fuel source material. For example, the leading coefficient for the exponential dissolution Model 2 in the TPA code is one of the most influential parameters (Mohanty, et al., 2004; parameter PSFDM1). Among the four alternative models for spent nuclear fuel degradation in the TPA Version 4.1 code (each of which has a markedly different release rate), there is a clear correlation between release rate and dose (Appendix D, Figure 4.3.4-1). Dissolution rate is a relatively important determinant of repository performance, however, the ultimate peak dose is less than directly proportional to it because other mechanisms like diffusion, solubility limit, and sorption affect the ultimate release rates from the engineered barrier.

Fissions at grain boundaries, diffusion of fission products to the grain boundaries, and a thermal process (enhancement of local burnup caused by plutonium production and fissioning) at the rim are key contributors to the grain boundary radionuclide inventory during irradiation of nuclear fuel. Cracks in the fuel pellet (caused by the radial thermal gradient) and interconnected open porosity contribute to the pellet-cladding gap inventory. The fraction of soluble or volatile radionuclides such as cesium, iodine, chlorine, and carbon located in the gap and cesium, iodine, and segregated metallic phases such as technetium located at the grain boundaries (Poinssot, et al., 2001) is referred to as the instant release fraction. Studies have shown that, in the presence of water, fission products present at the grain boundaries are released at a slower rate than at the gap. Because of the difficulties in separating gap and grain boundary contributions, however, these fractions are combined and assumed to be released instantaneously on contact with water. DOE discussed the data for the instant release fraction abstraction, however, no abstraction was provided in the commercial spent nuclear fuel process model report (CRWMS M&O, 2000b).

Following the instant release fraction, the degradation of spent nuclear fuel depends on the aqueous chemical environment. The process is generally referred to as dissolution, although it typically involves oxidation of the spent nuclear fuel. The commercial spent nuclear fuel dissolution rates have been measured using a wide range of techniques and conditions, including flow-through experiments with spent nuclear fuel and unirradiated UO_2 pellets, static tests in autoclaves, and unsaturated drip tests with spent nuclear fuel pellets contained in zirconium cladding. 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,i).

The DOE abstraction of the matrix dissolution rate of the commercial spent nuclear fuel CRWMS M&O (2000b) is an empirical regression model loosely based on irreversible thermodynamic reasoning (Stout and Leider, 1998a,b). For pH>7, the model does not represent a significant portion of the variance in the experimental data.

The statistical significance of the abstraction for the acid environment is difficult to estimate because the abstraction is based on only two data points, one of which is a calculated value. The model was compared with the literature data on spent nuclear fuel in acidic conditions and was found to predict rates higher than derived from the experiments, thus justifying its use as a bounding model. The selection of data arbitrarily takes either the initial portion or the steady-state portion of the normalized release behavior as a function of time. The temperature range for the application of the model exceeded the temperature range of the tests. The fuel burnup also is not considered directly in the abstracted model, although a variety of burnups was used in the flow-through tests.

DOE has performed unsaturated drip tests during the past 10 years. These tests involved spent nuclear fuel contained in Zircaloy holders exposed to dripping water or a moist environment. The scaling relationship between the drip rate used in the unsaturated drip tests and the drip rate used in the in-package calculations is not clear—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,c). Furthermore, differentiation between contributions from the gap and grain boundary inventories and the contributions from the matrix dissolution is not transparent (Johnson, et al., 1985).

Drip test results are used to estimate effective surface area. An estimate of the surface area is important for calculating the amount of radionuclides released from the commercial spent nuclear fuel. The exposed surface area of a fuel pellet after burnup is a complex combination of fragmentation that increases the surface area and fusion of grains that decreases the surface area. The spent nuclear fuel dissolution rate estimated based on geometric area is much higher than the rate based on measured surface area. Alternatively, if the measured surface area determined by the Brunauer-Emmet-Teller method is used to estimate the surface area, the calculated spent nuclear fuel dissolution rate is nonconservative because the Brunauer-Emmet-Teller method tends to include porosity rather than accessibility to water and adsorbs multiple layers of gas. The surface roughness factor brings the estimated surface area somewhere in between the geometric and Brunauer-Emmet-Teller surface area measurements. It is unclear if the surface roughness factor needs to be considered in the presence of an alteration layer. Also, oxidation and hydration prior to dissolution could influence the estimate of the surface area.

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, based on extensive measurements of spent nuclear fuel and unirradiated UO_2 dissolution in flow-through tests (CRWMS M&O, 2000c). 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.

The in-package chemistry calculation is linked to the spent nuclear fuel dissolution model (Bechtel SAIC Company, LLC, 2003b). The role of radiolysis, quantity and chemistry of incoming water, localized corrosion, and transient effects are reviewed in the in-package chemistry abstraction. The in-package chemistry analysis and model report includes a sensitivity study on differing dissolution rates of components, 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 the in-package abstractions, and the applicability of abstractions for incoming water (Bechtel SAIC Company, LLC, 2003b).

Overall, the available information is sufficient to expect that the information necessary to assess commercial spent nuclear fuel dissolution with respect to system description and model integration will be available at the time of a potential license application.

5.1.3.4.4.4.2 Data and Model Justification

DOE has not provided sufficient data to justify the abstracted model of spent nuclear fuel dissolution in the acid range of the model. For example, the current abstracted model eliminated the term related to burnup of fuel that was used in the previous model and neglected the results from high burnup fuels (>50 GWd/MTU). 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 to be insignificant (Shoesmith, 1999) when compared with other factors. Tests continue on high burnup fuel, however, and the results may necessitate the need to revise 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 the model represents a bounding case. The reason DOE moved from a more complex model to a simpler model is not clear. Furthermore, the statistical significance of the abstraction of the acid range of the model is difficult to estimate because the significance 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 from only 8 to 10. Unirradiated UO_2 test data span the pH range from 3 to 11.6 (CRWMS M&O, 2000c), however, the acid test data are used for confirmation purposes only. DOE agreed (Reamer, 2001b) to address this concern regarding the applicable range of spent nuclear fuel dissolution model based on experimental data.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.4.5), is sufficient to expect that the information necessary to assess commercial spent nuclear fuel dissolution with respect to data being sufficient for model justification will be available at the time of a potential license application.

5.1.3.4.4.4.3 Data Uncertainty

DOE has not provided detailed information on propagation of the uncertainties in spent nuclear fuel dissolution rate data and the various parameters used in the calculation of in-package chemistry 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 (Bechtel SAIC Company, LLC, 2003b) suggests the in-package chemistry is likely to be near neutral or alkaline during a long 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 fuel rods and assemblies, or between basket material and fuel. Additionally, uncertainties exist regarding incoming chemistry and volume of water. Similarly, there are uncertainties in the dissolution rates of spent nuclear fuel, especially in the acid range of the model, where data are sparse. Finally, DOE has tested high burnup fuel; however, these data are not included in the model abstraction.

Although the uncertainties associated with data have not been evaluated, DOE bounded the abstraction model. Consequently, the characterization and propagation of data uncertainty are not necessary because DOE bounded commercial spent nuclear fuel rates using a conservative forward reaction rate.

Overall, the available information is sufficient to expect that the information necessary to assess commercial spent nuclear fuel dissolution with respect to data uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.4.4.4.4 Model Uncertainty

DOE relied primarily on flow-through test data to construct its abstracted model for commercial spent nuclear fuel dissolution rate (CRWMS M&O, 2000b,c). DOE made several assumptions to estimate the effective surface area and the spent nuclear fuel dissolution rate in the acid range of the model, however, the range of parametric uncertainties adopted by DOE appears to account for model uncertainties. DOE also has suggested the flow-through test results form an upper bound of dissolution rates measured by other techniques, although this test method is not standardized. DOE has not considered alternate models derived from the unsaturated drip tests, the immersion tests, or natural analogs. The electrochemical mechanism was used to justify the dissolution rate data derived from flow-through tests (Shoesmith, 1999). Although models for the drip test, the immersion test, and natural analog data may provide more realistic assessments of the spent nuclear fuel dissolution rate, use of the flow-through test to support commercial spent nuclear fuel dissolution is more conservative.

Overall, the available information is sufficient to expect that the information necessary to assess commercial spent nuclear fuel dissolution with respect to model uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.4.4.4.5 Model Support

The model abstraction used for the commercial spent nuclear fuel dissolution rate (CRWMS M&O, 2000j) is based on experimental measurements. Key assumptions that lack model support are

- Selection of parameters for the acid range of the commercial spent nuclear fuel abstraction

- Selection of the range of the effective surface area for estimating the amount of radionuclides released from the commercial spent nuclear fuel

Recent studies by Torrero, et al. (1997) and Rollin, et al. (2001) show the model abstraction proposed by DOE for the acidic range adequately represents the commercial spent nuclear fuel response despite assumptions used by DOE to assign parameter values. The DOE estimation of the effective surface area based on drip tests is not adequately supported by the study of Poinssot, et al. (2001), which showed the presence of 10–15 major fractures, in addition to a loosely held rim region in a pellet. Additionally, the wet fraction of test samples was not determined in the drip tests. It should be noted, however, 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.

Overall, the available information are sufficient to expect that the information necessary to assess commercial spent nuclear fuel dissolution with respect to model abstraction output being supported by objective comparisons will be available at the time of a potential license application.

5.1.3.4.4.5 DOE Spent Nuclear Fuel Dissolution

5.1.3.4.4.5.1 Model Integration

An assessment of the DOE approach to the abstraction of the DOE spent nuclear fuel dissolution is important because radionuclide release rates are dependent on the amount and rate of spent nuclear fuel dissolution. Although the DOE spent nuclear fuel represents a small fraction of the total inventory of fuel in the potential repository, assessment is still important to determine whether the total system performance assessment dose estimations will be affected by the release rates of this fraction of radionuclide inventory. Postclosure boundary dosage, however, is shown to be insensitive to fuel degradation when best estimate and conservative models for the total DOE spent nuclear fuel inventory are used in the total system performance assessment.

DOE spent nuclear fuel consists of more than 250 distinct spent nuclear fuel types divided into 12 groups. Immobilized ceramic plutonium waste also was considered (CRWMS M&O, 2000b). This wasteform will consist of disks of a plutonium-containing, titanium-dioxide-based ceramic enclosed in stainless steel cans. DOE evaluated the following 12 types of fuels and wasteforms:

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

Three types of degradation models for the DOE spent nuclear fuel and wasteforms were considered: 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. Best-estimate models are semiempirical and predict release rates based on available experimentation data. DOE has not committed to which model type will be used in the total system performance assessment (CRWMS M&O, 2000k). Presently, there are no directly relevant experimental dissolution/degradation data for many DOE spent nuclear fuel wasteforms. Only limited test data are available on some DOE spent nuclear fuel wasteforms. 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 wasteforms except Group 1. Models for the Group 1 fuel—Naval spent nuclear fuel—will be provided later by the U.S. Navy. More recent experimental data than those included in this report may be available, but currently are not accessible to the public.

DOE conducted total system performance assessment sensitivity analyses for degradation models for the 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, 2000a) model, DOE conservatively assumed the dissolution rate is a constant value equal to the rate for uranium-metal-based fuel (CRWMS M&O, 2000j). 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, 2000j).

Description of the characteristics, dissolution processes, and integration of the dissolution rates for the DOE spent nuclear fuel types is limited. Additional information regarding system description and model integration for the DOE spent nuclear fuel degradation is not needed, if DOE uses the upper-limit model, which predicts instantaneous release of radionuclides for every type of DOE spent nuclear fuel. Thus, impact of the DOE spent nuclear fuel on the performance of the repository would depend only on the total inventory of the radionuclides in the DOE spent nuclear fuel (CRWMS M&O, 2000b,j), and that inventory is adequately defined.

Overall, the available information is sufficient to expect that the information necessary to assess DOE spent nuclear fuel dissolution with respect to system description and model integration will be available at the time of a potential license application.

5.1.3.4.4.5.2 Data and Model Justification

DOE recommends that the models to be used in the total system performance assessment are the best estimate and conservative models for the N–reactor spent nuclear fuel (Group 7). This group is predicted to have a faster dissolution rate than most other groups and makes up the majority of the DOE spent nuclear fuel inventory. The Group 3 fuel, which has the fastest dissolution rate, makes up a small percent of the total DOE spent nuclear fuel inventory. Because the DOE spent nuclear fuel makes up only a few percent by weight of the total inventory for the potential repository, the use of the general model for Group 3 fuel degradation is unlikely to impact predicted performance of the repository and should not result in nonconservatism.

Data are limited regarding the characteristics of a large number of the DOE spent nuclear fuel types presented in CRWMS M&O (2000b). Experimental procedures and surrogates aiding development of the conservative and best-estimate models would not affect the upper-limit model results. If 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, additional data to support abstraction of the DOE spent nuclear fuel degradation are considered unnecessary.

Overall, the available information is sufficient to expect that the information necessary to assess DOE spent nuclear fuel dissolution with respect to data being sufficient for model justification will be available at the time of a potential license application.

5.1.3.4.4.5.3 Data Uncertainty

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. If the upper-limit model is used, interactions with water and fuel damage will not affect model results. Because the model assumes instantaneous release of radionuclides from the time of waste package breaching, no additional information is needed regarding the characterization and propagation of data uncertainty through abstraction of the DOE spent nuclear fuel dissolution, if the upper limit model is used.

Overall, the available information is sufficient to expect that the information necessary to assess DOE spent nuclear fuel dissolution with respect to data uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.4.4.5.4 Model Uncertainty

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. The model used is conservative relative to individual models for the fuel groups. The individual models adequately incorporate data from experiments to predict release rates by encompassing upper-limit data and scenarios. No additional information is needed regarding the characterization and propagation of model uncertainty through abstraction of the DOE spent nuclear fuel dissolution.

Overall, the available information is sufficient to expect that the information necessary to assess DOE spent nuclear fuel dissolution with respect to model uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.4.4.5.5 Model Support

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. The upper-limit model includes dissolution of the wasteform in a single timestep, thus quantity and chemistry of water are adequately encompassed. Postclosure boundary dosage is shown to be insensitive to fuel degradation when best-estimate and conservative models for the total DOE spent nuclear fuel inventory are used in the total system performance assessment. No additional information is needed regarding model support for abstraction of the DOE spent nuclear fuel dissolution.

Overall, the available information is sufficient to expect that the information necessary to assess DOE spent nuclear fuel dissolution with respect to model abstraction output being supported by objective comparisons will be available at the time of a potential license application.

5.1.3.4.4.6 High-Level Waste Glass Dissolution

5.1.3.4.4.6.1 Model Integration

Risk insights pertaining to radionuclide release rates and solubility limits indicate wasteform degradation rate is of medium significance to waste isolation. An assessment of the DOE approach to the abstraction of high-level waste glass dissolution is important because radionuclide contributions per waste package from the high-level waste glass are comparable to spent nuclear fuel at 75 °C [167 °F] (Jain and Pan, 2004). As the temperature drops below 50 °C [122 °F], however, the normalized dissolution rates for glass waste become less significant compared to spent nuclear fuel normalized dissolution rates. Furthermore, Pan, et al. (2003) showed that within the first 10,000 years, the dose deriving from waste glass dissolution is of the same order of magnitude as the nominal case dose. The dose rates at earlier times are a consequence of the assumption that initial defects could be present in waste containers. Results indicate Np-237, Tc-99, and I-129 are the predominant radionuclides contributing to the mean dose during the first 10,000-years because of the relatively high solubility in water and low sorption and retardation.

The basic form of the rate expression adopted by DOE (CRWMS M&O, 2000l) to describe the dissolution of waste glass immersed in water is given by a form of transition state rate law. Test results indicated that the dissolution rate dependence on pH and temperature was independent of the glass composition and within the range of the glass compositions tested, and, therefore, the same values were used for all waste glasses. The exposed glass surface area was estimated based on 20 times the surface area of the waste glass, and it was assumed the entire surface corrodes at the same rate when exposed to water. In addition, the DOE model recalculates the exposed surface area based on the mass of the remaining glass.

DOE stated (CRWMS M&O, 1998) that dissolution rates of waste 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 conducted limited analyses of waste glass degradation in the presence of corrosion products from the dissolution of waste package internal components, such as FeOOH, Fe_2O_3, Fe_3O_4, and $FeCl_3$, that could influence waste glass corrosion processes (Jeong and Ebert, 2003). Based on this study, DOE concluded the influence of iron corrosion products could be represented adequately by the pH term in the proposed abstraction model for waste glass.

Overall, the available information is sufficient to expect that the information necessary to assess high-level waste glass dissolution with respect to system description and model integration will be available at the time of a potential license application.

5.1.3.4.4.6.2 Data and Model Justification

The DOE abstraction for waste glass is based on forward reaction rate measurements on a five-component glass composition (Knauss, et al., 1990). The forward reaction rate parameters developed by Knauss, et al. (1990) were supported by dissolution studies of the Hanford and Savannah Rivers waste glass compositions, standard environmental assessments of glass, and analyses of waste glass data from the literature. The work of Advocat, et al. (1991), cited in the analysis and model report (CRWMS M&O, 2000l) 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, potentially on the nonconservative side. Even though such comparisons are inadequate for model justification, the selection of bounding parameter values for the glass wasteform corrosion alleviates any influence of data by Advocat, et al. (1991). The parameter values selected by DOE also bound the existing literature data on long-term corrosion behavior, referred to as Stage III corrosion. Although the coefficients for pH and activation energy (E_a) are assumed independent of glass composition, the pH-dependent coefficient (η) and E_a values bound the variability expected from glass compositions.

Overall, the available information is sufficient to expect that the information necessary to assess high-level waste glass dissolution with respect to data being sufficient for model justification will be available at the time of a potential license application.

5.1.3.4.4.6.3 Data Uncertainty

Data used in the DOE abstraction for waste glass dissolution are based on experiments conducted by Knauss, et al. (1990). This study defines the glass dissolution dependence on pH and temperature for a five-component 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 data have not been evaluated by DOE using anticipated glass compositions. The DOE model lacks evaluation of data uncertainties. The characterization and propagation of data uncertainty through abstraction of the high-level waste glass dissolution, however, are not necessary because DOE bounded high-level waste glass dissolution rates using a conservative forward reaction rate.

Overall, the available information is sufficient to expect that the information necessary to assess high-level waste glass dissolution with respect to data uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.4.4.6.4 Model Uncertainty

The DOE model for high-level waste glass dissolution is based on a single set of experiments conducted by Knauss, et al. (1990). This study defines the glass dissolution dependence on pH and temperature for a five-component glass composition. In the alkaline range, DOE bounded the forward reaction-rate term in the model by performing several sets of experiments using various waste glass compositions. The observed mean and mean plus two standard deviations were used to define lower and upper bounds, respectively, for the forward reaction rate. The average of the lower and upper bounds was then used to define the mean forward reaction rate. Variability in the coefficients for temperature and pH was based on the linear regression analysis of Knauss, et al. data. The observed mean value plus one standard deviation for activation energy was used as the upper bound, and the mean value minus two standard deviations was used as a lower bound. Smaller activation energy term provides a more conservative assessment. The mean and standard deviation for η determined from the linear regression of Knauss, et al. data were used directly as input. The variability range for various terms used in the alkaline range of the model is shown in Eq. (5.3.1.4-7). In the acid range of the model, the mean and standard deviation observed from the linear regression of the Knauss, et al. data were directly used in the model represented by Eq. (5.3.1.4-6). The characterization and propagation of model uncertainty through the abstraction of the high-level waste glass dissolution are not necessary because DOE bounded high-level waste glass dissolution rates using a conservative forward reaction rate.

Overall, the available information is sufficient to expect that the information necessary to assess high-level waste glass dissolution with respect to model uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.4.4.6.5 Model Support

The forward reaction-rate model proposed by DOE, based on data by Knauss, et al. (1990) bounds the dissolution studies of the Hanford and Savannah Rivers waste glass compositions, standard environmental assessment of glass, and analyses of waste glass data from the literature.

Overall, the available information is sufficient to expect that the information necessary to assess high-level waste glass dissolution with respect to model abstraction output being supported by objective comparisons will be available at the time of a potential license application.

5.1.3.4.4.7 Radionuclide Solubility

5.1.3.4.4.7.1 Model Integration

Risk insights pertaining to radionuclide release rates and solubility limits indicate radionuclide solubility limits are of medium significance to waste isolation. Solubility limits can be important factors in the release of radionuclides from the engineered barrier and, ultimately, to dose. For example, the DOE analyses indicate the calculated dose is sensitive to neptunium solubility (Bechtel SAIC Company, LLC, 2002). In a recent numerical study, of the three radionuclides estimated to be major contributors to dose (i.e., Tc-99, I-129, and Np-237), only the release of Np-237 would be decreased significantly by its solubility limit (Mohanty, et al., 2003). However, 13 of 20 radionuclides considered in this study exhibited solubility-limited behavior. Mohanty, et al. showed radioelements such as americium and plutonium exhibit increased release in performance assessment models when solubility limits are artificially increased.

The solubility value for a particular radioelement used in performance assessment calculations will depend on the properties of the solubility-controlling solid phase for the radioelement of concern. Possible solubility-controlling solid phases may lead to radionuclide solubilities that differ by several orders of magnitude. For some radioelements, including neptunium, incorporation of radionuclide components is possible in secondary phases composed predominantly of other components, however, the evidence for this mechanism is limited (Fortner, et al., 2003).

Necessary information in solubility values pertains to the physical–chemical conditions of the system. The solubility of a radioelement will depend on composition of the aqueous phase and on its temperature and oxidation state. Inorganic and organic ligands that can form aqueous complexes with the radioelements may be present. Complexation increases the solubility-limited amount of the radioelement in the solution for elements such as uranium, neptunium, plutonium, and americium. Actinide solution chemistry in environmental waters is dominated by hydroxide and carbonate complexation; thus, the solubility of actinide solids would be highly dependent on pH, aqueous carbonate concentration, and partial pressure of carbon dioxide gas. The solubilities of some radioelements depend strongly on their oxidation states. Because of uncertainties in these variables in the waste package and near-field environment, and uncertainties in the properties of the solubility-limiting solid phases and aqueous radionuclide species, there is a wide range of possible radionuclide concentration limits.

In addition to the value for the solubility limit, the degree to which radionuclide release to the environment is controlled by solubility depends on relations among the waste-form leaching rate, the degree of waste-form exposure to water (flow rate), the radionuclide half-life, the radionuclide inventory, and the position of the radionuclide in the decay chain.

Risk insights pertaining to radionuclide solubility limits indicate radionuclide solubility modeling depends on integration with models for waste package chemistry and near-field chemistry, which, in turn, depend on models for unsaturated zone geochemistry. The significance of radionuclide solubilities in repository performance modeling also depends on wasteform dissolution models. For rapid release by dissolution or prompt release, which may be conservative approaches, solubility limits acquire greater importance because they are more commonly achieved. For sufficiently low release rates, performance is unaffected by solubility limits.

The DOE approach to calculate aqueous concentrations of radionuclides in water that reacts with the wasteform is initially to derive concentrations from the wasteform dissolution model, which is based on wasteform dissolution rates. Subsequently, comparisons are made between these potential dissolution-based aqueous concentrations of the radionuclides and values for the solubility limits. In most cases, applied solubility limits are determined using analytical relations based on thermodynamic modeling and data, which express solubility limits as functions of key independent parameters such as pH and CO_2 pressure. If the solubility-limited value is lower for a given radionuclide than its concentration derived from wasteform dissolution, the aqueous concentration is set to the solubility-limited value, and the difference in mass is modeled 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 in proportion to its isotopic abundance (CRWMS M&O, 1998).

For the total system performance assessment–site recommendation (CRWMS M&O, 2000a), 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).

A revised document on dissolved concentration limits of radioactive elements has been published by the DOE (Bechtel SAIC Company, LLC, 2003a). The NRC staff evaluation is ongoing of the DOE radionuclide solubility limit abstraction with respect to system description and model integration.

Overall, the available information is sufficient to expect that the information necessary to assess radionuclide solubility limits with respect to system description and model integration will be available at the time of a potential license application.

5.1.3.4.4.7.2 Data and Model Justification

Application of solubilities in performance assessment models for concentration limits have a theoretical justification in chemical thermodynamics. In some cases, nonconservatisms caused by possible metastable supersaturations can be addressed justifiably by selecting relatively unstable and readily precipitated solubility-limiting solids such as amorphous hydrous phases.

Models for trace radionuclide releases limited by precipitation in secondary solid solutions (e.g., coprecipitation) and subsequent release according to the stability and reaction rates of the host solid phase could improve substantially estimates of performance for Yucca Mountain (Murphy and Codell, 1999). These models, however, are unjustified as a basis for solubility limits for radionuclides in performance assessment models. Equilibrium between a minor or trace component of a solid solution and the bulk solution is unlikely to be maintained because the rate of equilibration would be restricted by slow homogenization of the solid phase. For these conditions, the thermodynamic basis for solubility limits does not pertain to the minor radionuclide component in the host phase. Furthermore, use of equilibrium between a trace or minor component of a solid solution and the aqueous phase does not place an upper limit on the aqueous concentration without fixing precisely the concentration of that component in the solid phase and the activity-composition relations for the solid solution. Defining these properties of the solid solution phase are difficult problems and unlikely to be well constrained for the repository system.

In a prior NRC technical position on solubilities (NRC, 1984), data justifications for solubility limits were required to be based on reversed equilibrium solubility experiments for the phase composed of the radionuclide component. Such precise data are unavailable for most solid phases with radionuclides of significance to repository performance.

For solubilities based on analytical relations abstracted from thermodynamic modeling, the effective probability distribution functions for solubility limits depend on distributions of the controlling independent variables such as pH and CO_2 pressure. These variables are based on models for waste package and near-field chemistry and affect many aspects of repository performance including corrosion, wasteform dissolution, and radionuclide transport in addition to solubility.

DOE provided a description how the experimental data and EQ3 modeling results were used, interpreted, and synthesized into the abstraction of radionuclide concentration limits. For radionuclides with high solubility limits, an arbitrary large number is assigned to their solubilities such that release will be controlled by the waste inventory and the wasteform degradation rate.

The DOE approach to concentration limits is based on postulated ranges of conditions and thermodynamic modeling (Bechtel SAIC Company, LLC, 2003a). Functional relations are developed between calculated solubilities and the intensive parameters, temperature, pH, and log CO_2 pressure. Then, additional uncertainty functions are supplemented. Concentration limits used in the DOE probabilistic performance assessment will be derivative properties. Parameter sampling may occur for pH, temperature, and CO_2 pressure. Then, based on the functional relations derived from thermodynamic modeling, values of solubilities will be calculated. This approach provides an alternate evaluation of distributions of solubility limits. It deviates from the established approach of characterizing concentration limit distributions and sampling from those distributions. This derivative approach of DOE is based on suites of thermodynamic models and abstractions of their results in functional relations specific to the Yucca Mountain performance assessment problem.

The NRC staff evaluation is ongoing of the DOE radionuclide solubility-limited abstraction with respect to data being sufficient for model justification. However, overall, it appears the available information is sufficient to conclude that the information necessary to assess radionuclide solubility limits with respect to data being sufficient for model justification will be available at the time of a potential license application.

5.1.3.4.4.7.3 Data Uncertainty

Potentially important uncertainties in thermodynamic data include the relative stability of aqueous neptunium in valence states of +3 and +5, and the possibility of incompletely characterized polynuclear species or aqueous complexes with multiple ligands. Data for the properties of solubility controlling solid phases are far more uncertain.

DOE model solubilities were based on thermodynamic properties for solids and aqueous species and extended Debye-Huckel activity coefficient (b-dot) relations given in the EQ3NR database (data0.ymp.R2). This database depends strongly on recent Nuclear Energy Agency compilations of thermodynamic properties for actinides. These compilations tend to impose scientific conservatism by omitting species for which data are uncertain. Omitting species, however, is not conservative for performance assessment applications because omissions can lead to calculated total concentrations smaller than if the omitted species were included.

The DOE solubility studies considered extremely broad ranges of pH and CO_2 pressure. The range in pH (3–11) was intended for the range of in-package conditions for codisposal materials. The range of CO_2 pressure (10^{-5}–$10^{-1.5}$ bar) was selected with little apparent basis, although it encompasses any reasonably expected values. In many cases, these broad conditions, coupled with the sulfuric acid and sodium hydroxide treatments to achieve desired pH, led to aqueous solutions or solid-solution equilibria that are thermodynamically unreasonable. A common feature of the concentration limits tabulated in Bechtel SAIC Company, LLC (2003a) is arbitrarily high values adopted for conditions and systems that failed to permit a valid thermodynamic model to be computed.

In addition to ranges of calculated solubilities for variable solution compositions and solid phases, variability terms were included for some radioactive elements to accommodate uncertainties in the thermodynamic data or in the potential aqueous complexation effects of fluoride. Treatment of fluoride variations is conservative, particularly in the absence of likely mechanisms for generating solutions of high fluoride concentration.

The NRC staff evaluation is ongoing of the DOE radionuclide solubility limit abstraction with respect to data uncertainty being characterized and propagated through model abstraction. However, overall, it appears the available information is sufficient to expect that the information necessary to assess radionuclide solubility limits with respect to data uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.4.4.7.4 Model Uncertainty

The current basis for concentration limits for the DOE total system performance assessment is described in the analysis and model report (Bechtel SAIC Company, LLC, 2003a). The general DOE approach to constrain concentration limits is to postulate a range of water compositions and to calculate (model) solubilities for radioactive elements at equilibrium with these waters. Water chemistries were based on J–13 Well water with large variations in pH and CO_2 partial pressure. These variations required corresponding changes in water chemistry to maintain thermodynamic correctness. For example, sulfuric acid and sodium hydroxide were added to generate acid and base solutions, invoking sources of sulfur from metal and glass alterations, respectively. Solubility calculations were performed for 25 °C [77 °F] and oxidizing conditions.

Models for the aqueous chemistry that controls solubilities are uncertain, and practical conservatisms in values of chemical variables are difficult to apply because of the complex nonlinear relations between solution compositions and concentration limits based on thermodynamic solubilities. Conservatisms in modeled solution compositions with respect to a given radionuclide may be nonconservative with respect to other radionuclides or nonconservative with respect to other processes that affect performance, such as corrosion rates. Broad ranges of conditions encompassing possible conditions do not ensure conservatism because the ranges may extend to conditions for which solubilities are low. Probability distribution functions for solubility limits have little or no scientific basis (Murphy, et al., 2004). Log-uniform distributions of solubilities, which have been invoked in some cases, strongly emphasize low solubility values relative to uniform distributions, which is a nonconservative approach. The current DOE practice for performance assessment is to calculate solubilities based on relations, derived in separate suites of equilibrium calculations, between key solution composition variables (e.g., pH and CO_2 pressure) and thermodynamic solubilities for selected radionuclide solids. The DOE approach to establishing ranges and distributions of solubilities that eventually control performance assessment results is not transparent.

Substantial uncertainties exist in models regarding transient waste package chemical conditions, leading to broad ranges and poorly defined distributions of chemical variables that control solubility limits. Solubilities, however, will affect releases at long times in the future when chemical conditions are likely to be buffered by ambient geochemistry and stable alteration products of the waste package. Transient conditions in the waste package that could lead to high solubilities are unlikely to extend far along transport pathways, so these conditions are unlikely to affect radionuclide transport at distances where performance measures are applied. Transient waste package chemical conditions that lead to lower solubilities would be beneficial to waste isolation, and their neglect in performance modeling would be conservative.

Generally, correlations have been neglected among concentration limits for different radioactive elements used in performance assessment. The thermodynamically based approach taken by DOE and the derivative approach of selecting concentration limits by sampling pH and CO_2 pressure should provide a useful set of data to evaluate correlations between concentration limits.

The NRC staff evaluation is ongoing of the DOE radionuclide solubility limit abstraction with respect to model uncertainty being characterized and propagated through model abstraction. However, overall, it appears the available information is sufficient to expect that the information necessary to assess radionuclide solubility limits with respect to model uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.4.4.7.5 Model Support

DOE does not provide a general treatment of model support for solubilities. Each radionuclide poses a separate set of problems, and varying degrees of support are provided for individual elements. In some cases, no support is provided except to note the radionuclide may bear chemical similarities to another relatively better understood radionuclide.

Comparing neptunium concentrations in spent nuclear fuel experiments to hypothetical solubility calculations compares different things. Nevertheless, Bechtel SAIC Company, LLC (2003a) states "This comparison shows that the Np_2O_5 solubility model developed in this report is conservative and thus is adequate for TSPA use." Concentrations from experiments not designed as solubility studies generally do not provide limiting concentrations, so comparisons to these values are not validation of solubility limit estimations.

The NRC evaluation is ongoing of the DOE radionuclide solubility limit abstraction with respect to model abstraction output being supported by objective comparisons. However, overall, it appears the available information is sufficient to expect that the information necessary to assess radionuclide solubility limits with respect to model abstraction being supported by objective comparisons will be available at the time of a potential license application.

5.1.3.4.4.8 Colloidal Release

5.1.3.4.4.8.1 Model Integration

Colloids can enhance radionuclide release if they form and remain stable in the source area and if radionuclides are effectively attached to them. The DOE abstraction of colloidal radionuclide release from wasteforms and transport within the drift is addressed primarily in a technical basis document (Bechtel SAIC Company, LLC, 2003d) and an analysis and model report (Bechtel SAIC Company, LLC, 2003e). This colloid release abstraction defines colloid-associated concentrations of certain radionuclides in water as these leave the waste package and again as they exit the invert. For high-level waste glass, the abstraction allows reversible and irreversible radionuclide attachment to colloids. For spent nuclear fuel wasteforms, 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 and in-drift ionic strength and pH. 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 wasteform (clay colloids from high-level waste glass), ground water (preexisting), or iron oxyhydroxide (from corrosion) colloids. True colloids (i.e., products of radionuclide precipitation) are not included. A new feature of the abstraction (Bechtel SAIC Company, LLC, 2003d,e) is the retention of a large proportion of released plutonium and americium on iron corrosion products in the waste package, resulting in reduced release relative to the earlier abstraction (CRWMS M&O, 2000m). The colloidal mass released from the waste package is then passed to the invert, adjusted for in-drift chemical conditions, and passed to the unsaturated zone for transport calculations. Complete evaluation of the abstraction of colloidal transport within the engineered barrier system outside the waste package is dependent on a report that is not yet available to the public.

Overall, the available information is sufficient to expect that the information necessary to assess colloid release with respect to model integration will be available at the time of a potential license application.

5.1.3.4.4.8.2 Data and Model Justification

The majority of wastes proposed to be disposed at the potential repository at Yucca Mountain are commercial spent nuclear fuel rods. Therefore, the DOE exclusion of wasteform colloids with irreversibly attached radionuclides derived from spent nuclear fuel must have a strong technical basis. A detailed review will be needed about the recent spent nuclear fuel corrosion test results—not yet publicly available—that form the basis for excluding wasteform colloid generation from spent nuclear fuel. For example, DOE refers to published corrosion tests on unirradiated UO_2 (Wronkiewicz, et al., 1997), which included the "formation of a dense mat of alteration products" that may have "reduced particulate release by trapping particulates in the altered products" (Bechtel SAIC Company, LLC, 2003d, pp. 43–44; 2003f, pp. 3–9). It is not clear from the DOE reports (Bechtel SAIC Company, LLC, 2003c,d) if this process could have artificially masked colloid production during the recent spent nuclear fuel tests. That is, the information presented in the report does not demonstrate this process could inhibit colloid formation in a repository setting. In addition, recent results from spent nuclear fuel corrosion tests at Pacific Northwest National Laboratory indicate formation of a plutonium-enriched surface alteration layer that may serve as a source for colloidal mobilization (Buck, et al., 2004a). These results and observations of colloids in UO_2 tests helped form the basis for an alternative conceptual model for spent nuclear fuel colloid generation (Buck, et al., 2004b). The latter report concluded conditions necessary for colloid mobilization are not expected for typical repository conditions, however, the model has not been quantified. An argument for exclusion based on physical and chemical improbability of mobilization (Buck, et al., 2004b) is not consistent with inclusion of the glass wasteform colloidal release.

Bechtel SAIC Company, LLC (2003e, Section 6.6.1), in describing basecase model results, refers to output quantities of "dissolved wasteform radionuclides derived from commercial and DOE SNF [spent nuclear fuel] wasteform colloids." It is not clear if and how these radionuclides are included in the release abstraction, if colloid release from these wasteforms is neglected.

The basis for selecting radionuclides for inclusion in the reversible and irreversible colloid release abstractions is included in Bechtel SAIC Company, LLC (2003e). Plutonium and americium are the only radionuclides included in the irreversible model; no explicit basis was provided for excluding other radionuclides. The appropriateness of this DOE choice will be judged on how effectively, in the total system performance assessment, the colloidal species are transported relative to more mobile dissolved elements such as neptunium and uranium. For the reversible model, plutonium, americium, thorium, cesium, and protactinium were chosen, while neptunium, uranium, and strontium were not. Neptunium and uranium were judged relatively insensitive to colloid enhancement, while strontium was eliminated because of the short half-life of Sr-90. Independent calculations (Contardi, et al., 2001; Pickett and Dam, 2003) confirm the effect of reversible attachment is small on retardation of uranium and neptunium.

A significant change in the DOE abstraction for the release of plutonium and americium is the addition of retention of these radionuclides on stationary iron corrosion products within the waste package, which has the effect of greatly reducing the masses available for transport to ground water (Bechtel SAIC Company, LLC, 2003d, Section 3.4.3; 2003c, pp. 73–77). The model "is implemented such that a large fraction of total Pu is sorbed to [stationary] corrosion products, a small fraction to colloids, and a small fraction remains dissolved in the fluid"

(Bechtel SAIC Company, LLC, 2003e, page 75). Clearly, this new abstraction, while more realistic, will result in significantly lower plutonium and americium release from the engineered barrier system compared with past abstractions that took no credit for retention. The model is developed in an unreleased analysis and model report and, therefore, cannot be evaluated at this time. The descriptions and bases for the model in available reports are qualitative and do not provide detailed information on the algorithms and parameters employed. DOE has stated that it will provide the analysis and model report with the details of this model.

Overall, the available information is sufficient to expect that the information necessary to assess colloid release with respect to data being sufficient for model justification will be available at the time of a potential license application.

5.1.3.4.4.8.3 Data Uncertainty

The magnitude of release of plutonium and americium irreversibly bound to colloids depends on the calculated wasteform colloidal plutonium concentration, the concentration of ground water and iron oxyhydroxide colloids, and the distribution coefficients governing radionuclide colloid attachment. Available reports (Bechtel SAIC Company, LLC, 2003d,e; CRWMS M&O, 2000m,n) document the laboratory and field data supporting parameter distributions for ground water and glass wasteform colloids. Recent DOE reports (Bechtel SAIC Company, LLC, 2003d,e) do not provide the University of Nevada, Las Vegas, measurements of iron corrosion product colloids that support the adopted distributions. Although it appears the new iron oxyhydroxide colloid maximum mass concentration distribution is conservative (e.g., the maximum changed from 1 to 50 ppm), the data would help corroborate the large colloid instability field in pH/ionic strength space in the plot governing the colloid concentration algorithm (Bechtel SAIC Company, LLC, 2003e, Figure 7). The position of calculated in-package water chemistry on this plot will determine if iron oxyhydroxide colloid concentration is set to a minimum value of 0.001 ppm or a maximum value sampled from a uniform distribution from 0.05 to 50 ppm. The figure is based on only one published paper (Liang and Morgan, 1990), which had no data in the important pH range 7–10, and the DOE reports (Bechtel SAIC Company, LLC, 2003d,e) do not discuss whether or not the algorithm is corroborated by the new laboratory results or the more recent literature. Two literature references cited in Bechtel SAIC Company, LLC (2003e, p. 48) show iron oxyhydroxide colloid concentrations in ground water of up to 261 and 0.04 ppm (Vilks, et al., 1993; Laaksoharju, et al., 1995). Ledin, et al. (1994) maintained stability of synthetic iron oxyhydroxide colloid suspensions across the pH range 8–9, and measured pH of zero point of charge values of 9.1–10.5. This information suggests such colloids may be stable between pH 8 and 9 for some circumstances. {It should be mentioned that the $Fe(OH)_3$ colloids in the Ledin, et al. (1994) study did aggregate to particle sizes greater than 1,000 nm [4×10^{-5} in]; however, FeOOH colloids maintained smaller sizes.}

In summary, the case for iron oxyhydroxide colloid concentration at 0.001 ppm throughout the instability range of Figure 7, Bechtel SAIC Company, LLC (2003e), at ionic strengths below 0.05 needs to be bolstered with explicit reference to conditions in the University of Nevada, Las Vegas, experiments and literature references.

Transport by diffusion only in the invert provides an effective barrier to radionuclide release from the engineered barrier system. Because colloid diffusion coefficients are modeled 100 times higher than dissolved species, this barrier is especially effective for colloids with irreversible attachment. DOE presented new data supporting the diffusion coefficient for dissolved species in crushed tuff invert materials as a function of volumetric moisture content (Bechtel SAIC Company, LLC, 2003d, Appendix F). Uncertainty in the diffusion data results from uncertainty in the measured diffusion coefficient for dissolved species, in crushed tuff, variability in colloid radius, and uncertainty in the invert volumetric water content. Full evaluation of the adequacy of model diffusion coefficients in capturing these uncertainties depends on review of a DOE document that has not yet been released to the public. DOE has stated that it will provide the document with the details of this model.

Overall, the available information is sufficient to expect that the information necessary to assess colloid release with respect to data uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.4.4.8.4 Model Uncertainty

As discussed in the original report (NRC, 2002, Section 3.3.4.4.8), the DOE performance assessment calculations demonstrate the high degree of sensitivity of calculated colloid concentrations to modeled in-package and in-drift chemistry (CRWMS M&O, 2000a). This concentration sensitivity to uncertain ionic strength and pH calculations is amplified now by the adoption of an irreversible attachment to iron oxyhydroxide colloids. Bechtel SAIC Company, LLC (2003d) argues that models of pH and ionic strength variations in the waste package show the probability is low of achieving a stable colloidal suspension. In addition, elevated temperatures during the first 1,000 years will contribute to instability. Therefore, DOE considers that sensitivity to modeled chemical conditions is not significant.

The information about predicted pH and ionic strength in the waste package as a function of time in Bechtel SAIC Company, LLC (2003d, Table H–1, pp. 3-1 through 3-4) is provided without supporting data. Implications of the uncertainty in in-package ionic strength for colloid stability cannot be evaluated without a description of the uncertainty in the predicted ionic strength or of the range of predicted chemical environments for different wasteforms, packages, and hydrologic conditions. DOE should provide supporting information such as plots of calculated pH and ionic strength relative to colloid stability fields for the broad range of scenarios. This information also should address in-drift chemistry and temperature, which DOE claims will suppress colloid concentrations (Bechtel SAIC Company, LLC, 2003d, Appendix H).

As discussed in Section 5.1.3.4.4.8.2, the DOE abstraction now relies on a great deal of retention of plutonium and americium on stationary iron corrosion products. The model is not described in sufficient detail in available documents to allow evaluation of how uncertainty is handled.

Alternative conceptual models have been proposed that incorporate colloid release from spent nuclear fuel wasteforms (Bechtel SAIC Company, LLC, 2003e, Sections 6.4.1, 6.4.3; Buck, et al., 2004b). These models are inherently less conservative than the basecase, such that their exclusion from performance assessment must be justified. The justifications in Bechtel SAIC Company, LLC (2003e, Section 6.8) are that necessary supporting data are unavailable or necessary conditions are unlikely. These models have not yet been fully developed, however, and arguments for the neglect of spent nuclear fuel colloids are not yet complete in available documents (Section 5.1.3.4.4.8.2). Moreover, the uncertainty this conceptual model contributes to assessing the potential impact on radionuclide release has not been quantified. DOE has stated that it will provide the documents with the details of this model.

Overall, the available information is sufficient to expect that the information necessary to assess colloid release with respect to model uncertainty being characterized and propagated through the model abstraction will be available at the time of a potential license application.

5.1.3.4.4.8.5 Model Support

NRC (2003, Section 2.2.1.3.4.2) calls for evaluation of model outputs in applying the model support review method. Available reports (Bechtel SAIC Company, LLC, 2003d,e) do not provide such outputs that will support a potential license application; therefore, model support evaluation is limited at this time. DOE has stated that it will provide the documents with the details on this model.

Modeling colloid processes is highly uncertain, thus, model results must be shown to reflect conservative assumptions that will ensure DOE does not underestimate the effects of colloids on radionuclide release. Model support is addressed directly in Section 7, Validation, of Bechtel SAIC Company, LLC (2003e), with corroborative information references listed in a table. Model validation arguments in that section focus on the use of sound scientific principles in light of the significant uncertainties on how colloids would behave in a repository system. Corroborative data include natural analogs studies supporting limited colloid-associated uranium mobility in the vicinity of mines, as well as nonqualified data supporting natural ground water colloid concentrations. Reference also is made to studies of plutonium colloid association unrelated to the Yucca Mountain studies. No site-specific or field-scale information was used by DOE to independently corroborate the abstraction. Furthermore, abstraction outputs are not yet available for comparison with any of the corroborative information. These types of model support arguments are expected to be provided when model outputs are reported at the time of a potential license application.

The original version of this report (NRC, 2002) discussed an error in the DOE algorithm for calculating iron oxyhydroxide colloid concentration that would result in consistently predicting the minimum value for all values of ionic strength and pH. This error has apparently been corrected, as indicated in Figure 18e of Bechtel SAIC Company, LLC (2003e); however, the text logic statement below the figure retains the error.

Overall, the available information is sufficient to expect that the information necessary to assess colloid release with respect to model abstraction output being supported by objective comparisons will be available at the time of a potential license application.

5.1.3.4.4.9 Engineered Barrier System Flow and Transport

5.1.3.4.4.9.1 Model Integration

The significance of diffusional release will depend on numerous assumptions: water-film thicknesses, diffusion distances inside and outside the waste packages, unclogged openings, and a mechanism to sweep away contaminants, to keep the concentration gradients high. Advective releases rely on the quantity of water entering and leaving the waste package, and could be more significant than diffusion after degradation processes cause openings in the waste package sufficiently large to allow dripping water to come into direct contact with the waste. Processes leading to openings in the waste packages include localized corrosion, stress corrosion cracking coupled with mechanical loading events from dynamic and static rock-fall loads, and intrusive igneous activity disruptive events.

Scenarios capable of allowing diffusion in amounts that would be sufficient to cause significant releases are highly unlikely. There would have to be a continuous and substantial water pathway for diffusion and a mechanism to keep the concentration gradient high. Water films are likely to be thin or discontinuous. For conditions where only water vapor is present in the drift, water films inside the waste package would be limited to layer thicknesses measured in tens of molecules or less. Mechanisms for flushing diffused radionuclides away from the waste package (i.e., to keep the concentration gradient high) are unlikely, requiring liquid water to drip onto cracks or holes in the waste package. Calculations of diffusion under any likely condition show that releases by this mechanism are unlikely to cause doses of any significance, even with the failure or underperformance of other barriers.

Risk insights pertaining to radionuclide release rates and solubility limits indicate invert flow and transport is of low significance to waste isolation. The invert has a short travel pathway relative to the geologic barriers and is not expected to have a significant effect on radionuclide transport in the aqueous phase. Although the invert is likely to consist of porous or crushed rock material with desirable properties for radionuclide sorption and possibly colloid filtration, it is quite thin compared with other porous materials in the pathway of radionuclide transport, such as the Calico Hills vitric unit and alluvium. Performance assessment studies showed essentially no effect of eliminating the invert as a barrier (Mohanty, et al., 2004).

The release of radionuclides from the engineered barrier system 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 wasteforms must degrade. Thus, radionuclide transport from the engineered barrier system into the unsaturated zone is dependent on a complex series of events in the potential repository (CRWMS M&O, 2001b). Several factors will affect the mobilization and transport of radionuclides through the engineered barrier system: (i) drip shield performance, (ii) waste package performance, (iii) cladding performance, (iv) wasteform dissolution rates, (v) entry and movement of water through the waste package, (vi) solubility limit for each radionuclide, (vii) radionuclide transport through and out of the waste package, (viii) radionuclide transport through the invert, and (ix) radionuclide transport via colloids.

The DOE conceptual model for engineered barrier system flow abstraction relies on several key elements. Flow through the engineered barrier system is abstracted to a 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 system). 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 (CRWMS M&O, 2000o). Separation of the drip shields in response to rockfall, seismic events, or thermal expansion is assumed by DOE not to occur.

The DOE conceptual model for engineered barrier system 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 system 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 system, DOE made several assumptions, as discussed in the analysis and model report (CRWMS M&O, 2001b). 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 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.

- The potential for evaporation in and on the waste package is ignored.

- The stainless steel components of the waste package, which include the inner liner and inner lid, provide no resistance to corrosion or flow.

- Radionuclide transport through a stress corrosion crack is assumed limited to diffusive transport through a thin, continuous film that is always present (i.e., 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.

- 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).

A description is provided on the approach and technical basis for abstraction of the engineered barrier system flow and transport and integration into total system performance assessment analyses. Overall, the available information is sufficient to expect that the information necessary to assess engineered barrier system flow and transport with respect to system description and model integration will be available at the time of a potential license application.

5.1.3.4.4.9.2 Data and Model Justification

The DOE abstraction of the engineered barrier system flow and transport relies on a bounding approach because of the uncertainty in the response of a complex engineered system for long periods of time. Radionuclide transport out of the wasteform and waste package, through the invert, and into the unsaturated zone is dependent on a complex series of events in the repository. Data to support the DOE abstraction of the engineered barrier system flow and transport are presented in CRWMS M&O (2001b) and in references cited in the document. The NRC staff evaluation is ongoing with respect to sufficiency of data for model justification.

Overall, it appears the available information is sufficient to expect that the information necessary to assess the DOE abstraction of the engineered barrier system flow and transport with respect to data being sufficient for model justification will be available at the time of a potential license application.

5.1.3.4.4.9.3 Data Uncertainty

DOE made several assumptions in its abstraction of flow and radionuclide transport processes in the engineered barrier system to bound the uncertainties in the model parameters (CRWMS M&O, 2001b). The NRC evaluation is ongoing with respect to data uncertainty in the DOE abstraction.

Overall, the available information is sufficient to expect that the information necessary to assess the DOE abstraction of the engineered barrier system flow and transport with respect to data uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.4.4.9.4 Model Uncertainty

DOE made several assumptions in its abstraction of flow and radionuclide transport processes in the engineered barrier system to bound the uncertainties in the conceptual models (CRWMS M&O, 2001b). The NRC evaluation is ongoing with respect to model uncertainty in the DOE abstraction.

Overall, the available information is sufficient to expect that the information necessary to assess the DOE abstraction of the engineered barrier system flow and transport with respect to model uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.4.4.9.5 Model Support

DOE made several assumptions in its abstraction of flow and radionuclide transport processes in the engineered barrier system (CRWMS M&O, 2001b). The NRC evaluation is ongoing with respect to model support.

Overall, the available information is sufficient to expect that the information necessary to assess the DOE abstraction of the engineered barrier system flow and transport with respect to model abstraction being supported by objective comparisons will be available at the time of a potential license application.

5.1.3.4.4.10 Near-Field Criticality

DOE has not included near-field criticality as part of the radionuclide release rates and solubility limits model abstraction. DOE indicated it intends to exclude nuclear criticality events from the performance assessment based on low probability. The DOE evaluation of nuclear criticality is assessed in Section 5.1.2.2, Identification of Events with Probabilities Greater Than 10^{-8} Per Year.

5.1.3.4.5 Summary and Status of Key Technical Issue Subissues and Agreements

Table 5.1.3.4-1 provides the status of all key technical issue subissues referenced in Section 5.1.3.4.2 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 review methods discussed in Section 5.1.3.4.4. Note the status and detailed agreements pertaining to all the key technical issue subissues are provided in Table 1.2-1 and Appendix A.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (e.g., specified testing or analyses), indicates that information necessary to begin a technical review will likely be available at the time of a potential license application.

Table 5.1.3.4-1. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreement*
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	CLST.4.01 through CLST.4.11

Table 5.1.3.4-1. Related Key Technical Issue Subissues and Agreements (continued)			
Key Technical Issue	Subissue	Status	Related Agreement*
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.5.04 CLST.5.05 CLST.5.07
Evolution of the Near-Field Environment	Subissue 3—Effects of Coupled Thermal-Hydrological-Chemical Processes on the Chemical Environment for Radionuclide Release	Closed-Pending	ENFE.3.03 ENFE.3.04 ENFE.3.05
	Subissue 4—Effects of Coupled Thermal-Hydrological-Chemical Processes on Radionuclide Transport Through Engineered and Natural Barriers	Closed-Pending	ENFE.4.06
	Subissue 5—Effects of Coupled Thermal-Hydrological-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 review methods. Note: Key Technical Issue Agreement GEN.1.01 pertains to multiple integrated subissues, as well as some specific issues related to this integrated subissue.			

5.1.3.4.6 References

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5.1.3.5 Climate and Infiltration

5.1.3.5.1 Description of Issue

The Climate and Infiltration Integrated Subissue addresses features, events, and processes that affect the near-surface hydrologic cycle, such as precipitation, temperature, climate change, vegetation, soil, and shallow bedrock properties. These features, events, and processes strongly influence the rate of net infiltration which, in turn, affects deep percolation and the rate at which water reaches the potential repository horizon. Relationship of this integrated subissue to other integrated subissues is depicted in Figure 5.1.3.5-1. 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 supporting analysis and model reports cited in the following sections. This section documents current NRC understanding of the abstractions of climate and infiltration incorporated by DOE into its total system performance assessment. Because the technical basis document for climate and infiltration had not been released as of the preparation of this section, some values or interpretations attributed to DOE may change by the time of the potential license application. The assessment is focused on those aspects most important to repository safety based on the risk insights gained to date, including the NRC Risk Insights Baseline Report (Appendix D). The scope of the assessment presented here is limited to examining whether the data gathered and methodology developed by DOE are likely to be adequately documented for the staff to undertake a detailed technical review of the potential license application. This assessment is not a regulatory compliance determination review of a license application.

5.1.3.5.2 Relationship to Key Technical Issue Subissues

The Climate and Infiltration Integrated Subissue incorporates subject matter previously captured in the following eight 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 Geologic 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)

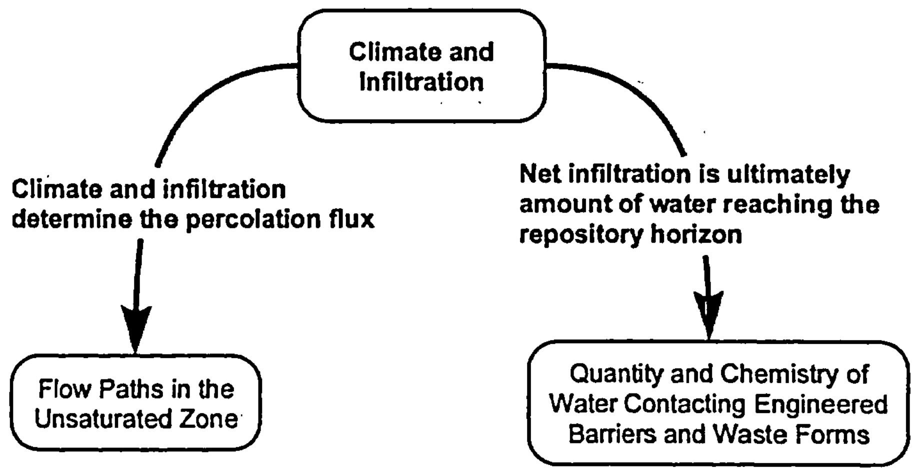

Figure 5.1.3.5-1. Diagram Illustrating the Relationship Between Climate and Infiltration and Other Integrated Subissues. Material in Bold Is Identified in the Text.

- 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 issue resolution status reports and also were the bases for technical exchanges with DOE where agreements were developed on what additional information DOE would provide to NRC. 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.

5.1.3.5.3 Importance to Postclosure Performance

One aspect of risk-informing of the NRC understanding of postclosure repository performance is to determine how this integrated subissue is related to the DOE repository safety strategy. Risk insights pertaining to climate and infiltration indicate that the present-day net infiltration rate and long-term climatic change are of medium significance to waste isolation. The details of the risk insights ranking are provided in Appendix D.

Some precipitation that falls on Yucca Mountain is expected to move into the bedrock as net infiltration. Estimates of present-day net infiltration rates are used to directly estimate the deep percolation rate at the potential repository horizon, assuming no lateral diversion of flow. Some fraction of this deep percolation is expected to seep into the potential repository drifts and contact the engineered barrier system. Water coming into contact with engineered barrier systems could affect corrosion rates. Assuming degradation of the drip shield, the effect of water contacting the waste packages could be detrimental or beneficial, depending on the rate and water chemistry. The release of radionuclides from failed waste packages would be increased by water contacting the wasteform. The quantity of water has the most significant effect on the rate of release of radionuclides that have lower solubility limits. The net infiltration rate and, thus, deep percolation rate also directly affect the transport of radionuclides from the repository potential horizon to the saturated zone.

Because the technical basis document for climate and infiltration had not been released as of the preparation of this section, some values or interpretations attributed to DOE may change by the time of the potential license application. DOE identified surficial soils and topography as natural barriers important to waste isolation (Bechtel SAIC Company, LLC, 2002a; CRWMS M&O, 2000b). Bechtel SAIC Company, LLC (2002b) examined the sensitivity of the mean annual dose estimate to net infiltration. Basecase {modern climate infiltration of 4.6 mm/yr [0.18 in/yr]} results were compared with an infiltration flux of 150 mm/yr [5.9 in/yr] (similar to the modern climate precipitation flux). Results of this analysis indicated only a small change in the mean annual dose from the basecase and an increase in the mean annual dose of only 0.01 mrem [1×10^{-4} mSv] for the disruptive igneous scenario. Although these results indicate the details of the climate and infiltration models do not play a significant role in the estimate of the mean annual dose, these results are contingent on the fact that the drip shield remained intact in the scenario without igneous activity. Even though portions of the drip shield and waste packages were breached in the scenario with the disruptive igneous event, the mean annual dose is weighted by the probability of an intrusive igneous event. The significance of infiltration on waste isolation derived from these sensitivity analyses is strongly conditioned by the abstraction models controlling drip shield and waste package integrities, and the occurrence of the disruptive event.

Using the TPA Version 4.1 code in sensitivity analyses, Mohanty, et al. (2002) determined that the mean areal average infiltration into the subsurface was one of the two most influential parameters contributing to overall peak risk. The peak dose estimates from each realization were also found to be most sensitive to the mean areal average infiltration into the subsurface. In addition, the subarea wetted fraction, which is correlated to mean annual net infiltration, was found to be an influential parameter.

5.1.3.5.4 Technical Basis

NRC developed a review plan (NRC, 2003) consistent with the acceptance criteria and review methods found in previous issue resolution status reports. A status assessment of the DOE approaches for including climate and infiltration in total system performance assessment abstractions is provided in the following subsections. The assessment is organized according to the five review methods identified in Section 2.3: (i) Model Integration (including system description), (ii) Data and Model Justification, (iii) Data Uncertainty, (iv) Model Uncertainty, and (v) Model Support.

5.1.3.5.4.1 Model Integration

Integration of the climate and infiltration conceptual and numerical models into the performance assessment model is described in CRWMS M&O (2000a). The influence of these models on the performance assessment model is only indirect, however, because the representation of climate (other than the timing of climate states) is used only as an input to the infiltration model. The outcomes from the infiltration model are used as input to the site-scale unsaturated zone flow and transport model, which is used to compute deep percolation and seepage at and below the potential repository horizon. Thus, integration of climate and infiltration models into performance assessment involves three levels of abstraction:

- Representation of the climate states in the infiltration model
- Abstractions of the infiltration model output into the unsaturated zone flow and transport model
- Abstractions of the unsaturated zone flow and transport model results into the performance assessment model

The abstraction of the unsaturated flow and transport model into performance assessment is discussed in Section 5.1.3.6.

The approach and technical basis for the abstraction of climate change are documented by DOE in CRWMS M&O (2000a,b). Key assumptions are (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. Based on these assumptions, a 10,000-year climate history, beginning approximately 400,000 years before the present, was selected as the most probable analog for the next 10,000 years. During this period, DOE identified three different climate states: (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,b).

For each climate state, a temporal record of precipitation and temperature boundary conditions was developed from measurements at local and climate analog sites. Lower bound, mean, and upper bound records were created for each climate state. The basis for the choices of analog sites in Washington, Utah, Nevada, Arizona, and New Mexico was the relationship between climate and the movement of the jet stream across the western United States and the geographical distribution of key paleoclimate indicator species (CRWMS M&O, 2000c). The precipitation and temperature record developed for each climate state was used as input to the net infiltration process model (CRWMS M&O, 2000d). For the net infiltration abstraction, DOE also added consideration of climate-induced changes in vegetation during future climates (CRWMS M&O, 2000d). 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 potential repository level to the water table during the monsoon and glacial-transition climate states. The NRC staff, during a Technical Exchange and Management Meeting (Schlueter, 2000), closed Subissue 1, Climate Change, because the staff believed that sufficient information had been submitted to allow them to evaluate the climate abstractions.

The scope of the DOE net infiltration process model is limited to surficial hydrological processes, with estimates of net infiltration defined as water that flows deeper than the root zone. 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 most important portions of the infiltration model domain are the 4.7-km^2 [1.8-mi^2] area of the potential 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 net infiltration estimates as steady-state boundary conditions. Note, however, the potential repository footprint referred to in CRWMS M&O (2000a) is not necessarily the same as the footprint that may be presented in the potential license application. The net infiltration model is documented in CRWMS M&O (2000d).

Processes considered in the net 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 transpiration submodel; an evaporation and net radiation submodel; a snowpack submodel; a two-dimensional (horizontal) surface-water flow-routing submodel; and a volume-balanced model for vertical flow in the shallow, unsaturated zone based on a bucket-routing method. Depending on the climate state, synthetic or measured meteorological data from local or climate analog sites are used as input to the net infiltration model. Combinations of a 15-year precipitation and temperature record developed from multiple local meteorological stations and two 100-year stochastically generated records were 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 (2000c) were used for lower- and upper-bound monsoon and glacial-transition climate net infiltration simulations. The meteorological inputs were 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.

The infiltration model assumes vegetation density and root-zone depth will increase during wetter future climates. The infiltration analysis and model report, however, indicates these changes in vegetation are considered only for the upper-bound future climate scenarios (CRWMS M&O, 2000d, 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 net infiltration. Although the large increases in precipitation assumed for the upper-bound future climate scenarios would reasonably support increased vegetation cover and vegetation types with greater root-zone depth, no basis or sensitivity analysis was presented for the magnitude of the assumed changes. Because the extent and characteristics of the vegetative cover depend on factors other than precipitation such as native plant species, species migration rates, and soil type, it is difficult to access the reasonableness of the magnitude of these changes. DOE agreed to provide justification for use of the evapotranspiration model and use of the analog site temperature data (Reamer, 2001).

Output from the DOE infiltration model is used to define spatially distributed, time-averaged estimates of net infiltration for each climate state as steady-state flux boundary conditions for the site-scale unsaturated zone flow model. The nine boundary conditions for the unsaturated zone flow model consisted of 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 model grid is coarser than that of the infiltration model. Temporal averaging also is 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 unit beneath the surface at Yucca Mountain is postulated to attenuate episodic surface infiltration pulses and spatially smooth localized zones of high infiltration.

The climate and infiltration abstractions are generally consistent with the available data, and the important physical phenomena and couplings are adequately described. Assumptions are clearly stated and used consistently. The climate, infiltration, and unsaturated zone process model reports and supporting analysis 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 net infiltration process model, (iii) the approach and technical basis for the net infiltration model, and (iv) integration of the net infiltration process model into total system performance assessment analyses. The climate process model generally incorporates the important features, events, and processes that may characterize future climates. The net infiltration process model incorporates features, events, and processes important to net infiltration. The spatial scale of the net infiltration process model is consistent with the scale of the site-scale unsaturated zone flow and transport model. The DOE assumption that infiltration rates can be represented as discrete, steady-states is adequately described to allow it to be evaluated. The abstraction of the spatial and temporal variations in net infiltration is dependent, however, on assumptions regarding the smoothing effect of flow through the Paintbrush nonwelded tuff unit.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.5.5), is sufficient to expect that the information necessary to assess climate and infiltration with respect to system description and model integration will be available at the time of a potential license application.

5.1.3.5.4.2 Data and Model Justification

As discussed previously, the representation of climate variations and near-surface conditions and processes significantly affects the estimation of net infiltration and deep percolation. The climate and infiltration models rely on data needed to describe

- Current and future climate states, including soil and vegetative cover, precipitation and temperature

- Physical parameters such as soil and shallow bedrock hydraulic properties, soil thickness, and topography affecting net infiltration

- The relationship between episodic precipitation and infiltration events

The modern climate data used for computing net infiltration were based on averaging precipitation and temperature records from meteorological stations located on the nearby Nevada Test Site and adjusting precipitation to account for orographic effects at Yucca Mountain (CRWMS M&O, 2000d). The longest of the available precipitation records was 30 years. This record was extended to 100 years using a stochastically generated record based on the statistics of the 30-year record. Upper- and lower-bound estimates of net infiltration were developed by selective sampling of the infiltration estimates for the wetter and dryer portions of the actual and stochastic climate records. Average net infiltration derived from the modern climate simulations ranged from approximately 1 to 11 mm/yr [0.04 to 0.43 in/yr] (Table 5.1.3.5-1). Because the stochastic climate record was developed using a relatively short period of record and apparently assumed a stationary process (no trend or periodicity), the stochastic record may not properly represent decadal or longer cycles in precipitation for modern climate conditions. Nevertheless, a relatively large range in net infiltration estimates suggests the range of climatic conditions used as input to the infiltration model was likely sufficient to represent the actual range of infiltration likely to occur during periods of tens to a few hundred years.

Detailed descriptions of the climate data sets and how they can be used to justify the abstraction approach are provided in CRWMS M&O (2000c). 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 stable oxygen isotopes that provides insights about long-term changes in average annual ground water temperature (i.e., climate change) (CRWMS M&O, 2000c). 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 the Earth's orbital precession is evident where maximal values of precession mark the ends of the Devils Hole interglacials and other warm periods (CRWMS M&O, 2000c). This relation was developed to provide a rationale

Table 5.1.3.5-1. 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	1.3 [0.051 in/yr]	4.6 [0.18 in/yr]	11.1 [0.44 in/yr]
Monsoon	4.6 [0.18 in/yr]	12.2 [0.48 in/yr]	19.8 [0.78 in/yr]
Glacial-Transition	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. Rev. 00. Las Vegas, Nevada: CRWMS M&O. 2000.

for timing future climate change based on 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 ostracods from cores drilled at Owens Lake were used (CRWMS M&O, 2000c). Owens Lake is located on the eastern side of the Sierra Nevada Mountains, east of Los Angeles. The known environmental tolerances of ostracod 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 were then used to qualitatively infer a range of likely climate conditions—namely precipitation and temperature—during the Owens Lake stage 11 (interglacial period approximately 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 over the next10,000-year period. The DOE justification for selecting this sequence of future climate states is adequately explained in the documents available for this report.

Once qualitative descriptions of future climate states were obtained from the Owens Lake record, analog sites were identified where present-day climate conditions are qualitatively consistent with those inferred for the monsoon and glacial-transition climates (CRWMS M&O, 2000c). Meteorological stations within these analog areas were selected to obtain precipitation and temperature data 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. Net 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 manner in which the analog sites were selected (CRWMS M&O, 2000c) and the climate records used as input to the infiltration model are adequately documented in CRWMS M&O (2000a). The NRC staff, during a Technical Exchange and Management Meeting with DOE (Schlueter, 2000), closed Subissue 1, Climate Change, because the staff believed that sufficient information had been submitted to allow them to evaluate the climate abstractions.

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. Short periods of heavy precipitation (including an occasional snowmelt) produce short duration surface run-on and stream flow events. The data also indicate areas with thin soils and highly fractured bedrock permit rapid infiltration of water below the root zone. These data and observations generally are 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. The data generally are consistent with the infiltration model simulations in that both indicate net infiltration only occurs after infrequent, significant precipitation events.

The characterization of bedrock hydraulic properties plays an important role in the infiltration model and relationships between net infiltration and deep percolation. The DOE infiltration model uses a lumped parameter approach to simulate the vertical flow of water in the shallow soil and bedrock that requires estimating an equivalent bedrock hydraulic conductivity for the fractured tuff. Bedrock hydraulic conductivity is a sensitive parameter for net infiltration estimates where soils are thin or nonexistent (NRC, 1999). According to CRWMS M&O (2000d), the equivalent bedrock hydraulic conductivity for the various bedrock units in the infiltration model was computed assuming the shallow fractures are filled with a material having a saturated hydraulic conductivity of 43 mm/d [0.14 ft/d]. The effective bedrock hydraulic conductivity was then calculated as the area-weighted average of the fracture-filling hydraulic conductivity and the matrix hydraulic conductivity. The averaging was performed assuming a fracture aperture of 250 microns {250×10^{-6} m [8.2×10^{-4} ft]} and estimates of the fracture densities in the various bedrock units.

Alcove 1 is the only location where large-scale infiltration measurements into soil and bedrock have been made at Yucca Mountain. According to Bechtel SAIC Company, LLC (2003), infiltration rates up to approximately 30 mm/d [0.1 ft/d] could be sustained without producing surface runoff. The test site is underlain by the Tiva Canyon member of the Paintbrush Tuff and the sustained infiltration rate can be assumed to approximate the equivalent, saturated hydraulic conductivity of the shallow bedrock fracture and matrix system. According to CRWMS M&O (2000d), the equivalent bedrock hydraulic conductivity assigned to the Tiva Canyon Tuff units in the net infiltration model ranged from 0.06 mm/d [2×10^{-4} ft/d] to approximately 14 mm/d [0.046 ft/d], with most values less than 1.0 mm/d [3×10^{-3} ft/d]. The Alcove 1 test was performed in an area underlain by Tiva Canyon caprock and the upper lithophysal unit, which were assigned values of 0.35 and 1.13 mm/d (1.1×10^{-3} and 3.7×10^{-3} ft/d). Thus, the Alcove 1 infiltration tests indicate net infiltration rates for areas of Yucca Mountain with thin soil underlain by Tiva Canyon Tuff or with exposed Tiva Canyon Tuff could be higher than those estimated from the net infiltration model.

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 net 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. This issue also relates to the manner in which shallow, lateral flow is simulated in the infiltration model. It is not immediately clear if a more rigorous treatment of near-surface and overland lateral flow processes would result in greater or lesser focusing of net infiltration at locations such as the bottom of steep slopes and wash bottoms. The amount and rate of near-surface and overland flow is affected by numerous variables, including the intensity and duration of precipitation, soil thickness, soil and bedrock hydrologic properties, slope and roughness of the ground surface, amount and type of vegetation, evapotranspiration potential, and antecedent soil moisture conditions. The overall effect may be that net infiltration is more variable spatially than is predicted by the model.

Net infiltration is highly sensitive to soil thickness. The potential repository footprint is dominated by thin soils. Characterizing soil thickness over a 30-m [98-ft] pixel—the grid size for the net infiltration model—is 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] over a 30-m [98-ft] distance. The approach described in CRWMS M&O (2000d) 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 be constrained only by qualitative visual observations in the field because of the highly irregular bedrock surface. Although the DOE approach leads to qualitatively reasonable results, uncertainty in soil thickness estimates for the potential repository footprint where the soils are dominantly thin leads to uncertain results. This uncertainty, combined with the uncertainty in the constraints on the model results described in Section 5.1.3.5.4.5, leads to uncertain model results, particularly for future climate conditions. DOE agreed (Reamer, 2001) to propagate such uncertainty through the abstraction in the total system performance assessment as described in Sections 5.1.3.5.4.3 and 5.1.3.5.4.4.

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, 2000d). Lateral runoff routing is incorporated by tracking the amount of water 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 soil. In response to the NRC request (Reamer, 2001), DOE provided technical arguments in a letter (Ziegler, 2003) and a report (Rickertsen, 2003) that the water-balance plug-flow model adequately represents the nonlinear flow processes such as are represented by Richards' equation, particularly for the potential repository where there is thin soil. The Rickertsen report presents multiple lines of evidence called upon to indicate that local point estimates of the net infiltration rate at Yucca Mountain are as high as 80 to 100 mm/yr [3 to 4 in/yr], whereas the DOE plug-flow submodel provides grid cell estimates of the net infiltration rate that range between 0 and 250 mm/yr [0 and 9.8 in/yr] for the present-day climate state. Independent analysis methods provide estimates for a constrained range of net infiltration rates above the potential repository horizon {e.g., neutron-logging, 10–30 mm/yr [0.4–1.2 in/yr] and borehole temperature profiles, 5–12 mm/yr [0.2–0.47 in/yr]} based on interpretation in Rickertsen (2003). The technical content and references in the Rickertsen report provide adequate information on the DOE approach. The significance of not evaluating nonlinear, unsaturated flow processes on net infiltration at Yucca Mountain has not been determined. Incorporating unsaturated flow processes into the infiltration model could either increase or decrease the net infiltration estimates.

The DOE infiltration model was calibrated by comparing simulated to measured stream flow in five sub-watersheds. Data from two storms were used (CRWMS M&O, 2000d). The streamflow generated by the two events varied by nearly a factor of ten. 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, the approach may lack the ability to estimate accurately the net infiltration because 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.

Increased or focused infiltration could be important to performance assessment evaluations because of the resulting potential for localized increased seepage into repository drifts that could mobilize radioactive waste in the event of a waste package failure. Hence, NRC requested DOE to demonstrate that the effects are appropriately considered of near-surface lateral flow on the spatial variability of net infiltration. DOE responded to this request in a letter (Ziegler, 2003) and a report (Rickertsen, 2003). Although DOE originally agreed to provide such a demonstration, the agency subsequently decided on an alternative approach of demonstrating that multiple lines of evidence support the current net infiltration estimates and, in any event, total system performance assessment calculations are not affected significantly by net infiltration and drift seepage rates. Staff found the DOE total system performance assessment did not adequately account for the effects of near-surface and overland flow in the net infiltration submodel (Schlueter, 2003a). Staff suggested net infiltration estimates from portions of the DOE submodel could be compared with estimates obtained using a smaller model (such as a sub-watershed model) that treats overland and near-surface flow with more physically based numerical methods. Alternatively, multiple lines of field evidence could be used to evaluate quantitatively the range of uncertainty in modern net infiltration. These two approaches could be used to determine reasonable bounds for the net infiltration values in the total system performance assessments.

In summary, much of the available data at Yucca Mountain have been collected using acceptable techniques, and the conceptual models for climate and infiltration are generally consistent with the available site-specific data. Review of the paleoclimate data for the Yucca Mountain region and meteorological data from climate analog sites indicates these data have been collected using acceptable techniques. Although the DOE net infiltration model adequately includes important features and processes, direct measurements of net infiltration are lacking, values of some parameters (such as equivalent bedrock hydraulic conductivity) and certain model process simplifications (such as those for vertical flow and lateral flow) are uncertain. The net infiltration estimates, however, can be supported 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 net infiltration abstraction (see Section 5.1.3.5.4.3), adequate DOE and NRC agreements and sufficient data exist to support development of the net infiltration process model for Yucca Mountain.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.5.5), is sufficient to expect that the information necessary to assess climate and infiltration with respect to data being sufficient for model justification will be available at the time of a potential license application.

5.1.3.5.4.3 Data Uncertainty

The estimates of infiltration and deep percolation for modern and future climate conditions are affected by uncertainties in

- Data used to describe the characteristics of future climate states

- Data used to describe the timing of the sequence of climate states

- Data used to describe the hydraulic and other properties affecting net infiltration

With regard to the description of future climate states, the DOE approach assumes past and future climate states have and will be controlled by the same climate systems that currently affect climate in the western United States (CRWMS M&O, 2000c). The nature of future climates is thus controlled by north-south shifts in the major air circulation patterns.

Specific characteristics of future climates are derived from interpretation of the influence of climate on microfossil assemblages in Owens Lake, specifically species of ostracods and diatoms. Because these aquatic organisms are sensitive to the temperature and salinity of the water body they inhabit, changes in their makeup and abundance in Owens Lake were used to infer changes in temperature and precipitation that were then correlated with a climate chronology. Analog meteorological stations were identified that were believed to have modern climate conditions similar to those inferred from the microfossil interpretations at Owens Lake. The selection of analog stations was based, in part, on the geographical distribution of ostracod species characteristic of the inferred paleoclimates at Owens Lake. Uncertainty exists in the selection of analog sites, however, because as noted in CRWMS M&O (2000c), a key microfossil indicator of the Monsoon Climate, Limnocythere bradburyi, is currently found in lakes in areas with mean annual precipitation varying from 284 mm [11 in] (Lordsburg, New Mexico) to as high as 2,000 mm [79 in] in Central Mexico.

CRWMS M&O (2000a) identifies several sources of data uncertainty with respect to the sequence and timing of future climate states. First, there is uncertainty in knowing whether changes in stable oxygen isotope ratios 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 and the stable oxygen isotope content of the Devils Hole calcite. Second, each Devils Hole sample integrates a particular thickness of carbonate in a continuous sample series and represents approximately 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 used to infer relative ages of the microfossils obtained from cores in Owens Lake. 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 calcite age 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 the standard deviation is unknown. A final source of uncertainty is the choice of a starting point—400,000 years before the present—assumed equivalent to modern climate for purposes of projecting forward.

Two important uncertainties pertaining to climate change are the timing of the onset of climate change and the magnitude of temperature and precipitation changes that may occur as a result of the climate change. DOE and NRC use different approaches to represent future climatic conditions. NRC uses a smooth transition from the modern climate to a glacial-transition climate, combined with random sampling of a precipitation multiplier and a temperature shift. DOE uses an instantaneous step-function approach, combined with mean, upper-bound, and lower-bound precipitation and temperature records, which results in higher estimates of net infiltration rates for the next 10,000 years. The DOE step-function approach is based on recent evidence presented in the scientific community supporting much faster climate transitions than previously believed likely to occur. The DOE mean, upper-bound, and lower-bound precipitation and temperature records are based on measurements obtained from a range of analog sites that are believed to adequately bound the likely magnitude of climate changes that might occur at Yucca Mountain. The NRC staff, during a Technical Exchange and Management Meeting with DOE (Schlueter, 2000), closed Subissue 1, Climate Change, because the staff believed that sufficient information had been submitted to allow them to evaluate the climate abstractions.

Another model uncertainty is that periods of climate transition may lead to increased net infiltration as vegetation and soil thickness (e.g., erosion) do not immediately adjust to the new climate conditions. The net infiltration modeling included increased evapotranspiration losses because of assumed increases in vegetative cover and root-zone depth in both the monsoonal and glacial-transition climate simulations. The abstraction for performance assessment uses the average infiltration rates from these simulations (along with stochastic sampling) and assumes specific step changes in the climate states. In actuality, periods may exist during which the vegetative cover may not be adjusted to the climate, so net infiltration could be higher than predicted.

To address data uncertainty in the net infiltration model, DOE developed distributions for values of 12 input parameters to the infiltration process model (CRWMS M&O, 2000e, Table 4-1). These input parameters were sampled stochastically using a Latin hypercube sampling algorithm in a 100-realization Monte Carlo analysis of infiltration for a glacial-transition climate state. CRWMS M&O (2000e) did not, however, provide evidence that 100 realizations would adequately represent the uncertainty distribution. The parameters chosen for developing the 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 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, were not clearly described in CRWMS M&O (2000e), and the methods used to deduce these parameter distributions were not transparent to the NRC staff. DOE provided corrections to the model report and additional justification for the uncertainty analysis in a transmittal letter (Ziegler, 2002) and report (Wang and Zhu, 2002). The NRC staff responded to this report (Schlueter, 2003b) noting additional information should be provided on the technical bases for the parameter ranges.

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, (CRWMS M&O, 2000e, Table 6-2). For example, for a total system performance assessment realization with stochastically sampled inputs, there is a 48-percent chance 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 (Reamer, 2000) to provide the documentation sources and schedule for the Monte Carlo method for analyzing infiltration.

In summary, staff identified several concerns related to the propagation of data uncertainties in the abstraction of climate and infiltration. In each case, however, 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 those abstraction approaches in which uncertainty is not incorporated (such as in the deterministic approach used to estimate magnitude, type, and duration of climate change). DOE agreed 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 is expected to provide sufficient information for review.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.5.5), is sufficient to expect that the information necessary to assess climate and infiltration with respect to data uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.5.4.4 Model Uncertainty

In addition to uncertainties in model results caused by uncertainties in the input data, the results of models of climate and infiltration are affected by the following two uncertainties inherent in the models

- Characteristics, duration and time of future climate states

- Processes controlling net infiltration

Available information indicates that the most significant model uncertainty is not knowing the magnitude 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 (2000c, 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 the common ostracods and diatoms found in Owens Lake, thus integrating the biology, hydrology, and climate linkages 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. The DOE estimates of annualized mean, lower-, and upper-bound values of precipitation and temperature for the three climate states are listed in Table 5.1.3.5-2. These annualized values are for comparison only; actual inputs to the infiltration process model are time varying on a daily basis (CRWMS M&O, 2000d).

DOE also considered alternative approaches for establishing the future climate states such as extrapolating from the climate trends for the last 18,000 years or using global climate circulation models (CRWMS M&O, 2000a). DOE argues that extrapolating future climate from the last 18,000-year record would lead to uncertainties that increase with time, so that the projection would be unreliable within 10,000 years. Without further analysis, however, it is difficult to determine if this approach would be any more or less uncertain than the approach actually used. With regard to the use of global climatic models, DOE argues the capabilities of these models currently are limited to projections for relatively short timeframes (CRWMS M&O, 2000c). The time limitations of global climatic models notwithstanding, such models might not be any more reliable than the analog approach taken by DOE.

It can be seen in Table 5.1.3.5-2 that the ranges of precipitation between lower and upper bounds for all climate states are quite large; hence, a large range of model uncertainty is incorporated into the abstraction. Note 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), but have a more rigorous technical basis by 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 input conditions defined by the climate abstraction, and general uncertainty in the validity of various conceptual model assumptions. Although the fundamental watershed processes controlling the transformation of precipitation to net infiltration are reasonably well known, uncertainty in the model results from the manner in which these processes are mathematically implemented in the model. Uncertainties related to the model descriptions of vertical and lateral water movement have been discussed previously.

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 the combined uncertainty. The approach described in CRWMS M&O (2000a), however, is not sufficient 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 (Table 5.1.3.5-2). 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 5.1.3.5-1. Note net infiltration flux to the unsaturated zone flow model is spatially variable; the values in Table 5.1.3.5-1 are averaged for the unsaturated zone flow model domain and are used for comparison only.

It is not clear that model parameter uncertainty has been fully 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 considered acceptable for the viability assessment (DOE, 1998). DOE proposed to address this NRC concern using the following approach (Reamer, 2000): (i) develop an

Table 5.1.3.5-2. 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. Rev. 00. Las Vegas, Nevada: CRWMS M&O. 2000.

upper-bound infiltration case based on the 90th percentile from the Monte Carlo analysis of the glacial-transition climate documented in CRWMS M&O (2000e), (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 (2000e).

At a technical exchange (Reamer, 2000), the 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, the DOE staff explained, because selecting the high-infiltration scenario from the end of the Monte Carlo distribution translates to a decreased probability this scenario would occur. It was stated the revised probability weighting factor for the high-infiltration scenario will be approximately 20 percent. Although the weighting factor is lower, total system performance assessment simulations would still sample a reasonably large proportion of high-infiltration scenarios.

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 the NRC staff (NRC, 1999) and appear reasonable for the current abstraction. Staff is concerned the range of net infiltration estimates used for the abstraction is not sufficient to bound the model and parameter uncertainties in the net 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.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.5.5), is sufficient to expect that the information necessary to assess climate and infiltration with respect to model uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.5.4.5 Model Support

As discussed previously, the representation of climate variations affects the estimation of net infiltration and deep percolation. For the most part, the climate and infiltration models enter into the total system performance assessment only indirectly. The climate model provides ranges of meteorological data that are input to the net infiltration model. The output from the net infiltration model then provides the time-averaged flux boundary conditions for the unsaturated flow model that computes deep percolation fluxes at the potential repository horizon. The climate model enters directly into the total system performance assessment through timing and duration of climate states.

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 Yucca Mountain and subject to the same climate cycles because regional changes in 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 approximately 7.9–8.5 °C [46.2–47.3 °F]. Although these estimates are uncertain, DOE maintains that they provide an independent and objective precipitation estimate for the last full glacial climate at Yucca Mountain consistent with the mean estimated for the glacial-transition climate (Table 5.1.3.5-2). In addition, DOE feels that the uncertainty in the estimates from packrat middens is conservatively bounded by upper-bound glacial-transition estimates (Table 5.1.3.5-2).

The future climate infiltration estimates were derived from meteorological records at the climate analog sites. These sites were selected to have modern climates similar to those inferred from the microfossil record at Owens Lake for the various paleoclimate states and thus to represent the future climate states. Validity of the analog climate sites depends on the conceptual model of the Earth-based climate systems that currently affect climate in the western United States (CRWMS M&O, 2000c). Although selection of the analog climate sites and application of their modern meteorological records cannot be independently validated, objective criteria were used to select the sites, and the precipitation and temperature records for the analog sites are consistent with those inferred from the packrat midden study mentioned previously. The NRC staff, during a Technical Exchange and Management Meeting with DOE (Schlueter, 2000), closed Subissue 1, Climate Change, because the staff believed that sufficient information had been submitted to allow them to evaluate data supporting the climate abstractions.

Net infiltration is directly related to climatic and surface conditions. Precipitation events of sufficient magnitude to produce net infiltration are infrequent and may be separated by years. According to DOE, near-surface processes such as evaporation, plant transpiration, and surface runoff reduce net infiltration to approximately 5 percent of total precipitation on an annual average basis (Tables 5.1.3.5-1 and 5.1.3.5-2). Modeling indicates that net infiltration is highest along the Yucca Mountain crest and the eastward trending ridge tops, because of the combination of thin soils, greater precipitation at higher elevations, intermediate permeability of the bedrock units, and high permeability of the open and soil-filled fractures. Surface water runs off toward channels and the toes of steep slopes, and can increase net infiltration at these locations, although these locations represent only a small portion of the potential repository footprint. Thin soil layers allow infiltration to enter fractures in the underlying bedrock more effectively and, thus, potentially escape loss through evaporation. Simulations of bare soil infiltration indicate that mean annual infiltration is strongly dependent on surface soil thickness (Stothoff, 1999; Stothoff, et al., 1997). Mean annual infiltration estimates are generally higher for areas where soil thickness is less than 0.5 m [20 in], except where exposed bedrock promotes runoff, thereby lessening infiltration. In areas with thin soil, once the water-holding capacity of soil is filled, open and filled fractures in the bedrock can transmit water to depths beyond the reach of transpiring plant roots, thus becoming net infiltration.

At Yucca Mountain, DOE data show that most of the potential repository footprint is overlain by thin soil layers less than 0.5 m [20 in] thick, with significant variability across the site. The spatial variation in precipitation, soil thickness, and bedrock properties over the potential repository footprint has been explicitly incorporated into the DOE calculation of the calculation of mean annual infiltration and, thus, deep percolation for each subarea of the potential repository.

For validation of the net-infiltration abstraction, CRWMS M&O (2000d) 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, 2000f) 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 also has 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], whereas samples from the South Ramp yielded estimates of 1–2 mm/yr [0.04–0.08 in/yr] (CRWMS M&O, 2000f). These estimates are broadly consistent with the DOE estimates for spatial distributions of infiltration for the modern climate (CRWMS M&O, 2000d). 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 previously assumed chloride concentration of precipitation and wind-blown soil particles 0.62 mg/L [0.62 ppm] was revised downward {0.30 mg/L [0.30 ppm]} based on historical interpretation of Cl-36 data. Zhu, et al. (2003) developed recharge estimates for the saturated zone and perched water at Yucca Mountain based on the chloride mass balance approach and Cl-36 analyses ranging from 5 to 15 mm/yr [0.20 to 0.50 in/yr]. The lower values were interpreted as representing Holocene (roughly modern) recharge and the higher values late Pleistocene (transitional between glacial and interglacial climates). These recharge estimates are consistent with the net infiltration estimates used in the infiltration model abstraction.

Uncertainties and potential biases are 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. Water samples from the saturated zone may contain a mixture of chloride from local infiltration and regional sources. Chloride measurements from the unsaturated zone 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, at least within the welded tuff units. For this reason, the chloride content of matrix pore-water may not represent that of the fracture water. Potential effects of the differences in fracture and matrix chloride contents are discussed by Lu, et al. (2003) who found matrix chloride concentrations can be influenced by the duration of leaching and chloride contributed by fluid inclusions and the rock minerals. If matrix pore-water concentrations are higher than those in the fracture water, the net infiltration rate will be underestimated when based on the matrix pore-water chloride concentrations.

To gain additional confidence in chloride-based infiltration estimates, the site-scale unsaturated zone flow and transport model, which includes fracture-matrix interactions, used matrix pore-water chloride concentrations in the Exploratory Studies Facility and East-West Cross Drift as calibration targets. Model results indicate a range of percolation flux from 3–10 mm/yr [0.12–0.39 in/yr] (CRWMS M&O, 2000a, Figure 3.8-4). Although this range of infiltration estimates is generally consistent with infiltration model calculations, the meaning of the results is not clear. The results may demonstrate (i) the model is self-consistent with its calibration to those same infiltration rates, (ii) the assumed chloride fluxes at the ground surface can be matched with the matrix chloride concentrations, and (iii) a deficiency exists in using a simple mixing model approach. Chloride content in the subsurface depends on the flux at the ground surface and also on the spatially variable evaporation in the subsurface, particularly in the Tiva Canyon where barometric pumping is likely prominent. Any water losses within the Tiva Canyon would reduce deep percolation to the potential repository horizon; thus reducing any significance it might have in biasing net infiltration estimates based on chloride mass balance.

Neutron probe profiles collected during a 4-year period were used to estimate net infiltration at approximately 98 locations throughout a range of geomorphic sites. The range of net infiltration estimates is 0–80 mm/yr [0–3.1 in/yr] for all geomorphic areas (CRWMS M&O, 2000d); an approximate average of 33 mm/yr [1.3 in/yr] is estimated for ridges and slideslopes only, which dominate the potential repository footprint (CRWMS M&O, 2000d, Figure 6-5). The high value of net infiltration may reflect the correspondence with wetter than average climatic conditions during the short period of measurements in the 1990s. Conversely, neutron probe data reflect minimum estimates because the probes respond primarily to bedrock matrix water content and flow bypassing in fractures may be missed by the probe. In addition, the infiltration model results are not entirely independent from the neutron moisture probe infiltration estimates because the neutron probe data were used in calibrating the evapotranspiration submodel (CRWMS M&O, 2000d). Thus, the neutron probe interpretations do not provide independent justification for the infiltration model.

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 infiltration estimates from borehole temperature profiles. Uncertainty in net 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 DOE performance assessment approach assumes an early and instantaneous transition to a monsoonal climate in an average of 600 years from the present and another instantaneous change to a glacial-transition climate in approximately 2,000 years from present. The assumed timing in the climate model abstraction is based on interpretation of the climate history and sediment deposition rate at Owens Lake. No other justification for this model assumption has been provided, and none may be possible given the intrinsic uncertainties in predicting climate change. The implications of this assumption for performance assessment depend, in part, on other model assumptions related to the behavior of the waste package and engineered barriers.

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 justify the DOE climate forecasts for Yucca Mountain. There is reasonable consistency between the net infiltration estimates from the infiltration model and those obtained from geochemical data, flow and transport modeling, and borehole thermal profiles. Unless predictions of future climate states or net infiltrations are substantially changed in final documents submitted in support of the potential license application, the climate and infiltration abstractions are considered to be adequately supported by independent data and analyses. Considering the manifold uncertainties in both the results of the model and in the independent estimates of net infiltration, however, repository performance should be assessed 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 previous sections address the range of uncertainty in climate change and in the spatial and temporal distributions of infiltration at Yucca Mountain.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.5.5), is sufficient to expect that the information necessary to assess climate and infiltration with respect to model abstraction output being supported by objective comparisons will be available at the time of a potential license application.

5.1.3.5.5 Summary and Status of Key Technical Issue Subissues and Agreements

The Climate and Infiltration Integrated Subissue addresses features, events, and processes that affect the near-surface hydrologic cycle such as precipitation, temperature, climate change, vegetation, soil, and shallow bedrock properties. These features, events, and processes strongly influence the estimated rates of net infiltration which, in turn, affect deep percolation and the rate at which water reaches the potential repository horizon. In the NRC Risk Insights Baseline Report (Appendix D), climate and infiltration were identified as being of medium significance to waste isolation.

Table 5.1.3.5-3 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 to the Climate and Infiltration Integrated Subissue. The agreements listed in the table are associated with one or all five generic review methods discussed in Section 5.1.3.5.4. Note the status and detailed agreements pertaining to all the key technical issue subissues are provided in Table 1.2-1 and Appendix A.

Table 5.1.3.5-3. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreement*
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
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 through 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 review methods.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (e.g., specified testing or analyses), indicates that information necessary to begin a technical review will likely be available at the time of a potential license application. Given that net infiltration for current (modern) climate conditions cannot be directly measured and that uncertainties exist in the (i) characterization of future climate states, (ii) estimation of net infiltration based on field studies, and (iii) estimation of net infiltration using the infiltration model, adequate justification should be provided that the range of infiltration estimates used in the total system performance assessment reasonably bounds the range of uncertainty. Such justification is expected to be provided in the DOE responses to key technical issue agreements addressing this subject.

5.1.3.5.6 References

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5.1.3.6 Flow Paths in the Unsaturated Zone

5.1.3.6.1 Description of Issue

The Flow Paths in the Unsaturated Zone Integrated Subissue addresses features of subsurface geology and processes in subsurface hydrology that affect the distribution and velocity of flow between the shallow subsurface and the water table at Yucca Mountain. The relationship of this integrated subissue to other integrated subissues is depicted in Figure 5.1.3.6-1. The overall organization and identification of all the integrated subissues are depicted in Figure 1.1-2. The DOE description and technical bases for abstraction of flow paths in the unsaturated zone were documented previously in CRWMS M&O (2000a) and several supporting analysis and model reports (CRWMS M&O, 2001a,b; 2000b–u). This abstraction approach is being revised by DOE, but the technical basis document for the unsaturated zone flow abstraction was not available for review at the time of this status assessment. DOE has, however, published a technical basis document that describes the most current conceptual model for water seeping into drifts (Bechtel SAIC Company, LLC, 2003a). DOE has also provided several new or revised analysis and model reports (Bechtel SAIC Company, LLC, 2003b–i) as supporting documentation for the most recent drift seepage model abstraction. Accordingly, this section documents the current NRC understanding of the DOE total system performance assessment abstraction for unsaturated zone flow based on a combination of new and previously reviewed information. The assessment is focused on those aspects most important to repository safety based on the risk insights gained to date, including Appendix D of this report. The scope of the assessment presented here is limited to examining whether data gathered and methodologies developed by DOE are likely to be documented adequately for the staff to undertake a detailed technical review of a potential license application. This assessment is not a regulatory compliance determination review of a potential license application.

5.1.3.6.2 Relationship to Key Technical Issue Subissues

The Flow Paths in the Unsaturated Zone Integrated Subissue incorporates subject matter previously described 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)

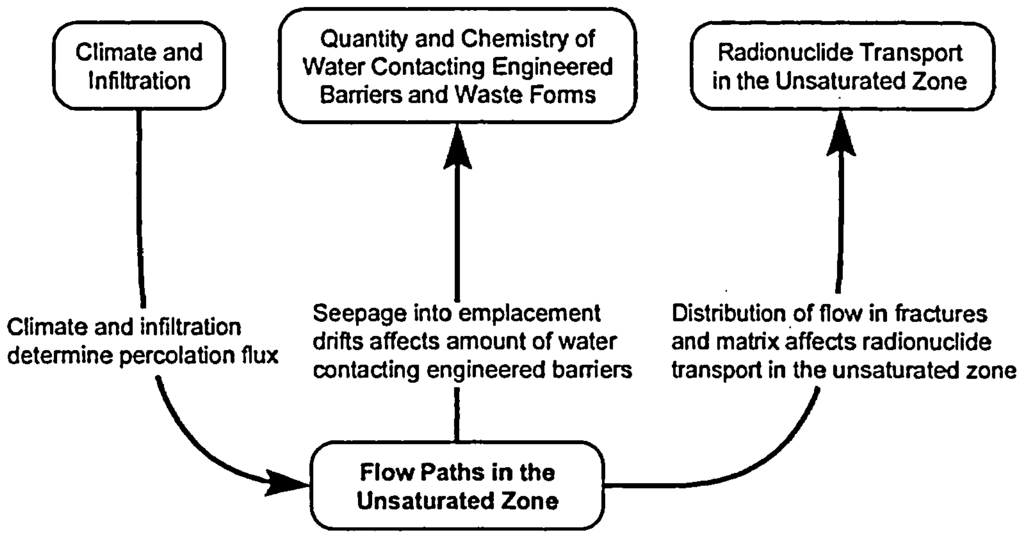

Figure 5.1.3.6-1. Diagram Illustrating the Relationship Between Flow Paths in the Unsaturated Zone and Other Integrated Subissues. Material in Bold Is Identified in the Text.

- Evolution of the Near-Field Environment: Subissue 1—Effects of Coupled Thermal-Hydrological-Chemical Processes on Seepage and Flow (NRC, 2000d)
- 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, 2000e)
- Repository Design and Thermal-Mechanical Effects: Subissue 3—Thermal-Mechanical Effects on Underground Facility Design and Performance (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 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 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 would provide to NRC. 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.

5.1.3.6.3 Importance to Postclosure Performance

One aspect of risk informing the NRC staff understanding of postclosure repository performance is to determine how this integrated subissue is related to the DOE repository safety strategy. Risk insights pertaining to flow paths in the unsaturated zone indicate that seepage is of high significance to waste isolation. Hydrological properties of the unsaturated zone are assigned medium significance, and transient percolation is assigned low significance. The details of the risk insights ranking are provided in Appendix D. 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 retardation of radionuclide transport within the unsaturated zone. The following features and processes significant to waste isolation, for both the nominal and igneous intrusive scenarios, will directly affect seepage into emplacement drifts, retardation of radionuclide transport, or both:

- Attenuation of transient infiltration by the Paintbrush nonwelded hydrogeologic unit
- Spatial redistribution of flow above the potential repository horizon by mechanisms such as capillary barriers, permeability barriers, and flow focusing caused by heterogeneous rock properties
- Near-field conditions affecting flux and spatial distribution of water seeping into potential repository drifts
- Percolation flux from the potential repository into the unsaturated zone, including that from film flow and condensation
- Distribution of flow in fractures and matrix within transport pathways below the potential repository, including the effects on flow of the distribution of zeolitically altered and vitric subunits within the Calico Hills nonwelded hydrogeologic unit

Above the potential 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, 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 is channeled or focused into a small area above a drift or if the water arrives as a transient pulse. The host rock at Yucca Mountain is heterogeneous, fractured, and faulted, thus some amount of focused flow is expected at all scales. Within the potential repository horizon, host-rock properties and engineering design features will affect the quantity of water that contacts drip shields or waste packages, which may affect drip shield or waste package corrosion and mobilize radionuclides in the event of a waste package failure. Below the potential repository horizon, it is necessary to understand how the spatial distribution of hydrologic properties may affect the flow paths from the potential 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 tuff will occur mainly in the rock matrix, yielding a slower transport velocity and providing radionuclides with greater exposure to the surface area of mineral grains for sorption and, thus, retardation. Examples of analyses used to evaluate the importance of the unsaturated zone to total system repository performance are provided in the following paragraphs.

Performance assessment sensitivity analyses by NRC (Mohanty, et al., 2002) using the TPA Version 4.1 code indicate the important aspect of the unsaturated zone flow system above the potential repository horizon to performance is that it limits the amount of water that can reach the waste packages and wasteform. In areas below the potential repository horizon where the Calico Hills vitric unit is present, retardation of sorbing radionuclides is substantial. In these analyses, the mean annual areal average infiltration into the subsurface and the fraction of water condensate moving toward the potential repository both ranked among the 10 parameters that most affect dose estimates for the basecase scenario.

Bechtel SAIC Company, LLC (2002) presents results of performance assessment analyses after neutralizing the barrier potential of the unsaturated zone and results of various seepage models. The basecase seepage model results in zero seepage over approximately 50 percent of waste packages and an average seepage rate of less than 0.1 m^3/yr [26 gal/yr] over the remaining waste packages. The elevated seepage model considers the effect of focusing a seepage rate of 1 m^3/yr [260 gal/yr] over every waste package. Results from the nominal scenario indicate no significant difference in the first 10,000 years, during which the drip shields remain intact. Results from the igneous intrusive scenario indicate an increase in mean annual dose of approximately a factor of 10 because of the release of solubility-limited plutonium isotopes associated with the increase in the amount of water contacting the waste. This unsaturated zone sensitivity study was conducted using the DOE basecase model, with the assumption that the calculated release from the potential repository drifts is discharged directly into the saturated zone. Results from this analysis show a demonstrable change in the mean annual dose for the nominal case and for the igneous intrusive scenario, indicating the effectiveness of the unsaturated zone transport pathways as part of a natural barrier system.

Appendix D states that seepage of water into drifts determines the amount of water that comes into contact with the drip shields and waste packages. Appendix D also states that seepage may affect the rate of corrosion of the drip shield and waste package. The amount of seepage may affect the formation of salts on the surfaces of the drip shield and waste package. Chemistry of the seepage water, however, may have a more significant effect on the formation of salts than the quantity of seepage water. The issues of quantity and chemistry of water contacting waste packages and drip shields are discussed further in Section 5.1.3.3. The seepage of water into drifts also controls the release and transport of lower solubility radionuclides (e.g., Np-237 and Am-241). Although seepage is the primary mechanism for transporting radionuclides out of the waste package, the significance of seepage is limited because only a small quantity of water is needed to mobilize radionuclides with high solubility limits (e.g., I-129 and Tc-99), thus, the dose attributable to these radionuclides is not significantly affected by the amount of dripping water.

Appendix D focuses on unsaturated zone hydrologic properties below the potential repository horizon. For unsaturated zone flow paths that occur mainly within fractured welded or zeolitized tuff units, where matrix conductivities can be significantly lower than the percolation rate, ground water travel times from the potential repository horizon to the water table are on the order of a few tens of years because water flows primarily in fractures. Longer travel times, on the order of several hundreds of years, are estimated for areas beneath the potential repository where the Calico Hills nonwelded vitric unit is present (Mohanty, et al., 2002, Section 3.3.5). The longer travel times in this unit are attributed to its relatively large matrix permeability such that water tends to flow in the matrix rather than in the fractures. The areal extent and thickness of this unit are considered to be moderately important aspects of unsaturated zone flow and transport at Yucca Mountain.

Appendix D divides the integrated subissue of flow in the unsaturated zone into three parts, each with a different level of significance to waste isolation. Seepage into drifts has high significance; hydrologic properties in the unsaturated zone have medium significance, and transient percolation has low significance to potential repository performance. It should be noted that the reason transient percolation is considered to have low significance to waste isolation is because the hydrologic properties of the Paintbrush nonwelded hydrogeologic unit are believed to greatly attenuate the transient nature of percolation below the root zone. Thus, while it may be of little importance to directly consider transient flux in the site-scale unsaturated zone flow model, it is considered of medium importance to verify that the hydrologic properties of the Paintbrush nonwelded hydrogeologic unit are indeed capable of reducing transient infiltration to an effectively steady-state condition. The following assessment of the DOE characterization and performance assessment abstraction of unsaturated zone flow paths was conducted at a level of detail commensurate with the assigned degree of significance.

5.1.3.6.4 Technical Basis

NRC developed a review plan (NRC, 2003) consistent with the acceptance criteria and review methods in previous issue resolution status reports. A status assessment of the DOE approaches for including flow paths in the unsaturated zone in total system performance assessment abstractions is provided in the following subsections. This assessment is organized according to the five review methods identified in Section 2.3: (i) Model Integration (including system description), (ii) Data and Model Justification, (iii) Data Uncertainty, (iv) Model Uncertainty, and (v) Model Support.

5.1.3.6.4.1 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 potential repository horizon and to evaluate transport pathways to the water table. Output from the DOE infiltration model is upscaled to define spatially distributed, time-averaged estimates of net infiltration for each climate state as steady-state flux boundary conditions for the site-scale unsaturated zone flow model. The nine boundary conditions for the unsaturated zone flow model consist of 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 model grid is coarser than that of the infiltration model (CRWMS M&O, 2000a). 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 relatively high matrix permeability of the Paintbrush nonwelded hydrogeologic unit above the potential repository horizon is postulated to attenuate episodic surface infiltration pulses and spatially smooth localized zones of high infiltration. This approach does integrate spatial and temporal variability of net surface infiltration into the site-scale unsaturated zone flow model. Staff has questions, however, related to justification of the assumptions of spatial and temporal averaging of flow within the Paintbrush nonwelded hydrogeologic model layer, which DOE has agreed to address. These questions are discussed in subsequent sections on Data and Model Justification and Model Uncertainty.

The site-scale unsaturated zone flow model used for the site recommendation (CRWMS M&O, 2000a) represents complex geology and stratigraphy using 32 layers with differing hydrologic properties. The model layers dip to the east and are offset by numerous faults that are explicitly considered. The potential repository horizon described for the site recommendation transected three different units of the Topopah Spring Tuff, with approximately 10 percent located in the middle nonlithophysal unit, 78 percent in the lower lithophysal unit, and 12 percent in the lower nonlithophysal unit (CRWMS M&O, 2000d). The proportion of these units that may be intersected by a repository design for a potential license application is currently unknown. Each model layer is assigned homogenous, isotropic hydrologic properties, with the exception of layers representing the Calico Hills nonwelded hydrogeologic unit, which are assigned hydrologic properties for either vitric or zeolitically altered rock types. Based on available information, this approach is expected to adequately represent hydrologic and structural features.

The DOE unsaturated zone flow model indicates that flow in the Topopah Spring welded tuff directly below the potential repository will occur mainly as rapid flow in fracture networks. When flow paths reach the underlying Calico Hills nonwelded tuff unit, the distribution of flow between fractures and matrix is expected to be spatially variable depending on the degree to which the rock matrix is zeolitically altered. Based on available information, variability of the Calico Hills nonwelded hydrogeologic unit is expected to be adequately included in the DOE site-scale unsaturated zone flow model by reproducing observations of perched water bodies, which are found primarily in the northern part of the potential repository footprint, overlying low permeability, sparsely fractured zeolitized portions of the Calico Hills unit. The perched water bodies result in reduced flow through porous matrix of the Calico Hills unit and lateral flow to nearby faults. Model results (CRWMS M&O, 2000a) indicate 35 percent of deep percolation in the model domain reaches the water table through faults. Radionuclide transport studies using unsaturated flow fields from the mean modern infiltration scenario indicate rapid flow in fault zones is a significant transport pathway for arrival of nonsorbing species at the water table (e.g., CRWMS M&O, 2000e, Section 6.12).

Before resulting percolation fluxes from the unsaturated zone flow model are input into the seepage abstraction, the flow rates are modified to account for effects of flow focusing at scales larger than the seepage model, but smaller than the grid-scale of the site-scale unsaturated zone flow model (Bechtel SAIC Company, LLC, 2003e, Section 6.6.4.2). Summarizing the DOE approach, percolation fluxes below the interface of the Paintbrush nonwelded/Topopah Springs welded units are sampled for thousands of locations above potential repository drifts; flow focusing factors are randomly sampled for each location, using a cumulative probability distribution of flow focusing factors ranging from 0.116 to 5.016 (Bechtel SAIC Company, LLC, 2003e, Figure 6.6-15); local flux values are multiplied by the sampled flow focusing factors to provide local percolation flux estimates for input to the drift seepage abstraction. The distribution of flow focusing factors is based on results of intermediate-scale, heterogenous, unsaturated zone flow modeling (Bodvarsson, et al., 2003) and is designed to provide conservation of mass of the sampled percolation fluxes. Based on available information, this abstraction approach for the distribution of flow above the potential repository horizon is expected to adequately incorporate the effects of focused flow caused by heterogeneity of rock properties.

Output from the site-scale unsaturated zone flow model is integrated into total system performance assessment analyses in two ways. First, estimates of fracture flow rates below the interface of the Paintbrush nonwelded hydrogeologic unit with the Topopah Springs welded hydrogeologic unit are used as input for the seepage abstraction (Bechtel SAIC Company, LLC, 2003e, Section 6.6.4.1). Second, calculated flow vectors in both fracture and matrix continua are used to delineate nine sets of unsaturated zone flow fields (three for each of three climate states), which are input for the abstraction of radionuclide transport in the unsaturated zone. The abstraction of radionuclide transport in the unsaturated zone is discussed in Section 5.1.3.7 of this report. The drift seepage abstraction is discussed in the following paragraphs.

The current version of the Seepage Model for Performance Assessment (Bechtel SAIC Company, LLC, 2003d) is significantly different from that used for the site recommendation (CRWMS M&O, 2000g). Similar to the previous approach, the current model uses a three-dimensional, drift-scale, heterogenous fracture continuum to provide a range of seepage estimates that account for spatial variability and uncertainty of hydrologic properties and percolation fluxes. The current approach reflects several improvements, including a conceptual framework and a level of grid refinement designed to be consistent with the Seepage Calibration Model used to simulate *in-situ* seepage testing. The model domain of the Seepage Model for Performance Assessment represents the upper left half of the drift and is 10.0 m high × 4.0 m wide × 2.4 m [32.8 ft × 13.1 ft × 8.0 ft] along the drift axis. The dimensions of each numerical grid cell are 10 cm high × 10 cm wide × 30.5 cm [4.6 in × 4.6 in × 12 in] along the drift axis. To represent the drift/wall interface, the nodal distance between the surface of the wall and the grid cell representing the open drift is set to be very small so that the drift boundary condition is effectively applied directly at the wall. The length of the last vertical connection between the wall and the neighboring gridblocks representing the geologic formation is set equal to 5 cm [1.97 in]; horizontal diversion is not allowed to occur below this last vertical connection. DOE indicates this representation of the drift interface implicitly accounts for small-scale surface asperities with less than 5 cm [1.97 in] of vertical irregularity (Bechtel SAIC Company, LLC, 2003a, Appendix D). The dimensions and numerical grid of the Seepage Model for Performance Assessment are consistent with the Seepage Calibration Model used to estimate model parameters from *in-situ* testing, and enough information is provided to perform a review of the DOE estimates of flux and the spatial distribution of water seeping into the potential repository drifts.

Thermohydrological models are used in two ways to estimate seepage flux into drifts. Additionally, thermohydrological models are used to estimate waste package temperature and relative humidity without regard to the seepage abstraction results (CRWMS M&O, 2001b), which is discussed further in Section 5.1.3.3. For unsaturated flow paths, ambient seepage is modified to account for the effect of the thermal pulse in the DOE seepage abstraction. Two models for incorporating thermohydrology effects are proposed by Bechtel SAIC Company, LLC (2003e) for use in the total system performance assessment. The first model excludes thermal effects and uses ambient seepage rates throughout the performance period. The second model specifies a zero seepage rate when the drift wall is above the boiling temperature and ambient seepage rates when the drift wall is below the boiling temperature. A third model, not proposed for use in the total system performance assessment, is used by DOE to illustrate reduced seepage rates when thermohydrological processes are directly modeled using a dual-continuum approach (Bechtel SAIC Company, LLC, 2003e). This numerical model lacks adequate parameterization, which is why it is not proposed for use in the total system performance assessment. DOE considers results from alternative models for flow in fractured rocks to reflect the upper bound in uncertainty analyses (Bechtel SAIC Company, LLC, 2003a,e). For example, Bechtel SAIC Company, LLC (2003a) provided a summary of the Phillips (1996) steady-state model for preferential flow breaching the dryout zone and the effect of incorporating transient behavior. Although seepage flux was less when transient behavior was included, seepage from preferential flow breaching the dryout zone was nonzero. NRC believes there is little basis for excluding preferential flow through the dryout zone, regardless of the dryout zone thickness, from the basecase seepage abstraction.

The effects of drift degradation on seepage rates have also been considered for the current abstraction of drift seepage (Bechtel SAIC Company, LLC, 2003e, Section 6.4.2.4). In nonlithophysal host rock units, changes to drift geometry are expected to result from local breakout of key blocks (Bechtel SAIC Company, LLC, 2001). Seepage modeling results for the key block breakout scenario show only a small effect on seepage rates, which is within the standard deviation of seepage estimates already included in the seepage abstraction (Bechtel SAIC Company, LLC, 2003d, Section 6.6.3). Different drift degradation modes for lithophysal units (Bechtel SAIC Company, LLC, 2003g) prompted DOE to also consider a drift degradation scenario in which the original drift opening is increased in size, but is filled with fragmented rubble and large voids. Simulation results indicate most of the percolation flux is still diverted around the collapsed drift, but seepage rates are larger for the collapsed drift scenario because the drift footprint is assumed to approximately double in size, thereby doubling the amount of percolation flux arriving at the collapsed drift. Increased seepage entering a drift does not necessarily translate to increased water contacting waste packages, however, because the footprint of the waste package remains unchanged. The collapsed drift scenario is integrated into the seepage abstraction by using a lookup table for collapsed drift seepage estimates that considers the same ranges of capillary strength, mean fracture permeability, and percolation flux as in the basecase abstraction. Based on available information, this approach is expected to adequately incorporate drift degradation processes and the resulting effects on seepage.

Within the drift, the effect of the sampled seepage rate depends on the scenario being evaluated in the DOE performance assessment model. For the nominal nondisruptive scenario, seepage is assumed to be diverted away from the waste packages by the drip shield barrier and, therefore, cannot directly contact waste packages or wasteforms. Thus, performance assessment dose estimates for the nominal scenario are relatively insensitive to seepage rates (Bechtel SAIC Company, LLC, 2002). For an igneous intrusion scenario, the waste packages and drip shields contacted by the intrusion are assumed to fail instantaneously and then be directly exposed to contact by seepage water. The result is that dose estimates for the igneous intrusion ground water release scenario are directly affected by the selected seepage rate (Bechtel SAIC Company, LLC, 2002). A potential staff concern with the integration of seepage with in-drift processes is that film flow on drift walls is not included as a factor that affects the degree of rock or invert saturation at the drift floor. Although film flow along drift walls may not contact waste packages or drip shields, it could result in greater saturation of drift floors or inverts, thereby reducing or eliminating drift shadow effects and enhancing rates of advection and diffusion in the rock below drifts. The staff is concerned that seepage models appropriate for predicting dripping from drift crowns might lead to erroneous conclusions regarding potential rates of advection and diffusion of radionuclides in the near field. For example, while it may be sufficient to treat processes such as flow along drift walls implicitly for predicting dripping at the drift crown, such processes may need to be considered explicitly for modeling drift shadow effects at the drift floor. The staff notes that assumptions about potential drift shadow effects and water content of drift inverts should be supported by modeling or observation data appropriate for those purposes. Additional staff assessment of in-drift processes that may affect radionuclide release rates is provided in Section 5.1.3.4, Radionuclide Release Rates and Solubility Limits. Assessment of the DOE abstraction of the chemical evolution of seepage water is reviewed in Section 5.1.3.3, Quantity and Chemistry of Water Contacting Engineered Barriers and Waste Forms.

Thermal processes that may affect the future spatial distribution of hydrologic properties in the unsaturated zone include thermal-mechanical-induced changes of hydrologic properties near heated drifts, such as potential increases in subhorizontal fracture apertures in adjacent rock pillars, and thermal-hydrological-chemical-induced changes to hydrologic properties, such as porosity and permeability. Thermal-mechanical effects on the distribution of percolation fluxes are not included in the drift seepage abstraction. DOE justifies the exclusion of these processes using a set of model simulations that couple heat-induced stress changes to fracture permeability and the resulting impact on seepage percentage (Bechtel SAIC Company, LLC, 2003e, Section 6.4.4.1). Thermal-mechanical modeling results show calculated seepage rates for the thermally perturbed permeability field are reduced by approximately 10 percent from the values calculated for the initial permeability field. This modeling addresses a previous NRC concern and the related DOE agreement (Reamer, 2001a, Agreement RDTME.3.20) regarding the effects of thermal-mechanical effects on fracture permeability and drift seepage. DOE proposes to exclude thermal-hydrological-chemical-induced changes to hydrological properties based on numerical simulations that show any such changes will have a negligible effect on seepage and flow paths or will not be detrimental to repository performance (Bechtel SAIC Company, LLC, 2003i). The numerical simulations (CRWMS M&O, 2001a) focused on the Topopah Spring welded unit near the potential repository emplacement drifts. Seepage and flow paths may also be affected by thermal-hydrological-chemical-induced changes to hydrological properties of the Paintbrush nonwelded and Calico Hills nonwelded hydrogeologic units. Revision 01 of the drift-scale coupled process models (CRWMS M&O, 2001a) includes the Paintbrush and Calico Hills nonwelded hydrogeologic units; however, results for those units were not provided. DOE agreed (Reamer, 2001b, Agreement ENFE.1.03) to provide additional documentation of results of thermal-hydrological-chemical simulations showing negligible porosity and permeability changes in the Paintbrush nonwelded and Calico Hills nonwelded hydrogeologic units. DOE also agreed (Reamer, 2001b, Agreement ENFE.1.04) to provide additional technical bases for 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. These technical bases have not been received to date.

Several features, events, and processes have been excluded from the Total System Performance Assessment–Site Recommendation abstraction of flow paths in the unsaturated 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. Screening arguments pertaining to the abstraction of flow paths in the unsaturated zone are outlined in CRWMS M&O (2001a). It is expected these features, events, and processes screening arguments will be updated in support of a potential license application. One potentially significant process not included in the DOE performance assessments is the mobilization of water vapor from warm areas in drifts and subsequent condensation in cooler areas. This process has been referred to as the cold-trap process. Observations from the Passive Cross-Drift Hydrologic test indicate the cold-trap process can lead to significant sources of liquid water within drifts when temperature gradients are present. Thus, staff is concerned with the exclusion of this process from performance assessments. To address this concern, DOE agreed (Reamer, 2001c, Agreement TEF.2.05) to represent the cold-trap process in appropriate models or provide a technical basis for its exclusion.

In summary, DOE has used several different computer models to simulate percolation flux, seepage flux, and seepage distribution in the unsaturated zone in one, two, and three dimensions. These simulations have been conducted at different scales ranging from the drift scale to the mountain scale. Based on the information flow outlined by DOE, it is expected the DOE model abstractions will take information from and be consistent with climate, infiltration, and geologic models used in other parts of the total system performance assessment. DOE agreed previously (Schlueter, 2000a) to provide the technical basis supporting its unsaturated zone flow models. DOE currently plans to provide this information to NRC in a technical basis document, however, the document was not yet available for review at the time this reprint was written. DOE also agreed to provide additional documentation of thermal-hydrological-chemical simulations and additional technical bases for treatment of the effects of cementitious materials on hydrologic properties (Reamer, 2001b), as well as to represent the cold-trap process in appropriate models or provide a technical basis for its exclusion (Reamer, 2001c).

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.6.5), is sufficient to expect that the information necessary to assess flow paths in the unsaturated zone with respect to system description and model integration will be available at the time of a potential license application.

5.1.3.6.4.2 Data and Model Justification

An extensive database is available for rock properties of Yucca Mountain hydrogeologic units, including moisture retention characteristics, permeability, porosity, and density. Rock matrix properties were generally measured in the laboratory on samples and cores collected from the site (e.g., Flint, 1998). The permeabilities of fracture networks in differing rock types are estimated from gas injection tests conducted in four niches in the Exploratory Studies Facility and a fifth in the Enhanced Characterization of the Repository Block cross drift (Bechtel SAIC Company, LLC, 2003c, Section 6.1). To date, DOE estimates 3,500 separate gas injections have been undertaken in the underground studies at Yucca Mountain, yielding nearly a quarter of a million pressure-response curves. These data provide a reasonable basis for the conceptual treatment of fracture networks as continuous interconnected media and for the incorporation of heterogeneous permeability fields in the seepage model for performance assessment. A staff concern regarding justification for the treatment of rock properties above the potential repository horizon is the lack of data to support the treatment of geologic contacts within the Paintbrush nonwelded hydrogeologic unit as laterally continuous capillary barriers, as described by Wu, et al. (2000). DOE agreed (Reamer, 2001d, Agreement GEN.1.01, Comment 24) to address this concern by providing additional information to justify the treatment of rock properties in model sublayers that represent the Paintbrush nonwelded hydrogeologic unit.

Because data from Borehole USW UZ–7a, used to characterize the Ghost Dance fault, represent the most complete data set from within a fault zone at Yucca Mountain, these data were applied to all faults in the site-scale unsaturated zone flow model for the site recommendation (CRWMS M&O, 2000a). Fault zones may act as fast pathways and are a potential concern for repository performance. Cl-36 data have been used to evaluate the potential existence of fast flow paths from the land surface to the potential repository horizon (Bechtel SAIC Company, LLC, 2003c, Section 6.1.4.2.1). Elevated concentrations of Cl-36 are indicators of the Cl-36 bomb-pulse that occurred during aboveground nuclear testing more than five decades ago. The earliest investigations of Cl-36 in the Exploratory Studies Facility reported several locations of elevated Cl-36 concentrations mainly associated with the presence of faults, although several of these locations had no clear association with faults. A subsequent systematic sampling study of bomb-pulse Cl-36 in the Exploratory Studies Facility by Lawrence Livermore National Laboratory, however, found no clear evidence of the presence of bomb-pulse Cl-36 at potential repository depths (Bechtel SAIC Company, LLC, 2003c, Section 6.1.4.2.1). DOE agreed to provide a Cl-36 validation study to reconcile these differing results (Schlueter, 2000a, Agreement USFIC.4.04; Reamer, 2001e, Agreement TSPAI.2.02, Comment J–20). Until the conflict is resolved, however, it is reasonable for DOE to continue using the site-scale unsaturated zone flow model that is (i) consistent with the earlier findings that bomb-pulse Cl-36 has penetrated to potential repository depths and (ii) includes relatively fast flow paths associated with fault zones that intersect the Paintbrush nonwelded hydrogeologic unit.

Observations of discontinuous perched water bodies below the potential repository horizon provide a conceptual basis for the modeler's treatment of rock and fault properties that affect flow and transport pathways between the potential repository horizon and the water table. Data from pumping tests were collected to evaluate the spatial extent of the perched water bodies, and water samples were collected for age dating. One modeling objective of the site-scale unsaturated zone flow model was to reproduce the observations of perched water encountered in boreholes at both the vitrophyre between the Topopah Spring welded units and Calico Hills nonwelded units and at the vitric-zeolitic interface within the Calico Hills nonwelded unit (CRWMS M&O, 2000a). These observations appear to support the modeling treatment of the Calico Hills unit as a heterogeneous hydrogeologic unit that variably results in vertical flow through vitric units and lateral flow atop low-permeability zeolitic units.

Test data and modeling results are available from several *in-situ* tests to justify the modeling approach used for the seepage abstraction (Bechtel SAIC Company, LLC, 2003b, Section 6.6.2; 2003c, Sections 6.2 and 6.11). Data from the Enhanced Characterization of the Repository Block Systematic Hydrologic Characterization tests and from Niche 5 were used in the Seepage Calibration Model to determine seepage-relevant parameters for the lower lithophysal zone of the Topopah Spring Tuff. For these tests, relative humidity and evaporation rate data were explicitly considered in the process models used to obtain calibrated parameter estimates. Data from Niches 3 and 4 were used for parameter estimation in the middle nonlithophysal zone of the Topopah Spring Tuff. Evaporation effects were determined not to be significant for these tests because ambient relative humidity was near 100 percent. The results obtained from forward modeling of the Seepage Calibration Model to match *in-situ* test data (Bechtel SAIC Company, LLC, 2003b) provide a reasonable demonstration that the modeling approach used to develop the seepage abstraction is applicable for a range of hydrologic and ambient relative humidity conditions in the unsaturated zone at Yucca Mountain. In addition to providing support for the conceptual basis of the seepage abstraction, the range of capillary strength parameters estimated from the Seepage Calibration Model provides justification for the range of parameter uncertainty considered in the seepage abstraction (Bechtel SAIC Company, LLC, 2003b, Section 6.6.4). These recent seepage test data collection and modeling activities address previous staff comments and a DOE agreement (Schlueter, 2000a, Agreement USFIC.4.01; Reamer, 2001e, Agreement TSPAI.3.25) to conduct testing to address ambiguities in seepage test results. Staff also commented previously that an approach needs to be in place to relate observed fracture patterns to possible drift seepage and transport properties. This approach is needed to justify the application of seepage predictions to potential repository drifts not presently in existence. DOE agreed (Schlueter, 2000b, Agreement SDS.3.01) to relate observations of seepage in the passive test in the East-West Cross Drift to full periphery fracture maps and other fracture data. Fracture characterization data from the Alcove 8–Niche 3 test also will be provided. This additional information has not yet been received.

Seepage into drifts also may be affected by thermally driven redistribution of water caused by waste-generated heat. An objective of the repository design evaluation for the site recommendation (DOE, 2001, Enhanced Design Alternative II) was to prevent coalescence of the boiling fronts associated with above-boiling drift temperatures in the rock pillars separating drifts. This design would support condensate drainage in the region between the boiling fronts. To achieve this objective, and to keep the spent nuclear fuel cladding temperature below 350 °C [660 °F], DOE places some reliance on the efficacy of the ventilation system. Results from the in-drift ventilation model presented in CRWMS M&O (2000o) estimated 0.7 for the heat load reduction factor, however, several simplifying assumptions did not appear to be supported by experimental data. To address this concern, a quarter-scale ventilation test was planned for execution at the Engineered Barrier Subsystem Test Facility in North Las Vegas, Nevada (CRWMS M&O, 2000p), and DOE agreed (Reamer, 2001e, Agreement TEF.2.07) to provide results of this test in an update to the ventilation model. Bechtel SAIC Company, LLC (2003h) summarizes this test and notes the inherent limitation in the ventilation model to simulate the measured data along the quarter-scale laboratory test. Nonuniform heat load to the waste canister circumference and, thus, the need to calibrate for variable heat transfer coefficients around the canisters, limited the usefulness of the laboratory test to support the reduction factors. Using an estimated wall rock effective thermal conductivity, basecase estimates of 0.86–0.88 for the heat load reduction factor for drift segment lengths of 600 and 800 m [1,970 and 2,620 ft] are presented in Bechtel SAIC Company, LLC (2003h). Because other models (Bechtel SAIC Company, LLC, 2003g, 2004) are using a heat load reduction factor of 0.9 for ventilation, which increased from the 0.7 value previously used, NRC is carefully reviewing the supporting basis for the ventilation model and parameters.

Another concern related to thermal effects on flow is the lack of data to support models of fracture saturations, extent of dryout, formation of heat pipes, liquid fluxes in heat pipes, and, ultimately, the fate of thermally mobilized water. This concern is important because a key design aspect of the potential repository (Enhanced Design Alternative II) is for thermally mobilized water to condense and drain through the rock pillars between emplacement drifts. Given uncertainties associated with the drift-scale heater test, such as losses of moisture through the bulkhead, and the lack of quantitative measurements of condensation and drainage in fractures, it is not clear if the results of the drift-scale heater test can be used to determine the fate of thermally mobilized water. NRC (2002, Section 5.1.3.6) suggested that measurements of mass losses through the drift-scale heater test bulkhead may help reduce this uncertainty, but, if significant losses have occurred through the bulkhead since the onset of the experiment, it may not be possible to assess those losses. To address this concern, DOE agreed (Reamer, 2001c, Agreement TEF.2.01) to provide a white paper on the technical basis for its understanding of heat and mass losses through the drift-scale heater test bulkhead and the effects of such losses on test results. The white paper (CRWMS M&O, 2001c), while identifying some benefits of the drift-scale heater test, states that (i) complete and accurate measurement of heat and mass flow through the bulkhead is intrinsically difficult, uncertain, and unnecessary; (ii) approximately one-third of the vapor produced by heating is lost through the bulkhead; and (iii) uncertainty in the DOE understanding of moisture redistribution in the drift-scale heater test is considered to be acceptable based on good agreement in the quantitative thermal and qualitative hydrological comparative analyses of corresponding observations, measurements, and simulations. Subsequent analyses by the NRC staff indicate thermal measurements dominate the proof that the drift-scale heater test modeling is accurate, and support for flow processes is masked by the wide range of flow properties that could be used to match the available quantitative thermal and flow data and related qualitative data. Based on the NRC conclusions, DOE agreed that parameter values from the drift-scale heater test could not be used to develop parameter values for other hydrologic or thermohydrologic models in the unsaturated zone, nor could the drift-scale heater test be used to support conclusions that liquid water will not breach the dryout zone and seep into drifts.

In summary, much of the available data on geology and hydrology 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 is expected to continue using a site-scale unsaturated zone flow model that is (i) consistent with the early findings that bomb-pulse Cl-36 has penetrated to potential repository depths and (ii) includes relatively fast flow paths associated with fault zones that intersect the Paintbrush nonwelded hydrogeologic unit and continue into the Topopah Spring tuff. DOE agreed to provide additional information, justification for certain assumptions, and results from several ongoing and planned tests to validate conceptual models for relationships between seepage into drifts and fracture patterns, thermal effects on flow and seepage. DOE provided a summary of the quarter-scale ventilation test (Bechtel SAIC Company, LLC, 2003h) to illustrate the effects of ventilation on the distribution of heat and water in rock pillars between emplacement drifts.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.6.5), is sufficient to expect that the information necessary to assess flow paths in the unsaturated zone with respect to data being sufficient for model justification will be available at the time of a potential license application.

5.1.3.6.4.3 Data Uncertainty

Uncertainties associated with the site-scale unsaturated zone flow model generally exist in estimated matrix, fracture, and fault hydrologic properties, such as capillary retention parameters and porosity, because of sparse data and limitations of the estimation procedures. Because these properties cannot be readily measured, they are indirectly estimated from other measurements such as gas 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 are fixed before calibration, whereas the remaining properties are further adjusted during model calibration. Thus, many parameter values used in the site-scale unsaturated zone flow model are more a product of calibration than of site data analysis (CRWMS M&O, 2000m). The data uncertainty of the parameters estimated through model calibration is handled within the site-scale unsaturated zone flow model by obtaining calibrations for high-, medium-, and low-infiltration scenarios. The DOE approach does consider a range of uncertainty for (i) the proportion of percolation flux that occurs in fractures, which is a key input to the seepage abstraction; and (ii) the distribution of flow in fractures and matrix below the potential repository, which is a key input to the unsaturated zone radionuclide transport abstraction.

Uncertainty related to the effects of host-rock heterogeneity on the distribution of flow above the potential repository horizon is accounted for by using a flow-focusing factor to adjust percolation flux inputs to the drift seepage abstraction. These flow-focusing factors are randomly sampled for discrete drift locations using a cumulative probability distribution of values ranging from 0.116 to 5.016 (Bechtel SAIC Company, LLC, 2003e, Figure 6.6-15). The distribution of flow-focusing factors is based on results of intermediate-scale, heterogeneous, unsaturated zone flow modeling (Bodvarsson, et al., 2003). This modeling activity incorporated uncertainty in permeability distribution of fracture networks by using model grids with heterogeneous fracture permeability values that vary by four orders of magnitude. Two different spatial correlation lengths,1 and 3 m [3.3 and 10 ft], were also considered in the modeling analysis. The range of permeability values and correlations is consistent with data from *in-situ* testing (Bechtel SAIC Company, LLC, 2003c, Section 6.1). The modeling analyses also included variable flux boundary conditions and showed the range of estimated flow focusing factors is largely insensitive to differing flow rates and flow distributions at the top model boundary. This approach for estimating and including the effects of spatially variable rock properties above potential repository drifts appears reasonable and is documented sufficiently to conduct a review.

The conceptual model used to develop the calibrated property sets for the site-scale unsaturated zone flow model for the site recommendation is described in CRWMS M&O (2000m). The flow model treats each hydrogeologic unit as homogeneous, with the exception of the Calico Hills nonwelded layer, which is divided into zeolitic and vitric regions. For drift-scale ambient and thermohydrological models, fracture permeability is considered a known parameter with values of 3.3×10^{-13} and 9.1×10^{-13} m^2 [0.33 and 0.92 Darcy] for the Tsw34 and Tsw35 model layers (Bechtel SAIC Company, LLC, 2003i); upscaling to the mountain scale using pneumatic data results in no change to these values. There is no uncertainty for fracture permeability. Spatial variability within a model layer is smoothed, or averaged-out by the use of homogeneous properties, which may be adequate for coarsely gridded models, but not for finely gridded models. A statistical analysis of gas-injection data collected from the niches in the Exploratory Studies Facility, however, found fracture permeabilities ranging from 1.53×10^{-15} to 7.15×10^{-10} m^2 [0.002 to 720 Darcies]. These data, all collected in the Tsw34 unit, indicate heterogeneity of fracture permeability can span at least four orders of magnitude within a single geologic unit. It is not clear how using homogeneous properties in a model layer can adequately represent variability and uncertainty that may range several orders of magnitude within a single geologic unit. Additional studies applying generally accepted methods of stochastic subsurface hydrology, sensitivity, and bounding analyses may be required to address the data and model uncertainties. DOE agreed (Reamer, 2001c, Agreement TEF.2.08) to (i) provide documentation of analyses of spatially heterogeneous fracture permeability using grid refinement for the heterogeneous fields in three dimensions and (ii) evaluate the effect of high-permeability features (e.g., fractures and faults) crossing the drifts. Increased fracture heterogeneity affects the magnitude of flow focusing of percolation near the drifts. DOE agreed (Reamer, 2001c, Agreements TEF.2.08 and TEF.2.09) to include heterogeneity in model properties that affect flow focusing of percolating water. Agreement TEF.2.08 is concerned with drift-scale models of seepage and the possibility of liquid water breaching the dryout zone; the DOE submittal for this agreement (Bechtel SAIC Company, LLC, 2003a) is currently being reviewed by the NRC staff. Agreement TEF.2.09 is concerned with the amount of flow focusing estimated by the Multiscale Thermohydrological Model, which provides percolation estimates to the seepage model. NRC reviewed the inclusion of fracture heterogeneity (using hypothetical statistical parameters) in a three-dimensional thermohydrological model for Agreement TEF.2.09 and determined the approach for incorporating heterogeneity was adequate, but using seepage quantity to determine the importance of heterogeneity was inappropriate. The thermohydrological model did not represent seepage in a manner consistent with the seepage process model and seepage abstraction.

An NRC concern regarding the implicit treatment of data uncertainty by calibrating to high-, medium-, and low-infiltration flux scenarios is that this approach has not been demonstrated to adequately account for the effects of measurement error, bias, and scale dependence in the saturation, water potential, and pneumatic pressure test data, which are used to calibrate and constrain model parameter values. 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); 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 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 approximately two orders of magnitude greater than those determined from gas-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 does not fully propagate uncertainty 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 (Reamer, 2001c, Agreement TEF.2.10) 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 significance to waste isolation. DOE also agreed (Reamer, 2001c, Agreement TEF.2.11) to provide an update to CRWMS M&O (2000m), which would incorporate uncertainties from all significant sources in the calibration process for site-scale parameters used in the unsaturated zone mountain-scale ambient and coupled process models and the drift-scale thermohydrological models. The calibrated properties model (Bechtel SAIC Company, LLC, 2003i) has been provided to NRC and is currently being reviewed.

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*. Preliminary field-based monitoring results from the Enhanced Characterization of the Repository Block cross drift indicate the rock mass in the potential repository horizon is wetter (i.e., water potentials are higher) and moisture is more uniformly distributed than indicated by earlier laboratory core analyses (Bechtel SAIC Company, LLC, 2003c, Section 6.10). Also, measurements of water potential taken in surface-based boreholes (e.g., Rousseau, et al., 1999, pp. 145–151) have gradually re-equilibrated to ambient conditions that are much wetter than the data used to calibrate the unsaturated zone flow model for the site recommendation. If the more recent measurements are validated, the staff concern is that the calibrated site-scale unsaturated zone flow model should be consistent with the validated findings. DOE agreed (Reamer, 2001e, Agreement TSPAI.3.26) to use recent, more equilibrated saturation and water potential data when calibrating the unsaturated zone flow model.

Potentially important data uncertainties that can affect drift seepage estimates include those used to estimate mean fracture-network permeability, variability and correlation length of fracture permeability, and the capillary strength of fracture networks intersecting drifts. Data uncertainties for mean fracture-network permeability and capillary strength are addressed by considering ranges of values for each of these parameters. The uncertainty range for mean fracture-network permeability of the seepage model includes 17 different values spanning five orders of magnitude. This range is consistent with the variability of *in-situ* test data and also reflects uncertainty related to the effects of drift excavation (Bechtel SAIC Company, LLC, 2003d, Section 6.3.2). Uncertainty in the capillary strength parameter, $1/\alpha$, is included by using a range of values from 100 to 1,000 Pa [1.45×10^{-2} to 1.45×10^{-1} psi], which is also consistent with the mean and standard deviation of this parameter estimated from the Seepage Calibration Model (Bechtel SAIC Company, LLC, 2003b, Table 16). For each parameter combination of mean fracture-network permeability and capillary strength, 20 different stochastic realizations of heterogeneity are considered using basecase values of 1.0 for the $\log_{10}$ standard deviation of permeability and 0.3 m [1.1 ft] for the fracture-network permeability correlation length. The technical bases for the probability distributions used for mean fracture-network permeability and capillary strength in the seepage abstraction (Bechtel SAIC Company, LLC, 2003e, Sections 6.5 and 6.6) are described in detail sufficient enough to conduct a review. DOE determined the standard deviation and the correlation structure do not need to be varied in the seepage abstraction because the basecase estimates for these parameters produced seepage rates either comparable to or larger than seepage rates calculated from selected sensitivity cases (Bechtel SAIC Company, LLC, 2003d, Section 6.6.2, 2003e, Section 6.4.2). Based on available information, this consideration of data uncertainties in the drift seepage abstraction is expected to include those parameters most significant to seepage flux and its spatial distribution into potential repository drifts, and the uncertainty ranges are expected to be reasonably based on appropriately conducted *in-situ* testing.

Thermal-chemical effects on seepage are also excluded from the current abstraction approach, based on numerical simulations that show any such changes will have a negligible effect on seepage and flow or will not have detrimental effects on performance (Bechtel SAIC Company, LLC, 2003i). DOE agreed (Reamer, 2001b, Agreement ENFE.1.05) 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 has been partially documented (CRWMS M&O, 2001a; Bechtel SAIC Company, LLC, 2003i). Additional supporting reports (Bechtel SAIC Company, LLC, 2003f) necessary to complete documentation of the uncertainty evaluation have not been received.

In summary, several concerns are related to the consideration of data uncertainty in the model abstractions about flow paths in the unsaturated zone. To address these concerns, DOE agreed to provide additional analyses or information to support the abstraction approach. Analyses will (i) 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 provide technical bases that a reduced representation is appropriate, given significance to waste isolation; (ii) include heterogeneity in model properties affecting flow focusing of percolating water; and (iii) use as input recent and, thus, more equilibrated, saturation and water potential data when calibrating the site-scale unsaturated zone flow model. Additional information needed includes justification for calibrating models to high-, medium-, and low-infiltration flux scenarios rather than explicitly accounting for effects of measurement error, bias, and scale-dependence associated with field data; consideration of fracture patterns, low flow-regime processes, and small-scale tunnel irregularities in the seepage abstraction; and consideration of data uncertainty in the multiscale thermohydrologic model and in the thermal-hydrological-chemical process model. DOE has revised its calibrated properties model to include uncertainties in the calibration process for site-scale model parameters (Bechtel SAIC Company, LLC, 2003i). This document is currently in review.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.6.5), is sufficient to expect that the information necessary to assess flow paths in the unsaturated zone with respect to data uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.6.4.4 Model Uncertainty

Input data from Geologic Framework Model 3.1 (CRWMS M&O, 2000s) are used to develop the grid for the site-scale unsaturated zone flow model. DOE attempts to closely match the numerical grid to the Geologic Framework Model 3.1 layers. Because borehole data used to construct this model are limited, there is uncertainty in assumptions regarding lateral continuity and thickness trends of geologic units at Yucca Mountain. Although layers in Geologic Framework Model 3.1 represent a valid interpretation, the effect of greater lateral discontinuity on flow, resulting from the inclusion of small faults, could be significant, especially in areas where little or no information has been collected. Areas of sparse data are generally outside the potential repository area, hence, the effect of this data uncertainty is somewhat mitigated. Numerous fault zones and associated layer offsets within the potential repository area are explicitly included in the unsaturated flow model grid. Hence, although considerable uncertainty exists in the accuracy of the site-scale unsaturated zone flow model grid at any particular location, the grid does allow consideration of the important effects on flow of faults and layer discontinuities at the scale and location of the potential repository.

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 geologic units implies the model grid-block scale is larger than the scale of variability in hydrologic properties (i.e., heterogeneity). It is thus assumed 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 hydrogeologic unit at scales larger than the grid scale near the potential repository. Except for the Calico Hills nonwelded vitric unit, the only heterogeneities considered in the model occur at layer interfaces and where layers are offset by faults. Within the Calico Hills nonwelded vitric 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.

DOE models that assume capillary pressure data for a single borehole are representative of an entire model domain may result in predictions of significant lateral flow along dipping beds that form capillary or permeability barriers; whereas, in reality, naturally occurring heterogeneities would act to limit the extent of lateral diversion. A related concern regarding 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, expressed as height of water). As a result, current models may not be able to adequately represent lateral capillary diversion at layer interfaces. Preliminary modeling by Lawrence Berkeley National Laboratory staff using refined vertical grid discretization has simulated lateral capillary diversion in the Paintbrush nonwelded hydrogeologic unit (Wu, et al., 2000). There is little objective evidence, however, that this phenomenon is occurring at the site (e.g., high matrix saturation or perched water above suspected flow barrier geologic contacts has not been observed). The difference noted between the highly discretized Lawrence Berkeley National Laboratory preliminary model and site observations may be a result of the model not incorporating intralayer heterogeneity in the Paintbrush nonwelded hydrogeologic unit that could interrupt lateral diversion or the model not adequately representing the gradational contacts between subunits. The CNWRA staff is presently evaluating the effects of heterogeneity in the Paintbrush nonwelded hydrogeologic unit on the potential for lateral flow along capillary or permeability barriers located at geologic contacts. Work by Dinwiddie, et al. (2004) provides field evidence of secondary heterogeneities associated with fault zone deformation within the nonwelded Bishop Tuff, an analog to the Paintbrush nonwelded hydrogeologic unit. Work by Ofoegbu, et al. (2001) indicates heterogeneity in the hydrologic properties of the Paintbrush nonwelded hydrogeologic unit, caused by either depositional or secondary overprinting processes (e.g., small fault or slumping), could lead to localized flow focusing beneath the Paintbrush nonwelded hydrogeologic unit. The potentially erroneous prediction of large-scale lateral flow in the Paintbrush nonwelded hydrogeologic unit is a concern because it could lead to underprediction of percolation flux reaching the potential repository horizon. To address this concern, DOE agreed (Reamer, 2001d, Agreement GEN.1.01, Comment 69) to evaluate the potential for lateral diversion of percolating water along flow barriers at geologic contacts and to justify the modeling approach. For current conditions, it is not expected that lateral diversion 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 flow into fault zones, and such an effect could reduce the amount of seepage if DOE could identify faulted zones at depth and avoid placement of waste packages in those areas.

Below the potential repository, where perched water occurs above and within the Calico Hills nonwelded vitric 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 potential 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 potential repository. To further reduce this source of uncertainty, DOE agreed (Reamer, 2001e, Agreement TSPAI.3.24) to provide an analysis of data used to support model predictions of the flow field below the potential repository, particularly in the nonwelded vitric portions of the Calico Hills, Prow Pass, and Bullfrog hydrostratigraphic units.

Another important model uncertainty lies in the use of a steady-state infiltration boundary, which rests on the assumption that the Paintbrush nonwelded hydrogeologic unit acts to completely attenuate infrequent pulses of infiltration predicted by the infiltration model. DOE researchers 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, models 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 potential repository (CRWMS M&O, 2000a, Figure 3.7-11). To address this concern, DOE agreed (Schlueter, 2000a, Agreement USFIC.4.04) to provide additional documentation to support the steady-state infiltration assumption.

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 consist 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 models: (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 show 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 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. These flow fields provide a reasonable approach for bounding the spatial and temporal distributions of water flux in the unsaturated zone.

As previously discussed, the DOE seepage abstraction explicitly considers effects of flow focusing above the potential repository horizon caused by heterogeneous rock properties, as described by Bodvarsson, et al. (2003). Model uncertainties considered when developing the flow focusing abstraction include both uniform and focused infiltration at the upper boundary of the process model, which represents uncertainty in the spatial distribution of flow below the base of the Paintbrush nonwelded hydrogeologic unit. Additionally, both two- and three-dimensional process model simulations were performed. The frequency distribution of percolation fluxes is only slightly narrower in the case of three-dimensional simulations; the three-dimensional simulations had fewer occurrences of flow focusing factors greater than 1.0 (Bodvarsson, et al., 2003, Figure 11). It appears, therefore, reasonable or somewhat conservative to include two-dimensional model results in the development of the uncertainty distribution for flow focusing factors. This abstraction approach does include model uncertainties related to model boundary conditions and dimensionality.

For the drift seepage abstraction, the NRC staff previously raised a concern regarding if the heterogeneous porous continuum modeling approach used in the drift seepage abstraction can be reliably applied to flow in networks of discrete fractures (Schlueter, 2000a, Agreement USFIC.4.06). DOE has addressed this model uncertainty by providing a summary of modeling studies that use differing approaches for explicitly representing fracture networks as discrete features (Bechtel SAIC Company, LLC, 2003a, Appendix D). The DOE modeling studies indicate continuum models with stochastic heterogeneity distributions can provide seepage predictions consistent with the behavior of flow in discrete features. DOE further explains that the seepage abstraction basically serves as a transfer function, based on physical principles and site data, that provides average seepage rates for a range of hydrogeologic conditions. Staff agree with the DOE conclusion that continuum-based seepage models can reproduce observations from *in-situ* seepage tests, and this modeling uncertainty is addressed in a manner sufficient for conducting a review.

Uncertainty in seepage estimates also results from the variability of model results with different realizations of the stochastic heterogeneity fields used in the Seepage Model for Performance Assessment. This uncertainty is included in the seepage abstraction by obtaining model results for 20 different stochastic realizations of permeability for each combination of permeability, capillary strength, and percolation flux considered in the abstraction (Bechtel SAIC Company, LLC, 2003e, Section 6.5.1). The results are means and standard deviations provided in lookup tables for seepage percentage. Thus, in addition to parameter uncertainty considered at each seepage location, model uncertainty is also considered for the variability in seepage percentage estimates resulting from different stochastic realizations of fracture heterogeneity. Based on available information, the DOE approach is expected to include the spatial variability of seepage estimates likely to result from natural variability in *in-situ* fracture patterns.

Another important model uncertainty in the drift seepage process is whether use of the van Genuchten-Mualem model for moisture retention and relative permeability is adequate for modeling unsaturated flow in a fracture network near drifts. The NRC concern related to the seepage abstraction is the effects of film flow, intermittent rivulet flow, and small-scale tunnel irregularities are not explicitly considered in the drift seepage abstraction (Schlueter, 2000a, Agreements USFIC.4.02 and USFIC.4.03). The appropriateness of the van Genuchten-Mualem relationship for flow in fractures, which theoretically could account for the effects of film flow, intermittent rivulet flow, and small-scale drift wall irregularities is also a concern expressed in Agreement TEF.2.13 (Reamer, 2001c), as is representing flow processes along a fracture using spatial averaging of continuum models. Film flow is a term used to describe flow on rough fracture surfaces or drift walls that does not bridge a fracture aperture and, thus, cannot be described by a model of capillary retention based on fracture apertures. Similarly, intermittent rivulet flow does not follow classic porous media capillary retention and has been shown to occur in fractures (e.g., Su, et al., 1999). Drift wall irregularities may lead water to drip points where lateral capillary diversion around the drift opening is not possible. The combination of film flow, intermittent rivulet flow, and small-scale drift wall irregularities theoretically could affect the threshold at which seepage would be estimated if included in the seepage abstraction. This combination could lead to low rates of dripping above many more waste packages than currently accounted for in the seepage abstraction. For Agreements USFIC.4.02 and 4.03, DOE addressed this concern by providing performance assessment sensitivity studies, for both nominal and igneous intrusion scenarios, that use an assumed upper bound seepage rate of 1.0 m^3/yr [264 gal/yr] on every waste package (Bechtel SAIC Company, LLC, 2003k). These sensitivity studies indicate the contribution of low-flow regime processes and small-scale tunnel asperities to the total amount of dripping from drift ceilings would likely constitute only a small fraction of such a high seepage rate, which is a key aspect for resolving these agreement items. Additional model uncertainties are discussed in Birkholzer, et al. (2003); Liu, et al. (2002);

Pruess (1999, 1997); and Phillips (1996). The DOE submittal for TEF.2.13 (Rickertsen, 2003) provides an analysis of fracture heterogeneity and is currently under review.

Another NRC concern related to model uncertainty in the drift seepage abstraction is that fracture patterns affecting seepage rates during *in-situ* testing may not be applicable to fracture patterns in the walls of presently nonexistent repository drifts. Because of this uncertainty, NRC suggested the development of an improved understanding of the role of fracture characteristics in predicting drift seepage. Toward that goal, DOE agreed (Schlueter, 2000b, Agreement SDS.3.01) to relate any observed seepage in the Enhanced Characterization of the Repository Block cross drift passive test to full periphery maps of fractures and to provide a three-dimensional representation of fracture characterization in documenting ongoing Alcove 8–Niche 3 seepage testing. DOE also indicated it will provide a report on fracture and lithophysae analyses of the potential repository host horizon that will synthesize fracture characterization studies, including information from detailed line surveys and full-periphery geologic mapping. One outcome of this effort is that 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 potential repository at Yucca Mountain proceeds, a qualitative basis would exist for evaluating whether fracture patterns in emplacement drifts are consistent with those used in the seepage studies to validate the drift seepage abstraction.

The DOE multiscale thermohydrologic model (CRWMS M&O, 2000t) 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 with actual test results. All thermal tests to date at Yucca Mountain have been conducted in the Tsw34 unit, hence 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 (Reamer, 2001c, Agreement TEF.2.10) to represent the full variability and uncertainty in results of the thermal effects on flow simulations in the abstraction of thermodynamic variables to other models or provide technical basis for why a reduced representation is appropriate. DOE also agreed (Reamer, 2001c, Agreement TEF.2.12) to provide a revision to the unsaturated zone flow and transport process model report that includes consideration of these 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. DOE has not yet provided a basis for completing Agreements TEF.2.10 and TEF.2.12.

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 or will not have detrimental effects on performance (Bechtel SAIC Company, LLC, 2003j). DOE agreed (Reamer, 2001b, Agreement ENFE.1.05) to provide an evaluation of the various sources of uncertainty in the thermal-hydrological-chemical process model, including details regarding how the propagation of various sources of uncertainty are calculated in a systematic uncertainty analysis. Conceptual model uncertainties in these simulations have been partially addressed (CRWMS M&O, 2001a; Bechtel SAIC Company, LLC, 2003j). Additional supporting reports (Bechtel SAIC Company, LLC, 2003f) necessary to complete documentation of the uncertainty evaluation have been received, but were not reviewed for this status report. In addition, DOE agreed (Reamer, 2001b, Agreement ENFE.1.03) to provide additional information about the treatment of fully dry conditions in the reactive transport simulations, including information about the amount of unreacted solute mass trapped in the dryout zone, as well as how this amount would affect precipitation of solutes and the resulting change in hydrological properties. Information contained in Bechtel SAIC Company, LLC (2003j) indicates these concerns are addressed in Revision 2 of the Drift-Scale Coupled Processes Model. Supporting documentation (Bechtel SAIC Company, LLC, 2003f), however, has not been provided.

In summary, several concerns are related to 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 approaches. This additional information includes justification for using a steady-state infiltration boundary; evaluation of the potential for lateral flow diversion and justification of the modeling approach, justification for continuum modeling of a system of discrete features, film flow, intermittent rivulet flow, and small-scale tunnel irregularities in the seepage abstraction; and consideration of parameter and model uncertainties in the multiscale thermohydrological model and in the thermal-hydrological-chemical process model.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.6.5), is sufficient to expect that the information necessary to assess flow paths in the unsaturated zone with respect to model uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.6.4.5 Model Support

Several analyses are available to support the DOE site-scale unsaturated zone flow model for Yucca Mountain. One analysis performed to support the site recommendation was a comparison of the basecase unsaturated zone flow model fluxes with fluxes estimated from observed chloride data from the Exploratory Studies Facility and the Enhanced Characterization of the Repository Block cross drift (CRWMS M&O, 2000u). Results of this analysis indicate 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). 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. A more recent analysis by Flint, et al. (2003), however, concludes that 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 tended to underpredict percolation estimates from the Enhanced Characterization of the Repository Block cross drift data. Results of the Flint, et al. (2003) analyses can also be used to infer lateral diversion of flow in either capillary or permeability layers must be of limited spatial extent and is, therefore, not likely to shed significant amounts of water away from Yucca Mountain. Several other analyses that generally provide support for the magnitude of percolation fluxes predicted by the DOE unsaturated zone flow model are summarized by Flint, et al. (2002). These analyses include monitoring borehole water content profiles using neutron probes, modeling of borehole thermal profiles, and analysis of atmospheric radionuclides. DOE obtained additional model validation from an analysis of calcite minerals in the unsaturated zone. In this analysis, observations of precipitated calcite in the unsaturated zone were used to provide additional evidence for validation of the unsaturated zone flow model. One-dimensional reactive transport modeling of calcite deposition in a deep surface-based borehole (WT–24) was performed to estimate the net infiltration rate. Using a range of infiltration rates from 2 to 20 mm/yr [0.08 to 0.8 in/yr], the simulated calcite abundances generally fell within the range observed in the field. This combination of different analytical methods provides an appropriate level of support for the spatial distribution of percolation fluxes estimated by the DOE site-scale unsaturated zone flow model.

Information provided by DOE to support the seepage abstraction includes results from the Alcove 8–Niche 3 test and the Enhanced Characterization of the Repository Block cross drift passive test, which are not otherwise used in determining seepage-relevant parameters (Bechtel SAIC Company, LLC, 2003a, Appendixes B and C). For the Alcove 8–Niche 3 test, seepage observed in Niche 3 has been consistently less than 10 percent of the infiltration rate applied to the overlying floor of Alcove 8. This low seepage is generally consistent with the seepage process models. For the Enhanced Characterization of the Repository Block cross drift passive test, the limited data available for temperature and relative humidity gradients in the drift suggest that condensation accounts for most of the liquid water that has been observed in droplets and small puddles following long periods (several months or more) of unventilated conditions. No observations of water in the Enhanced Characterization of the Repository Block cross drift passive test have been clearly linked to dripping ambient seepage from drift ceilings. It has not been clearly established, however, that moisture conditions in the sealed-off portion of the cross drift have returned to ambient background conditions following the long period of dryout caused by ventilation. The Alcove 8–Niche 3 tests and the Enhanced Characterization of the Repository Block cross drift passive test are ongoing. Although observations from these tests to date generally support the seepage abstraction, staff emphasize the importance of continuing such tests, especially the passive monitoring tests, to establish a long-term record of observations to validate the seepage abstraction for demonstrably ambient moisture conditions.

In another study, seepage rates were calculated assuming (i) a volume fraction of 0.9 for calcite in mineral coatings and (ii) every coating was deposited during a period of 10 million years. Results of this analysis suggest not all lithophysal cavities encounter seepage and seepage flux derived from mineral deposits is a very small fraction of percolation flux, consistent with the conceptual model used for the abstraction of drift seepage. Such geochemical models are subject to large uncertainties regarding initial and boundary conditions. Even recognizing these model limitations, staff agree these interpretations of calcite mineralization provide support for conceptualization of the DOE unsaturated zone flow and seepage models.

DOE agreed (Reamer, 2001c, Agreement TEF.2.08) to consider the NRC suggestion of comparing the numerical seepage model results with the Phillips (1996) analytical solution as a means of model validation. Finely gridded continuum simulations and a modified Phillips solution are summarized in Bechtel SAIC Company, LLC (2003a) and described more fully in cited reports. The detailed descriptions are currently being reviewed by NRC, including a review of why these alternative models are used only for support of upper bound percolation and seepage estimates and not for basecase estimates.

The low-, medium-, and high-infiltration scenarios for the site-scale 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 is performed to match observations of perched water associated with the Calico Hills nonwelded unit. Thus, the flow model results are reasonably consistent with those observations. However, supporting data for the predicted flow vectors within, adjacent to, and below the perched water are not presented in the process model report or in the analysis and model report. DOE agreed (Reamer, 2001e, Agreement TSPAI.3.24) to provide documentation of the analysis of geochemical and hydrological data used to support the predicted three-dimensional unsaturated zone model flow fields below the potential repository horizon, particularly below the perched water or through the vitric Calico Hills nonwelded unit, Prow Pass, and Bullfrog hydrostratigraphic units. This documentation has not yet been received.

In summary, the site-scale unsaturated zone flow model of Yucca Mountain is broadly consistent with the 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 uncertainties through the abstraction, as discussed in the preceding sections. In particular, DOE previously agreed (Reamer, 2001e) to provide documentation of the analysis of geochemical and hydrological data used to support estimated flow fields below the potential repository horizon. DOE provided a comparison of the numerical seepage abstraction results with the Phillips (1996) analytical solution as a means of model validation. This analysis is currently being reviewed by NRC, as is the reason for only using these alternative models to support upper bound percolation and seepage estimates, and not to support basecase estimates.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.6.5), is sufficient to expect that the information necessary to assess flow paths in the unsaturated zone with respect to model abstraction output being supported by objective comparisons will be available at the time of a potential license application.

5.1.3.6.5 Summary and Status of Key Technical Issue Subissues and Agreements

Likely flow paths through the unsaturated zone pass through both welded and nonwelded volcanic tuffs. The extent of the percolation flux entering potential repository drifts as seepage is important because it will affect the chemistry of any water contacting the drip shield or waste package, which may affect corrosion of Engineered Barrier System materials and radionuclide mobilization. In Appendix D, seepage is assigned high significance, while hydrologic properties of the unsaturated zone are assigned medium significance. Aspects of performance related to retardation of radionuclide transport along flow paths through the unsaturated zone are considered in Section 5.1.3.7.

Table 5.1.3.6-1 provides the status of all key technical issue subissues, referenced in Section 5.1.3.6.2, 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 more of the generic review methods discussed in Section 5.1.3.6.4. Note the status and detailed agreements pertaining to all the key technical issue subissues are provided in Table 1.2-1 and Appendix A.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (e.g., specified testing or analyses), indicates that information necessary to begin a technical review will likely be available at the time of a potential license application.

Table 5.1.3.6-1. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreement*
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 through TEF.2.08 TEF.2.10 through 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 through 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 5.1.3.6-1. Related Key Technical Issue Subissues and Agreements (continued)			
Key Technical Issue	**Subissue**	**Status**	**Related Agreement***
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 review methods.

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.

5.1.3.6.6 References

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5.1.3.7 Radionuclide Transport in the Unsaturated Zone

5.1.3.7.1 Description of Issue

The radionuclide transport in the unsaturated zone model abstraction addresses the migration of radionuclides through the unsaturated zone below the potential 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 system. The rate radionuclides migrate through the unsaturated zone depends on the water flow rate and the flow regime of the water in which the radionuclides travel—fracture flow or porous flow through rock matrix. Radionuclide migration rates also depend on the water chemistry and mineralogy of the geologic system, because these control retardation processes. The relationship of this integrated subissue to other subissues is depicted in Figure 5.1.3.7-1. The overall organization and identification of all the integrated subissues are depicted in Figure 1.1-2. DOE documented its approach to modeling unsaturated zone transport in February 2002 in numerous reports prepared to support the recommendation of the site (CRWMS M&O, 2000a–f; Bechtel SAIC Company, LLC, 2001a,b; DOE, 2001a,b). DOE recently updated its models for colloidal transport of radionuclides (Bechtel SAIC Company, LLC, 2003a). DOE also intends to publish an updated technical basis for unsaturated zone flow and transport, but the supporting reports were not publically available at the time of this assessment.

This section documents the current NRC staff understanding of the model abstractions developed by DOE to incorporate radionuclide transport in the unsaturated zone into its total system performance assessment. The assessment is focused on those aspects that are important to waste isolation based on the risk insights gained to date (Appendix D). The scope of the assessment presented here is limited to examining whether data gathered and methodologies developed by DOE are likely to be adequately documented for the staff to undertake a detailed technical review. This assessment is not a regulatory compliance determination review of a potential license application.

5.1.3.7.2 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 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 4—Deep Percolation [Present and Future (Post-Thermal Period)] (NRC, 2000b)

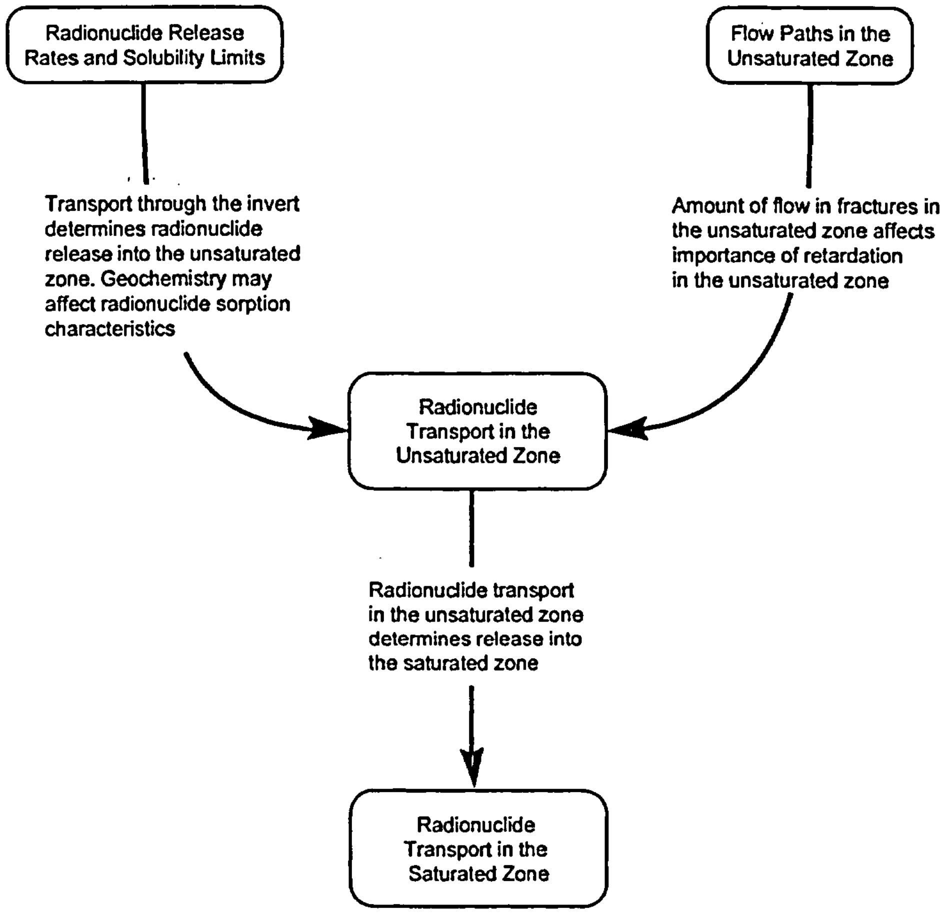

Figure 5.1.3.7-1. Diagram Illustrating the Relationship Between Radionuclide Transport in the Unsaturated Zone and Other Model Abstractions

- 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)

- Structural Deformation and Seismicity: Subissue 3—Fracturing and Structural Framework of the Geologic 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 Near Field? (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 would provide to NRC. 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 of this report incorporate applicable portions of these key technical issues subissues.

5.1.3.7.3 Importance to Postclosure Performance

One aspect of risk informing the NRC staff review was to determine how this integrated subissue is related to the DOE repository safety strategy (Appendix D). DOE identified radionuclide transport in the unsaturated zone at Yucca Mountain as a principal factor of the postclosure safety case (CRWMS M&O, 2000a). DOE also examined the role of the unsaturated zone as a barrier using neutralization analyses (Bechtel SAIC Company, LLC, 2002, Section 3.10). In these analyses, radionuclide transport in the unsaturated zone was demonstrated to be a potentially significant contributor to waste isolation. As described in CRWMS M&O (2000a), diffusion into the matrix and sorption on matrix minerals are important retardation mechanisms. Conceptual models of radionuclide transport between the potential repository horizon and the water table include flow and transport in both fractures and matrix in the volcanic tuffs. Processes considered in the DOE model abstractions of radionuclide transport through the unsaturated zone include advection, matrix diffusion, sorption, dispersion, colloid transport, and radioactive decay.

DOE defined five principal hydrostratigraphic units in the simulation of flow through the unsaturated zone (CRWMS M&O, 2000e, Section 3.2.2). In both the DOE and NRC model abstractions, radionuclide transport through fractures in the volcanic tuffs located between the potential repository and the water table is conservatively considered to be unretarded because of limited characterization regarding the distribution of fracture-lining minerals (DOE, 2001a,b; CRWMS M&O, 2000b; Mohanty, et al., 2002). In contrast to fracture transport, sorption onto minerals in the volcanic tuffs and delay of radionuclide migration are considered to occur within the rock matrix. Sorption parameters are based on a combination of batch experiments, process modeling, and expert elicitation (CRWMS M&O, 2000c; Bechtel SAIC Company, LLC, 2003c).

As part of the total system performance assessment for site recommendation (DOE, 2001a,b; CRWMS M&O, 2000b), the geochemical aspects of the DOE approach for considering radionuclide transport in the unsaturated zone are essentially the same as the approach previously used for viability assessment (DOE, 1998). Transport parameter values, represented by sorption coefficient (K_d) probability distribution functions, have been updated and modified (CRWMS M&O, 2000c; Bechtel SAIC Company, LLC, 2003c). Other changes in the radionuclide transport in the unsaturated zone model abstraction include using updated parameter values and inputs from the unsaturated zone flow model and incorporating the active-fracture conceptual model.

Because the conceptual model provides only 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 for the total system performance assessment for site recommendation, the mean dose rate from the basecase (which includes matrix diffusion) 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 basecase matrix diffusion coefficients (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.

Additional studies also evaluated the effects of the presence of a drift shadow beneath the potential repository, with decreased hydrologic saturation levels and reduced fracture flow. Simulations of this shadow result in decreased fracture transport and increased transport (and consequent retardation) in the tuff matrix. This change in the dominant flow and transport paths could result in a three-order of magnitude increase in unsaturated zone transport time (DOE, 2001b, Section 3.3.7.1; Bechtel SAIC Company, LLC, 2001a, Section 11.3.1).

Total system performance assessment sensitivity analyses using mean parameter values from the NRC TPA Version 4.1 code (Mohanty, et al., 2002, Section 3.3.5) suggest that, for the basecase, radionuclide transport in the unsaturated zone provides a limited reduction in the radionuclide release from the engineered barrier system as the release migrates to the water table. Assuming no retardation at all for plutonium, americium, and thorium in both the unsaturated and saturated zones increases the expected ground water dose by one to three orders of magnitude throughout a 100,000-year simulation period. Assuming no matrix diffusion results in a peak expected dose that is about 450 years earlier and 50 percent higher when compared with the basecase (Mohanty, et al., 2002, Section 3.5.3).

Risk insights pertaining to radionuclide transport in the unsaturated zone indicate that retardation in the Calico Hills non-welded vitric tuff (CHnV), matrix diffusion in the unsaturated zone, and the effect of colloids on transport in the unsaturated zone are of medium significance to waste isolation. The details of the risk insights ranking are provided in Appendix D. The following assessment of the DOE characterization and performance assessment abstraction of radionuclide transport in the unsaturated zone was conducted at a level of detail appropriate to the degree of significance assigned in Appendix D. DOE is planning to provide a technical basis document that updates the unsaturated zone flow and transport model abstraction before a potential license application. This document and many of the supporting references have not been finalized, however, and have not been considered by NRC in this assessment.

5.1.3.7.4 Technical Basis

NRC developed a review plan (NRC, 2003) consistent with the acceptance criteria and review methods documented in previous issue resolution status reports. A status assessment of the DOE approaches for including radionuclide transport in the unsaturated zone in total system assessment abstractions is provided in the following subsections. This assessment is organized according to the five review methods in Section 2.3: (i) Model Integration (including system description), (ii) Data and Model Justification, (iii) Data Uncertainty, (iv) Model Uncertainty, and (v) Model Support.

5.1.3.7.4.1 Model Integration

DOE defined five principal hydrostratigraphic units in the simulation of flow through the unsaturated zone (CRWMS M&O, 2000e, Section 3.2.2). Three of these units, the Topopah Spring welded tuff, Calico Hills nonwelded, and Crater Flat undifferentiated are of particular importance with respect to the performance assessment model abstraction of radionuclide transport from the potential repository horizon to the water table. Faults through the tuffs also are potentially important features for unsaturated flow and transport. The DOE transport models for the unsaturated zone use the same conceptual models, assumptions, and hydrologic parameters as those used in constructing the model abstraction for flow in the unsaturated zone (CRWMS M&O, 2000e, Section 3.11.1). Hydrostratigraphic properties and mineralogy are based on the DOE integrated site model (CRWMS M&O, 2000g) and the DOE mineralogical model (CRWMS M&O, 2002). The flow models for the unsaturated zone are run prior to the transport calculations and, assuming a quasi-steady flow state, the flow fields are saved for use in the total system performance assessment. Because the DOE transport analyses are based on the flow models, there is an explicit internal consistency in the hydrologic parameters and hydrostratigraphy. The current NRC understanding of the flow models is provided in Section 5.1.3.6.

To address uncertainty with regard to the location of waste package failure, release from the engineered barrier system to the unsaturated zone occurs at a random location within one of five discrete zones in the potential repository region. The five zones are based on ranges of infiltration (CRWMS M&O, 2000e, Section 5.2.4; DOE, 2001b, Section 3.3.7). Radionuclide mass transported through the invert at the base of the drift is fed as a boundary condition directly into the fracture network in the unsaturated zone. Staff understanding, however, is that the radionuclide transport abstraction for a potential license application may be modified to consider dissolved radionuclide sources that initiate in the matrix continuum that must then travel by advection or diffusion through the matrix before entering the more rapidly flowing fracture domain. Documentation for the flow and transport abstraction for a potential license application was not available at the time of this assessment.

After transport calculations, the radionuclide mass is collected at the base of the unsaturated zone for each time step and is provided as a boundary condition for transport through the saturated zone at a random location within one of four discrete zones at the water table (CRWMS M&O, 2000e, Section 5.2.4; CRWMS M&O, 2000b, Section 3.7.2). If the water table rises as a result of different climate/infiltration scenarios, transport paths through the unsaturated zone will be shorter. In the DOE total system performance assessment model, any radionuclides below the new water table are transferred directly to the saturated zone for transport.

DOE evaluated radionuclide transport through the unsaturated zone using three different modeling approaches (DOE, 2001b, Section 3.3.7.1). At the mountain-scale, DOE used two- and three-dimensional numerical simulations (CRWMS M&O, 2000e, Sections 3.11.5 and 3.11.6; DOE, 2001a, Section 4.2.8.3). The models include numerous processes that affect radionuclide transport, such as advection, dispersion, sorption, matrix diffusion, radioactive decay, and colloid transport. In the two-dimensional simulations, DOE used two vertical cross sections, based on the stratigraphy of boreholes SD–6 and UZ–14. Fracture flow, where radionuclides are retarded only by matrix diffusion, is dominant in the Topopah Spring welded and zeolitized Calico Hills nonwelded tuffs. Calculated transport times ranged from nearly 1 year for nonretarded solutes, such as technetium, through the Topopah Spring welded, to several thousand years for transport of strongly sorbed plutonium through the zeolitized CHnv. Calculated breakthrough at the water table is on the order of 10^3 years for technetium, and weakly sorbed radionuclides like neptunium reach the water table between 10^4 and 10^5 years after release from the engineered barrier system (CRWMS M&O, 2000e, Section 3.11.5).

For three-dimensional simulations at the mountain-scale, DOE considered three infiltration cases for each of three different climate scenarios for a total of nine settings (CRWMS M&O, 2000e, Section 3.11.6). The model includes the formation of perched water bodies at zones of low permeability associated with unfractured zeolite within the Calico Hills nonwelded. Compared with the high-infiltration case, the calculated arrival time for radionuclides at the water table for the low-infiltration case was from one to two orders of magnitude greater than its basecase. In addition, the 50-percent breakthrough did not occur within 10^4 year for any of the simulated radionuclides for the low-infiltration case.

At the intermediate scale of a single drift, DOE used a two-dimensional, dual-permeability model to investigate radionuclide transport to a depth of about 45 m [148 ft] below the potential repository (Bechtel SAIC Company, LLC, 2001a, Section 11.3). In these drift-scale simulations, estimated saturation levels in the fractures below the potential repository remain low because of the effects of seepage diversion resulting in a drift shadow beneath the potential repository (Bechtel SAIC Company, LLC, 2001a, Section 11.3). Radionuclide release from the engineered barrier system will, therefore, be into the tuff matrix where the flow rate is slower, and sorption processes will operate to retard radionuclide transport away from the drift. In contrast, the drift shadow is not incorporated in the total system performance assessment model abstraction, and any release from the engineered barrier system is input directly into the fracture flow system (CRWMS M&O, 2000e, Section 3.7.2; DOE, 2001a, Section 4.2.8.3.4). Because there is no retardation in the fractures, this approach is conservative relative to waste isolation. In the simulations of the potential effects of the drift shadow, transport is permanently in the matrix, and transport times over a distance of 45 m [148 ft] are increased by more than three orders of magnitude relative to the total system performance assessment abstraction where mass released from the engineered barrier system is directly input into the fracture network (Bechtel SAIC Company, LLC, 2001a, Section 11.3).

For site recommendation, the DOE total system performance assessment abstraction of radionuclide transport through the unsaturated zone (CRWMS M&O, 2000d) used a residence-time transfer function adapted to the FEHM particle-tracking algorithm (Zyvoloski, et al., 1997). The residence-time transfer function 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. After spending a randomly assigned residence-time in any given model cell, a particle moves from the resident cell to an adjoining cell. The probability of entering an adjoining cell is 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 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. In implementing the active-fracture model in the DOE total system performance assessment, however, matrix diffusion is modeled as retarded fracture transport rather than as transport into the matrix (CRWMS M&O, 2000b, Section 3.7.1.2).

The DOE total system performance assessment incorporates 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. The effect of using 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 welded 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 the 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. For example, it is not clear how fracture spacing, fracture porosity, and mean fracture aperture values (CRWMS M&O, 2000d, Table 3) are derived. The mean fracture aperture values seem large, but there is no discussion how these values 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. Also, it is not clear how or if 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, however, calculation of velocity is not discussed in CRWMS M&O (2000d). A sensitivity analysis using the mountain-scale, three-dimensional process model to examine the effects of fracture aperture on repository performance indicates there is only an impact with present-day infiltration conditions. For higher infiltration conditions associated with a wetter glacial-transition climate, however, the effects of fracture aperture relative to the basecase are subsequently smaller (CRWMS M&O 2000h). If the changes to fracture apertures are limited to fault zones alone, the impacts on flow and transport through the unsaturated zone are negligible. DOE agreed (Reamer, 2001a) to provide independent lines of evidence to support the use of the active-fracture model continuum concept in the transport model.

DOE represents all retardation processes using a linear sorption coefficient (K_d) (CRWMS M&O, 2000b,c,e,f). A lumped parameter, such as K_d, does not allow explicit consideration of different processes that might affect radionuclide sorption and retardation; care must be taken to ensure that the validity of the approach is not overextended. Although transport of radionuclide mass is distributed between colloids and dissolved components in the total system performance assessment model abstraction, aqueous speciation and other geochemical effects on sorption are considered indirectly through a K_d probability distribution function. These functions are developed for each radioelement for each of three different rock types: devitrified, vitric, and zeolitic tuffs. Retardation by sorption 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. Increased matrix flow allows increased access to the sorbing minerals, and, hence, radionuclide transport in the unsaturated zone is significantly retarded. DOE agreed to provide the technical basis for its transport parameter distributions (Reamer, 2000) and has provided an update in Bechtel SAIC Company, LLC (2003c, Attachments I and II). DOE has also agreed that where expert elicitation is used, the methodologies will be demonstrated to be consistent with guidance in NUREG–1563 (NRC, 1996).

The DOE model abstraction of radionuclide transport through the unsaturated zone uses a particle tracking method to account for transport of radionuclides that are either reversibly or irreversibly bound to colloids (Bechtel SAIC Company, LLC, 2003a, Section 5.7). Radiocolloid mass is input into the unsaturated zone from the engineered barrier system, and radiocolloids are allowed to form from the reversible sorption of dissolved radionuclides onto natural ground water colloids present in the system. Colloids with irreversibly attached radionuclides may be transported through both matrix and fracture, though diffusive matrix-fracture interaction is neglected (Bechtel SAIC Company, LLC, 2003a, Section 5.7.2). Colloids in matrix are permanently filtered by size exclusion at matrix unit interfaces. Colloids in fractures are split into two fractions, one traveling unretarded and one retarded; the unretarded proportion is less than 1 percent of the irreversible colloid mass (Bechtel SAIC Company, LLC, 2003a, Section 5.7.2; 2003b, Section 6.5.3). This new approach is more realistic, but less conservative, than the previous DOE total system performance assessment abstraction, in which all unsaturated zone fracture colloids were unretarded (CRWMS M&O, 2000f). Reversibly sorbed radionuclides are allowed to desorb from colloids and resorb onto immobile host rock in an equilibrium fashion.

DOE identifies radionuclides for the total system performance assessment model abstraction of colloidal transport (CRWMS M&O, 2000i) based on contribution to dose, inventory, and mobility considerations. Plutonium, americium, thorium, cesium, and protactinium were selected for reversible sorption onto colloids, whereas plutonium and americium were the only radionuclides selected for irreversible sorption. Because of estimated high solubility and low sorption for the geochemical conditions expected at Yucca Mountain, neptunium and uranium were judged to be relatively insensitive to colloid transport, while strontium was eliminated because of the short half life of strontium-90 (Bechtel SAIC Company, LLC, 2003b, Section 6.3.3).

DOE screened the occurrence of far-field nuclear criticality in either the unsaturated or saturated zones from its total system performance assessment based on low probability of occurrence within 10,000 years (CRWMS M&O, 2000j). This low probability is based on no waste package failures before 10,000 years; no fissile material is released; and there is no accumulation before 10,000 years through radionuclide transport in either the unsaturated or saturated zones. The DOE screening arguments are discussed in Section 5.1.2.2 of this report.

For site recommendation, DOE used arguments based on low probability, low consequence, or both to exclude numerous features, events, and processes from the total system performance assessment abstraction of radionuclide transport in the unsaturated zone. The screening arguments are outlined in CRWMS M&O (2000e), and the features, events, and processes are reported in CRWMS M&O (2001, 2000k). Scenario analysis and the NRC assessment of the DOE screening arguments are provided in Section 5.1.2 of this report. In a number of cases, the screening arguments are appropriate for exclusion of a particular feature, event, and process. In other cases, however, the DOE argument is incomplete at this time. Also, in some cases, DOE has not identified a feature, event, or process as either included or excluded. DOE agreed (Reamer, 2001b) to address concerns relating to the technical basis for its screening of features, events, and processes. In some cases, DOE assumes the transport parameter distributions used in the total system performance assessment are adequate to bound the potential effects of a given feature, event, or process on radionuclide transport in the unsaturated zone (CRWMS M&O, 2000b). DOE agreed to provide the technical basis for its transport parameter distributions (Reamer, 2000) and has provided an update in Bechtel SAIC Company, LLC (2003c, Attachments I and II).

DOE has considered the effects of thermally driven coupled processes on radionuclide transport in the unsaturated zone (Bechtel SAIC Company, LLC, 2001a, Section 11.3.5). The predominant effect is drying of the rock matrix and fractures in response to thermal loading from the potential repository. This low saturation precludes the release of any radionuclides from the engineered barrier system to the potential repository until saturation levels begin to rise. Even after the saturation levels begin to rise, the fractures in the drift shadow zone beneath the potential repository will remain comparatively dry. Transport will be predominantly through the matrix, where sorption processes will retard radionuclide migration. In the DOE sensitivity analyses, thermal-hydrological-mechanical-chemical coupled effects resulted in small increases in matrix porosity and permeability through dissolution in the zeolitized Calico Hills tuff. These increases would enhance flow and transport through the porous matrix, increasing retardation and slowing radionuclide transport. DOE concluded the effects from thermal-hydrological-mechanical-chemical coupled processes or matrix/fracture porosity and permeability would be of a magnitude similar to the existing natural variability and did not modify further the parameter distributions (Bechtel SAIC Company, LLC, 2001a, Section 11.3.5.4). DOE did not consider thermal-chemical effects on sorption coefficients or the possibility of an alkaline plume from the engineered barrier system to affect transport.

In summary, DOE appears to have a technical basis that addresses (or will address) the questions posed in the beginning of this section. DOE has used several different computer models to simulate radionuclide transport in the unsaturated zone in two and three dimensions using process models on particle tracking methods. These simulations have been conducted at different scales ranging from the drift-scale to the mountain-scale. Based on the information flows outlined by DOE, it is expected the DOE model abstraction will take information from and be consistent with seepage and infiltration models and unsaturated flow paths used in other parts of the total system performance assessment. In addition, DOE uses colloidal and dissolved radionuclide masses transported through the invert to provide the input for radionuclide transport in the unsaturated zone. DOE uses a model with an active-fracture mechanism to simulate diffusion from fracture flow to the tuff matrix. Retardation is handled in the DOE model abstraction using a lumped K_d approach, and chemistry effects on radionuclide transport are considered through parameter distributions, based on a combination of laboratory measurements and process modeling. DOE previously agreed (Reamer, 2000) to provide the technical basis supporting its flow and transport models. DOE currently plans to provide the information to NRC in a technical basis document, but the document was not available at the time of this assessment.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.7.5), is sufficient to expect that the information necessary to assess radionuclide transport in the unsaturated zone with respect to system description and model integration will be available at the time of a potential license application.

5.1.3.7.4.2 Data and Model Justification

Geochemical data used to support the flow field below the potential repository are sparse. Uncertainty about fracture and pore water compositions (Yang, et al., 1998, 1996; Browning, et al., 2000) results from limited data sets and questions regarding DOE attempts to account for the effects of extraction techniques on water chemistry. Questions exist regarding the 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 the occurrence of Cl-36 (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. Results of the study suggest active fractures are much more abundant than fractures associated with bomb-pulse Cl-36. In contrast, pneumatic monitoring evidence suggests the fracture system is well connected and can be viewed as a continuum. These types of uncertainties need to be resolved for the radionuclide transport in the unsaturated zone model abstraction. DOE agreed (Reamer, 2000) 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 pathways elsewhere in the tuffs. The DOE transport parameters are assigned by rock type only and do not include specific considerations of faults, unless the features are treated explicitly as zones of fracture flow. It is not clear that DOE accounted for the possible effects of faulting in formulating transport parameter distributions (CRWMS M&O, 2000c,l). DOE agreed (Reamer, 2000) to provide a technical basis for the importance to performance of transport through fault zones below the potential repository and also the technical basis for the parameters and distributions if such transport is found to be important to performance.

Data to support the initiation of dissolved radionuclide sources in fracture versus matrix modeling continua have not been made available by DOE. For the site recommendation, DOE conservatively assumed dissolved radionuclide sources entered directly into the fracture domain. If the DOE radionuclide transport abstraction is modified to consider dissolved radionuclide sources that initiate in the matrix continuum and then travel by advection or diffusion through the matrix before entering the more rapidly flowing fracture domain, information to justify such modifications must be provided. Documentation for the flow and transport abstraction for the potential license application was not available at the time of this assessment.

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 potential repository. These properties include fracture aperture, fracture porosity, spacing between flowing intervals, linear ground water velocity within the fracture, porosity of the rock matrix, sorption coefficients (K_d values), and the effective matrix diffusion coefficient. Comprehensive data sets (Flint, 1998; Triay, et al., 1997; Bechtel SAIC Company, LLC, 2003c) that include experimental, field, and process modeling support are used to support the estimates of hydrologic and transport properties of the rock matrix.

Data to support the conceptual model of diffusive solute transfer between fracture and matrix continua are supported by laboratory and field tests (CRWMS M&O, 2000c). 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. Efforts to collect 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 (Schlueter, 2000). 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 tests and is expected to incorporate the results, as appropriate, in the total system performance assessment abstraction (Reamer, 2000), but the results were not available at the time of this assessment.

The DOE abstraction for radionuclide transport in the unsaturated zone is based on a conceptual model that assumes radionuclide sorption occurs only within the rock matrix and that solutes can migrate by diffusion from flowing fractures into the surrounding rock by matrix diffusion. 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 formation proposed for possible construction of a repository. DOE agreed (Reamer, 2001a) 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 potential repository host rock, but the results were not available at the time of this assessment.

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. 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 (CRWMS M&O, 2000e) or in supporting 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 the observed *in-situ* fracture patterns. To address these shortcomings, DOE agreed (Reamer, 2000) results and analyses of ongoing seepage and transport studies in the Alcove 8–Niche 3 tests will include fracture information, but this information was not available for this assessment.

Earlier DOE total system performance assessment abstractions of radionuclide transport in the unsaturated zone relied on informal expert elicitation (Barnard, et al., 1992; Wilson, et al., 1994; CRWMS M&O, 2000c) for determining the K_d distributions. The elicitation methods used to arrive at the K_d probability distribution functions are described in general terms in Barnard, et al. (1992), however, many methods normally used in expert elicitation (e.g., panel selection, training, mitigating bias, consensus building, incorporating dissenting opinions, aggregation of results, and documentation) were not discussed. A recent update (Bechtel SAIC Company, LLC, 2003c, Attachments I and II) provided a more systematic technical basis for the selection of K_d distributions for americium, cesium, neptunium, plutonium, protactinium, radium, strontium, thorium, and uranium. The K_d distributions are based on experimental data from the DOE program, and the effects of variability in geochemistry and mineral surface area are characterized for the long-lived actinides using a surface complexation modeling approach (Bechtel SAIC Company, LLC, 2003c, Attachments I and II).

The radionuclides subject to colloidal transport in the DOE total system performance assessment are identified in the inventory abstraction (CRWMS M&O, 2000i; Bechtel SAIC Company, LLC, 2003b). The selection of radionuclides is reasonably based on considerations of dose, inventory, and mobility. DOE agreed (Reamer, 2000) to document how radionuclides were identified for colloidal transport in the total system performance assessment, and provided the updated information in Bechtel SAIC Company, LLC (2003a,b). In its analysis of colloidal transport, DOE included plutonium, americium, thorium, protactinium, and cesium for the reversible model. These radionuclides are potentially major contributors to the inventory at 10,000 years. For the irreversible model, only plutonium and americium are included. DOE did not include uranium and neptunium in the colloidal transport abstraction because they are highly soluble and weakly sorbing radionuclides for the conditions expected at Yucca Mountain (Section 5.1.3.4).

Stability of colloids within the drift is determined based on the stochastic sampling of in-drift chemical properties, including pH and ionic strength (Bechtel SAIC Company, LLC, 2003b, Section 6.5). The DOE model results indicate that most colloids leaving the waste package will be unstable because of high temperature and high ionic strength in the waste package. In addition to forming colloids, a significant portion of the iron corrosion products are also assumed to be immobile. In the total system performance assessment abstraction of colloidal transport, the sum of all in-drift colloid forms (embedded wasteform, reversibly sorbed, and irreversibly sorbed) and net dissolved radionuclides transported through the invert at the base of the drift is the radionuclide mass passed to the unsaturated zone for flow and transport. For estimates of the formation of reversibly sorbed colloids, ground water colloid concentrations in the unsaturated zone are based on field measurements from the saturated zone in the Yucca Mountain area and at the Idaho National Engineering and Environmental Laboratory. Wasteform colloid concentrations are based on long-term simulations performed at Argonne National Laboratory, and colloids from iron corrosion are based on small-scale studies performed at the University of Nevada at Las Vegas (Bechtel SAIC Company, LLC, 2003a,b). Details necessary for full evaluation of these field and laboratory results were not available at the time of this assessment. In addition, it is not clear how the recent University of Nevada at Las Vegas iron corrosion product colloid results, as well as newly cited literature data, were reconciled with the large pH-ionic strength colloid instability zone in the abstraction for calculating colloid concentrations (Bechtel SAIC Company, LLC, 2003b, Sections 6.3.1.3 and 6.3.2.3).

Once the colloid mass is passed as a source term to the unsaturated zone, DOE uses the particle tracking code FEHM (Zyvoloski, et al., 1997) to simulate the transport of colloids by advection in the fracture system. As a conservative assumption, diffusive transport of colloids is considered negligible (Bechtel SAIC Company, LLC, 2003b, Section 6.5.3). This assumption is probably reasonable given that colloids are likely to have free diffusion coefficients that are two to four orders of magnitude less than those for dissolved solutes. Colloid sorption at the air-water interface also is neglected in the model abstraction (Bechtel SAIC Company, LLC, 2003b, Section 5.7). In the total system performance assessment model abstraction of the natural barrier system (i.e., the unsaturated and saturated zones combined), radionuclides are considered as either reversibly sorbed or irreversibly sorbed on colloidal particles. The radionuclides that are reversibly sorbed to the colloid phase (1 to 10 percent of the total colloid mass) are permitted to desorb and resorb to immobile matrix minerals as determined by a sampled K_d distribution. The irreversibly sorbed colloids (90 to 99 percent of the colloid mass) only include plutonium and americium transport. The irreversibly sorbed colloids are divided into a "fast" component that travels unretarded through the fracture network to the water table and a "slow" component that is subjected to retardation. The retardation factors used for the slow irreversible colloids are based on experiments conducted with microspheres under hydrologically saturated conditions at the C–Wells Complex (Bechtel SAIC Company, LLC, 2003a, Section 5.7). The fast irreversible colloids are the most significant contributor from colloidal transport, with breakthrough at the water table in 100 years or less (Bechtel SAIC Company, LLC, 2003a, Section 5.7.3). Although slower than the fast irreversible component, the slow irreversible colloids arrived at the water table quicker than the dissolved species. The impact of the retardation factor for slow irreversible colloid transport on repository performance has not yet been evaluated (Bechtel SAIC Company, LLC, 2003a, Section 5.7.3).

It is important to note that DOE has provided no site-specific supporting data for the colloidal transport of radionuclides through the unsaturated zone, and those parameter distributions used to simulate colloid transport and retardation are based on tests conducted for hydrologically saturated conditions at the C–Wells Complex (Bechtel SAIC Company, LLC, 2003b, Section 6.5.3; 2003d, Section 6.6). DOE asserts that this treatment is conservative. This assertion is based on limited information that colloids preferentially attach to the air-water interface. If this interface is immobile, then colloids will be retained. Also, if present, higher ionic strength solutions in the pores in the matrix will lead to colloid instability and reduced colloidal transport.

In summary, DOE appears to have a technical basis that addresses (or will address) the questions posed in the beginning of this section. DOE abstraction of radionuclide transport in the unsaturated zone is based on a hydrologic flow model that is consistent with the model abstractions of flow paths in the unsaturated zone. Site characterization information on geochemistry and mineralogy is used to establish the physical–chemical framework for radionuclide transport through the unsaturated zone. The degree to which radionuclide transport occurs in the matrix may have a significant effect on waste isolation, however, data to support the initiation of dissolved radionuclide sources in fracture versus matrix modeling continua are not yet available. DOE uses a mixture of laboratory and field data with process modeling to provide a technical basis for parameters that describe the transport of dissolved radionuclides, but relies on information from tests conducted under hydrologically saturated conditions to support the abstraction of colloid transport and retardation. DOE uses expert judgment to establish parameter distributions for various parameters, however, the transparency of the judgment process is not sufficient to allow a reviewer to trace the origins of the judgments (NRC, 1996). DOE agreed previously (Reamer, 2000) to provide the documentation explaining the technical basis used to support the DOE process. Currently, DOE is planning to issue a technical basis document that provides a summary updating the information used to support its model abstraction for flow and transport through the unsaturated zone, but the report was not available at the time of this assessment.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.7.5), is sufficient to expect that the information necessary to assess radionuclide transport in the unsaturated zone with respect to data being and model justification will be available at the time of a potential license application.

5.1.3.7.4.3 Data Uncertainty

DOE uses stochastic approaches to identify and constrain data uncertainty in its model abstraction of radionuclide transport in the unsaturated zone (CRWMS M&O, 2000b,f). The data uncertainty is represented in the performance assessment model abstraction by using distributions to place bounds on parameter variability. During performance assessment calculations, the distribution is sampled in multiple realizations that are used to generate statistics of the estimated dose to the receptor. Depending on the parameter, the distributions may represent natural variability or areas where the available site characterization data are sparse. Uncertainty in those parameters related to matrix diffusion, sorption in the matrix, and colloidal transport are the most important for waste isolation (Appendix D).

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. 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]. 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 appear reasonable, 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 upper bounds for model-scale diffusion coefficients.

Another important uncertainty is the effective fracture aperture used in the total system performance assessment abstraction of unsaturated zone radionuclide transport. As discussed in 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 lognormal distribution of apertures for all the model layers beneath the potential repository (values are listed in CRWMS M&O, 2000d, Table 4).

According to 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 the fracture-matrix connection area. It is not clear how the active-fracture concept is factored into estimates of the fracture-matrix connection area. The mean fracture aperture values appear large, and there is no discussion how these values 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; thus, a constant value for each layer is used (CRWMS M&O, 2000e, Section 3.11.3.4). DOE agreed (Reamer, 2001a) to provide independent lines of evidence to support the use of the active-fracture model continuum concept in the transport model. At the time of this assessment, DOE has not provided that information.

Retardation in the CHnv has been assigned medium significance to repository performance (Appendix D). The K_d distributions used in previous total system performance assessment abstractions of radionuclide transport in the unsaturated zone (CRWMS M&O, 2000b,c,l) were based on expert elicitation (or expert judgment) (Barnard, et al., 1992; Wilson, et al., 1994; Triay, et al., 1997). Bechtel SAIC Company, LLC (2003c, Attachments I and II) includes a significant revision to the technical basis for the K_d distributions for americium, cesium, neptunium, plutonium, protactinium, radium, strontium, thorium, and uranium. Together with the nonsorbing (i.e., $K_d = 0$) radionuclides technetium, iodine, and carbon, these represent the most critical radionuclides for repository performance (Bechtel SAIC Company, LLC, 2003c, Attachments I and II; Mohanty, et al., 2002). The K_d distributions are based on experimental data from the DOE program using crushed tuffs and water from Wells J–13 and UE–25p#1.

The sorption parameter ranges for the actinides (americium, neptunium, plutonium, thorium, and uranium) are also supported by surface complexation modeling using the computer code PHREEQC (Parkhurst and Appelo, 1999). This process modeling is calibrated against experimental data external to the DOE program and is used to investigate the effects of observed variability in geochemistry and mineralogy. Process modeling has not been used to support the parameter distributions for cesium, protactinium, radium, and strontium. The use of process modeling to extend the limited chemical conditions considered in the batch experiments with crushed tuff has provided a stronger technical basis for the upper and lower limits, and the upper limit is conservative (less than) the observed sorption values.

Although the upper and lower limits of the K_d cumulative distributions are based on experimental data supported by process modeling, the shapes of the distributions are assigned through expert judgment. DOE investigated the significance of uncertainty in sorption in the unsaturated zone using a series of bounding analyses for mildly sorbing radionuclides, such as neptunium, and strongly sorbing radionuclides, such as plutonium (Bechtel SAIC Company, LLC, 2003c, Sections 6.9 and 6.10). In the case of neptunium transport, the uncertainties in the DOE K_d distributions result in a decrease in breakthrough time by one to two orders of magnitude for a given mass fraction release at the water table. For strongly sorbing plutonium, assuming no retardation in the unsaturated zone increases the mass fraction release at the water table by one order of magnitude. Because the uncertainty in sorption parameters has a potentially strong effect on transport through the unsaturated zone, documentation of the judgment to establish the K_d distributions 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 (NRC, 1996). DOE agreed (Reamer, 2000) to provide the documentation explaining the technical basis used to support the DOE process. This information has not been provided at the time of this assessment.

DOE has improved its capability to model unsaturated zone colloid transport in total system performance assessment (Bechtel SAIC Company, LLC, 2003a,b), however, limited site-specific information supports the parameters. DOE addressed this limitation by using parameter values based on tests conducted under hydrologically saturated conditions at the C–Wells Complex (Bechtel SAIC Company, LLC, 2003b, Section 6.5.3). DOE asserts this approach is conservative, given the potential role of an immobile air-water interface in the unsaturated zone reducing colloidal transport and the higher ionic strength solutions present in pores in the rock matrix. There are no available radioelement-specific data to determine if the uncertainty in colloid transport has been constrained in the radionuclide transport in the unsaturated zone model abstraction, however, DOE is addressing this data limitation through the use of bounding analyses and sensitivity analyses.

No site characterization data are available to support transport parameters for unsaturated zone colloid transport in the total system performance assessment, so DOE uses analogous data from hydrologically saturated systems. Uncertainty is reflected in parameter distributions adopted in total system performance assessment. The four parameters that affect unsaturated zone colloid transport are colloid size distribution, colloid K_c, colloid retardation R_c, and colloid matrix filtration factor; colloid matrix diffusion is neglected (CRWMS M&O, 2000f). In the unsaturated zone model abstraction, R_c is applied to irreversible slow colloids only. R_c is based on a sampled cumulative distribution developed from tests conducted under hydrologically saturated conditions at the C–Wells Complex (Bechtel SAIC Company, LLC, 2003b, Section 6.5.3). DOE asserts this treatment is conservative, given the potential role of an immobile air-water interface in the unsaturated zone reducing colloidal transport and the higher ionic strength solutions present in pores in the rock matrix. Matrix filtration factors are treated using a single value based on nonsite-specific theory and tests taken from the literature, but these factors are allowed to vary from unit to unit (Bechtel SAIC Company, LLC, 2003a, Section 5.5.1). The colloid size distribution is used for calculating removal by filtration at matrix unit interfaces; it is not based on site-specific data, but was chosen to be consistent with analogous laboratory data (CRWMS M&O, 2000f). Sensitivity studies suggest filtration is sensitive to colloid size, with smaller particles more likely to enter the matrix and be filtered; the affect on repository performance is small, however (Bechtel SAIC Company, LLC, 2003a, Section 5.5.1). In the radionuclide transport process model, a significant portion of the colloid mass is predicted to be retained at the contact between the Topopah Spring and zeolitized Calico Hills units and at the water table because of decreases in porosity and permeability (Bechtel SAIC Company, LLC, 2003a, Section 5.8).

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 (Bechtel SAIC Company, LLC, 2003a,b). The K_d values are represented by probability distributions for sorption onto smectite and iron oxyhydroxides (Bechtel SAIC Company, LLC, 2003b, Section 6.3.3.1). These distributions are determined separately from the K_d distributions for the rock matrix. They are supported by the DOE experimental data and also by experimental data and process modeling studies external to the DOE program (EPA, 1999; Honeyman and Ranville, 2002). Calculation of K_c also involves a term for colloid concentration in the water. The concentration of wasteform colloids is determined from the in-drift colloid concentration model abstraction (Bechtel SAIC Company, LLC, 2003b), while natural colloid concentrations, ranging from 0.001 to 200 ppm, are based on concentrations measured in wells from the Yucca Mountain vicinity (Bechtel SAIC Company, LLC, 2003a, Appendix B). The uncertainty in ground water colloid concentrations in the unsaturated zone is represented using a cumulative distribution function based on field measurements from the saturated zone in the Yucca Mountain area and at the Idaho National Engineering and Environmental Laboratory. Wasteform colloid concentrations are based on long-term simulations performed at Argonne National Laboratory, and colloids from iron corrosion are based on small-scale studies performed at the University of Nevada at Las Vegas (Bechtel SAIC Company, LLC, 2003a,b).

Because the sorption coefficient and the colloid concentration are sampled from parameter distributions, the K_c parameter is sampled also. Because of the potential for sorption onto the immobile rock matrix, in the DOE model repository performance is not sensitive to the parameters that control reversibly sorbed colloids, except at the highest ranges of the K_c parameters (high sorption coefficient, high colloid concentration).

In summary, DOE uses stochastic approaches to identify and constrain data uncertainty in its model abstraction on radionuclide transport in the unsaturated zone. Uncertainties represented by the parameter distributions are based on a combination of laboratory and field data, supported by process modeling. In various cases, however, the technical basis for the parameter distributions used to describe data uncertainty is not 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 judgments used to provide uncertainty estimates in accordance with NRC (1996) guidance and its own quality assurance program. DOE agreed previously (Reamer, 2000) to provide technical support demonstrating appropriate handling of data uncertainty, including sensitivity analysis. Currently, DOE is planning to issue a technical basis document that provides a summary updating the information used to support its model abstraction for flow and transport through the unsaturated zone, but the report was not available at the time of this assessment.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.7.5), is sufficient to expect that the information necessary to assess radionuclide transport in the unsaturated zone with respect to data uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.7.4.4 Model Uncertainty

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 a dual continuum particle-tracking model, were performed (CRWMS M&O, 2000m, Section 6.4.3). The two particle-tracking routines agree only if diffusion and dispersion are neglected. For cases that include diffusion and dispersion, the median breakthrough calculated with the FEHM transfer algorithm (Zyvoloski, et al., 1997) occurs at times more than one or two orders of magnitude earlier (DOE, 2001a, Section 4.2.8.3.5.2). The difference is more pronounced for radionuclides undergoing sorption in the matrix. DOE asserts that these differences stem from different implementations of the diffusive mass flow between fractures and the matrix in the two codes (CRWMS M&O, 2000m, Section 7). The difference between the predictive results of the two models is potentially significant. The finite-element heat and mass transfer model used for total system performance assessment predicts faster breakthrough.

In developing total system performance assessment model abstractions for radionuclide transport in the unsaturated zone, DOE has conservatively neglected radionuclide sorption in fractures and applied a linear sorption coefficient to simulate radionuclide transport through the rock matrix (Bechtel SAIC Company, LLC, 2001a, 2003c; DOE, 2001a,b). The K_d approach is a lumped parameter approach that does not explicitly take into account processes or spatial and temporal variabilities that may affect radionuclide sorption. Parameter distributions are based on experimental batch sorption data using water from Wells J–13 and UE–25 p#1. DOE asserts uncertainty because of geochemical processes and variability in mineralogy and water chemistry is contained within the probability distributions defined for K_d (CRWMS M&O, 2000c; DOE, 2001a, Section 4.2.8.4). Recently, DOE used surface complexation modeling to investigate the effects of geochemistry and mineral surface area and provide constraints on the parameter distributions (Bechtel SAIC Company, LLC, 2003c, Attachments I and II). The variability in geochemical conditions is appropriate (e.g., pH from 6 to 9), and the derived K_d distributions are bounded by the modeling results. Spatial variability also is indirectly addressed at the mountain-scale by using a three-dimensional model that incorporates changes in hydrologic flow caused by hydrostratigraphy. In addition, temporal variability is indirectly addressed by using different unsaturated zone flow fields for different climate/infiltration states. Transport parameters are held constant for each realization, however, and not allowed to change with time (DOE, 2001a, Section 4.2.8.4.5). *In-situ* testing planned for Alcove 8–Niche 3 and Busted Butte is anticipated to support the characterization of model uncertainty. Laboratory column experiments and block tests also will help evaluate the uncertainty in using a linear sorption coefficient, but these results were not available at the time of this assessment.

For unsaturated zone colloid transport modeling, DOE addresses model uncertainty chiefly by adopting each of two distinct attachment modes—reversible and irreversible (Bechtel SAIC Company, LLC, 2003a). DOE used a limited amount of site-specific information to develop parameter distributions that reflect the uncertainty of the colloid transport parameter. The colloidal transport model provides sensitivity studies that suggest colloid transport through the unsaturated zone is significant only for fast irreversible colloids that are not allowed to be retarded during fracture transport. The portion of the colloid mass assigned as fast irreversible colloids is not supported by site characterization data, however, and there is no objective evidence the assigned values are bounding. Sensitivity analyses in, and cited in, available reports (Bechtel SAIC Company, LLC, 2003a–e) do not quantitatively address the barrier performance effect of this new assumption that greater than 99 percent of colloids with irreversibly attached radionuclides are retarded in the unsaturated zone. In addition, some of the colloid model parameter distributions, such as fracture retardation and ground water colloid concentration, are developed from field experiments under hydrologically saturated conditions; the evidence used to support this assumption is qualitative (Bechtel SAIC Company, LLC, 2003a,b). In general, DOE does not make clear that its sensitivity analyses and parameter uncertainty distributions yield a high degree of confidence that the effect of unsaturated zone colloidal transport on repository performance has been bounded by the models.

In summary, DOE appears to have a technical basis that addresses (or will address) the questions posed in the beginning of this section. DOE has applied alternative process models and particle-tracking methods in two and three dimensions to simulate radionuclide transport in the unsaturated zone. Depending on how the models implement transport processes such as diffusion and dispersion, the calculated breakthrough may be significantly different. In cases such as colloid transport under unsaturated conditions, where the mode of transport is not well understood, DOE uses sensitivity analyses and bounding analysis. Model approaches used in the DOE total system performance assessment model abstraction provide for quicker breakthrough (i.e., are conservative) than the alternative models tested. NRC performance assessments, however, use more conservative assumptions than the DOE models, and the results show the delay in radionuclide transport through the unsaturated zone contributes less to waste isolation. DOE agreed previously (Reamer, 2001a) to demonstrate adequate consideration of model uncertainty. Currently, DOE is planning to issue a technical basis document that provides a summary updating the information used to support its model abstraction for flow and transport through the unsaturated zone, but the report was not available at the time of this assessment.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.7.5), is sufficient to expect that the information necessary to assess radionuclide transport in the unsaturated zone with respect to model uncertainty being characterized and propagated through the model abstraction will be available at the time of a potential license application.

5.1.3.7.4.5 Model Support

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 with predictions from analytical solutions and other numerical models (CRWMS M&O, 2000d,m). For cases where large numbers of particles are used, predictions 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 (registered trademark of Golder Associates Inc.) (GoldSim Technology Group, 2004), FEHM transfer (Zyvoloski, et al., 1997), 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 the finite-element heat and mass transfer unsaturated zone outflow mass flux curves trace the corresponding engineered barrier system release curves well. The results also provide support 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).

DOE developed information on a number of natural analogs to provide qualitative comparisons for model confidence building at the field scale (CRWMS M&O 2000e, Section 3.11, 2000n, Section 6.5.2; DOE, 2001a, Section 4.2.8.2.3; Bechtel SAIC Company, LLC, 2003a, Section 5.6). These natural analogs include uranium mines at Peña Blanca in Mexico and Cigar Lake in Canada and an archaeological site at Akrotiri, Greece. The model abstractions are not applied to these analog sites, but general observations of transport behavior are used to support the conceptual models. For example, uranium distribution at Peña Blanca is limited to short-lateral distances and restricted to fractures (CRWMS M&O, 2000n, Section 6.5.2.1). The Peña Blanca and Akrotiri sites both are in unsaturated volcanic tuffs. This qualitative comparison suggests that radionuclide transport is likely to be limited in the unsaturated zone at Yucca Mountain. DOE has undertaken a drilling program at Peña Blanca that it has stated should provide more detailed information for a more quantitative comparison with radionuclide transport at Yucca Mountain.

Field sites at Busted Butte south of Yucca Mountain and alcove tracer tests in the Exploratory Studies Facility have been used to provide limited quantitative evaluations of the radionuclide transport model abstraction. Because of environmental considerations, chemical homologues such as nickel, cobalt, and manganese have been used instead of radionuclides in these tracer tests. For example, tracer tests during Phase 1b at Busted Butte suggested laboratory-derived K_d values overpredict the transport distances of lithium through the unsaturated zone (CRWMS M&O, 2000e, Section 3.11.11.2). Problems with microsphere experiments at Busted Butte have limited the amount of independent information for colloid transport through the unsaturated zone.

In summary, DOE appears to have a technical basis that addresses (or will address) the questions posed in the beginning of this section. DOE provided support for its total system performance assessment model abstraction of radionuclide transport in the unsaturated zone through the use of alternative computer models, field tests, and natural analogs. Computer models are used for quantitative comparison at different scales. Results suggest the DOE total system performance assessment model abstractions are consistent with or bound transport predictions from more detailed two- and three-dimensional process models. Comparisons with field sites and natural analogs provide qualitative confidence building, but generally do not provide quantitative demonstration that results from laboratory sorption and transport experiments can be extended or used to bound transport over larger distances and longer times. If credit is to be taken for radionuclide attenuation, DOE should demonstrate that nonradioactive tracers used in field tests are appropriate homologues for radioelements. Alcove tests in the Exploratory Studies Facility, Busted Butte, and large block studies at Atomic Energy of Canada Limited Laboratories in Pinawa, Manitoba, provide limited transport data using a suite of tracers representative of conservative and weakly sorbing radionuclides (Vandergraaf, et al., 2000a,b). DOE considers these tests representative of transport of conservative radionuclides, sorbing radionuclides, and colloids. Natural analog studies are ongoing at Peña Blanca that may provide information suitable for testing transport models. For dissolved radionuclides, DOE is using these results as a means to demonstrate the appropriateness of conceptual models rather than as a source of transport parameters for total system performance assessment. DOE agreed (Reamer, 2000) to provide pretest predictions and results of field tests to demonstrate model abstraction is supported by objective comparisons.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.7.5), is sufficient to expect that the information necessary to assess radionuclide transport in the unsaturated zone with respect to model abstraction output being supported by objective comparisons will be available at the time of a potential license application.

5.1.3.7.5 Summary and Status of Key Technical Issue Subissues and Agreements

Table 5.1.3.7-1 provides the status of all key technical issue subissues, referenced in Section 5.1.3.7.2, 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 review methods described in NRC (2003) (Section 5.1.3.7.4 of this report). Note the status and the detailed agreements pertaining to all the key technical issue subissues are provided in Table 1.2-1 and Appendix A.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (e.g., specified testing or analyses), indicates that information necessary to begin a technical review will likely be available at the time of a potential license application.

Table 5.1.3.7-1. 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.1.01 through RT.1.05
	Subissue 2—Radionuclide Transport through Alluvium	Closed-Pending	RT.2.10
	Subissue 3—Radionuclide Transport through Fractured Rock	Closed-pending	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

Table 5.1.3.7-1. Related Key Technical Issue Subissues and Agreements (continued)			
Key Technical Issue	Subissue	Status	Related Agreement*
Evolution of the Near-Field Environment	Subissue 3—The Effects of Coupled Thermal-Hydrologic-Chemical Processes on Chemical Environment for Radionuclide Release	Closed-Pending	ENFE.3.05
	Subissue 4—The Effects of Coupled Thermal-Hydrologic-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 review methods.			

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5.1.3.8 Flow Paths in the Saturated Zone

5.1.3.8.1 Description of the Issue

The Flow Paths in the Saturated Zone Integrated Subissue addresses features and processes that affect the flow paths and flow velocities in the saturated zone between the area beneath the potential repository site and the compliance boundary. The relationship of this integrated subissue to other integrated subissues is depicted in Figure 5.1.3.8-1. The overall organization and identification of all the integrated subissues are depicted in Figure 1.1-2. The last complete description for the abstraction of flow paths in the saturated zone was provided by DOE in support of the site recommendation and documented in a process model report (CRWMS M&O, 2000a,b). Several supporting analysis and model reports (CRWMS M&O, 2000c–i) provided supporting documentation for the abstraction. More recently, DOE published a technical basis document (Bechtel SAIC Company, LLC, 2003a) that describes the current DOE conceptual model for saturated zone flow and radionuclide transport. Additionally, DOE submitted two new analysis and model reports (Bechtel SAIC Company, LLC, 2003b,c) that provide important supporting information for the recent saturated zone flow model abstraction. At the time of this assessment of issue resolution status, not all the supporting documentation for the most recent abstraction approach was available. Accordingly, this section documents the current NRC understanding of the DOE total system performance assessment abstraction for saturated zone flow based on a combination of new and previously reviewed information. This assessment is focused on aspects important to repository safety based on the risk insights gained to date, including those summarized in Appendix D. The scope of the assessment presented here is limited to examining if data gathered and methodology developed by DOE are likely to be adequately documented for the staff to undertake a detailed technical review. This assessment is not a regulatory compliance determination review of a potential license application.

5.1.3.8.2 Relationship to Key Technical Issue Subissues

The Flow Paths in the Saturated Zone Integrated Subissue incorporates subject matter previously described in the following 12 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)

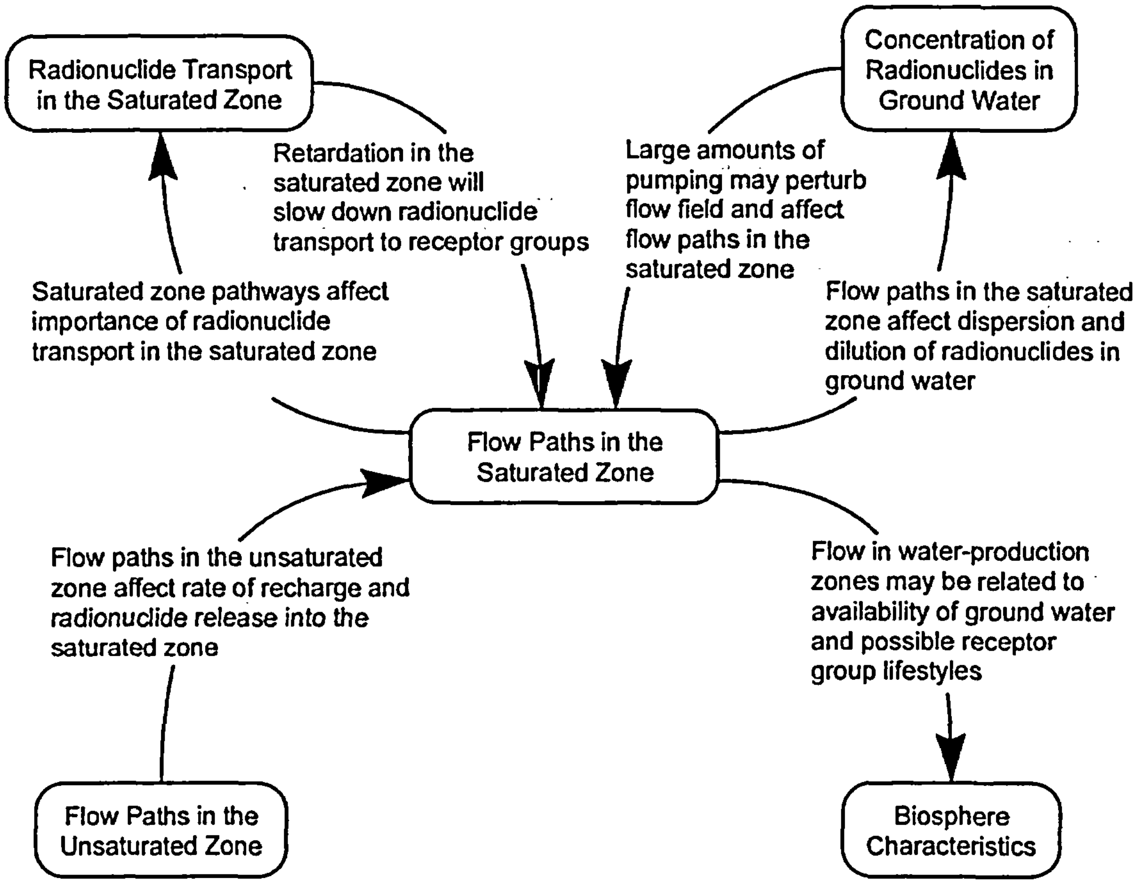

Figure 5.1.3.8-1. Diagram Illustrating the Relationship Between Flow Paths in the Saturated Zone and Other Integrated Subissues

- 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)

- 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 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 would provide to NRC. 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.

5.1.3.8.3 Importance to Postclosure Performance

One aspect of risk informing of the NRC understanding of postclosure repository performance is to determine how this integrated subissue is related to the DOE repository safety strategy. Saturated zone flow paths from Yucca Mountain to the compliance boundary comprise both fractured rock and porous alluvium. The portion of the flow path that occurs in alluvium is important because of the large capacity of the alluvium to retard a majority of the radionuclides. Sensitivity analyses using the NRC TPA Version 4.1 code (Mohanty, et al., 2002), however, indicate at least 500 m [1,640 ft] of the total 18-km [11.2-mi]-flow path must occur in alluvium to have a significant influence on retarded radionuclides. Examples of analyses used to evaluate the importance of the saturated zone to total system repository performance are provided in the following paragraphs.

Performance assessment sensitivity analyses by NRC (Mohanty, et al., 2002) using the TPA Version 4.1 code also indicate the importance of the saturated zone flow system to potential repository performance. In these analyses, the flow distance traveled in saturated alluvium was ranked among the 10 parameters that most affect dose estimates for the basecase performance scenario.

Appendix D states the velocity of water within fractured rock and porous alluvium units can be quite different because of differences in the hydrologic properties. The ground water traveltime in the saturated zone is expected to be on the order of several hundreds of years and longer. Because flow velocities in the alluvium are small relative to the fractured tuff, the majority of the traveltime occurs in the alluvium. Radionuclide transport through the alluvium also is important because of the capability of the porous media to delay a majority of radionuclides through sorption onto mineral surfaces.

Bechtel SAIC Company, LLC (2002a) presents the results of performance assessment analyses after neutralizing the barrier potential of the saturated zone. This study was conducted using the DOE basecase model for the unsaturated zone and assuming the calculated release from the unsaturated zone is discharged directly into the water usage volume at the accessible environment. The results of this analysis show almost no perceptible change in the mean annual dose for the nominal case, and approximately double the annual dose for an igneous intrusive scenario in which affected waste packages and drip shields are assumed to fail. Bechtel SAIC Company, LLC (2001) also presents conclusions from several studies on enhanced or degraded processes related to saturated flow and transport. One analysis involves uncertainty inherent in the model, and results show virtually no difference between the basecase and the studied case. Another analysis involves a comparison of the basecase model to a model using the minimum flow path length in the alluvium; results show the minimal alluvium case had approximately a 10-percent higher simulated dose. The apparently low significance of the saturated zone flow and transport system in these analyses is due in part to the fact that the saturated zone is at the downstream end of a multiple-component barrier system. That is, radionuclide releases are limited by the engineered system and attenuated by the unsaturated zone flow system, giving the appearance that the saturated zone does little to retard radionuclide migration.

Bechtel SAIC Company, LLC (2003a) indicates the process of flow and transport in the saturated zone is considered an important barrier because it affects the arrival time of radionuclides at the receptor location that potentially may be released from the potential Yucca Mountain repository. DOE identifies saturated zone flow and transport as one of eight principal model components of its total system performance assessment (Bechtel SAIC Company, LLC, 2002b; CRWMS M&O, 2000c).

Risk insights pertaining to flow paths in the saturated zone indicate that the saturated alluvium transport distance is of medium significance to waste isolation. Aspects of the flow system that affect alluvial transport distance include the prevailing hydraulic gradient, the potentially anisotropic permeability of volcanic tuff flow system, and the geometry of the tuff-alluvium contact. The details of the risk insights ranking are provided in Appendix D. The following assessment of the DOE characterization and performance assessment abstraction of saturated zone flow paths was conducted at a level of detail commensurate with the assigned degree of significance.

5.1.3.8.4 Technical Basis

NRC developed a review plan (NRC, 2003) consistent with the acceptance criteria and review methods in previous issue resolution status reports. A status assessment of the DOE approaches for including flow paths in the saturated zone in total system performance assessment abstractions is provided in the following subsections. This assessment is organized according to the five review methods identified in Section 2.3: (i) Model Integration (including system description), (ii) Data and Model Justification, (iii) Data Uncertainty, (iv) Model Uncertainty, and (v) Model Support.

5.1.3.8.4.1 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 (Bechtel SAIC Company, LLC, 2003b,c).

The site-scale flow model domain occurs within the Alkali Flat–Furnace Creek ground water basin, which is part of the larger Death Valley regional ground-water flow system. The rectangular saturated zone site-scale flow model domain is 30 km [18.7 mi] from west to east by 45 km [28.0 mi] north to south. The model domain extends vertically from the interpreted water table elevation to a fixed 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 × 1,640.4-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

(Bechtel SAIC Company, LLC, 2003b, Table 10). The model domain and grid structure used by DOE are adequate to model any potential flow path between Yucca Mountain and the compliance boundary. All simulations used in the performance assessment abstraction assume steady-state Darcy flow.

Constant-potential lateral boundary conditions are assigned to the vertical sides of the model based on an interpretation of regional water level and hydraulic head data. The constant boundary potentials vary laterally but are assumed constant with depth. The vertically constant boundary heads do not preclude the model from reproducing the observed upward hydraulic gradient observed in the central model region (Bechtel SAIC Company, LLC, 2003b).

Surface recharge is assigned to the top of the saturated zone flow model based on information from three sources, as described by CRWMS M&O (1999). First, total volumetric recharge from the approximately 50-km^2 [19.3-mi^2] area of the unsaturated zone model domain is assigned as an average recharge rate over the corresponding portion of the saturated zone flow model domain. Second, estimates of recharge from surface flows in Fortymile Wash are assigned in areas corresponding to linear reaches along the wash. Third, recharge rates estimated for the Death Valley Regional Groundwater Flow Model are applied to the northern-most portion of the site-scale model area.

The Hydrogeologic Framework Model (U.S. Geological Survey, 2001) provides the basis for assigning hydraulic properties to the numerical grid cells of the flow model. A major input data source for this framework is the Geologic Framework Model (U.S. Geological Survey, 2000). 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 stratigraphic surfaces developed for the Nevada Test Site (U.S. Geological Survey, 2001). The top of the Hydrogeologic Framework Model is truncated by an interpreted water-table surface based on borehole water elevation data (U.S. Geological Survey, 2000). The Hydrogeologic Framework Model describes the layer geometries of the 19 hydrogeologic units included in the flow model (Bechtel SAIC Company, LLC, 2003b, Table 11). Homogenous permeability values assigned to each of these units and features are obtained through the model calibration process. Large-scale heterogeneity is considered in the model by including 17 additional hydrologic features to represent faults, fault zones, and areas of mineralogical alteration (Bechtel SAIC Company, LLC, 2003b, Table 12).

The nominal case permeability assigned to each hydrogeologic unit and feature is determined by calibration, using an inverse approach to minimize differences between model calculations and calibration targets. The calibration targets include 115 water-level and head measurements (Bechtel SAIC Company, LLC, 2003b, Table 13). Weighting factors are used to assign relative importance to each calibration target. For example, a weighting factor of 20 is used for water levels in wells along flow paths downstream of Yucca Mountain; a factor of 0.05 is used for calibration targets north of Yucca Mountain where the hydraulic gradient is high. This weighting approach appropriately places greater importance on matching those calibration targets most important for calculating flow paths downgradient from the Yucca Mountain area. Ground-water specific discharge estimates from the Death Valley Regional Groundwater Flow Model (D'Agnese, et al., 1997) also are used as calibration targets for specific discharges through lateral boundary segments of the site-scale model. The NRC staff previously expressed a concern (Reamer, 2000) that the Death Valley Regional Groundwater Flow Model has been significantly improved since it was used as a calibration target for the site-scale model, and it is not clear how boundary specific discharge and recharge estimates from the improved regional-scale model compare to those used in developing the site-scale model. DOE agreed (Agreement USFIC.5.02) to provide information to address this concern, but that information was not available at the time of this status assessment.

Effects of anisotropic permeability (i.e., permeability that varies with direction) also are included in the site-scale flow model abstraction (Bechtel SAIC Company, LLC, 2003a, Appendix E; 2003b, Section 6.4.3; 2003c, Section 6.5.2.10). To account for effects of stratification, vertical permeabilities of tuff and alluvial units in the calibrated flow model are assumed to be one-tenth of the horizontal permeability. Horizontal anisotropy is not considered in the calibrated model, but a range of horizontal anisotropy values is used to create a set of flow fields for the 200 Monte Carlo realizations of radionuclide transport for total system performance assessment (Bechtel SAIC Company, LLC, 2003c). These saturated zone flow fields are developed from the site-scale model using 12 different horizontal anisotropy ratios ranging from 0.05 to 20 for a section of the volcanic tuff units downgradient from Yucca Mountain (Bechtel SAIC Company, LLC, 2003c, Table 6-8). Anisotropy ratios less than one represent preferential east-west permeability; ratios greater than one represent preferential north-south permeability. Conceptual models both with and without vertical anisotropy also are included. Anisotropic permeability also is considered for faults by treating them as horizontally anisotropic features that have higher permeability in the strike and vertical directions and lower permeability in the direction across the fault (Bechtel SAIC Company, LLC, 2003b, Table 12).

Uncertainty in present-day ground-water specific discharge is considered by developing steady-state flow fields that consider a range of scaled permeability and recharge rates. Scaling of all permeability and recharge rates by the same proportion throughout the model domain results in proportional increases or decreases in specific discharge throughout the model domain while maintaining the same model calibration. The set of flow fields developed for the radionuclide transport abstraction includes the use of five different specific discharge scaling factors: 1/30, 1/3, 1.0, 3, and 10 (Bechtel SAIC Company, LLC, 2003c, Table 6-8).

To include the effects of increased ground-water specific discharge under future wetter climate conditions, calculated present-day radionuclide transport times for the saturated zone are reduced in proportion to the estimated increase in specific discharge (Bechtel SAIC Company, LLC, 2003c, Section 6.5). A scaling factor of 3.9 is used for the glacial-transition climate state, and a factor of 2.7 is used for the monsoon climate state. The assumption that ground-water specific discharge will increase in proportion to increased recharge during wetter climate periods is consistent with the principle of conservation of mass and is an acceptable means of integrating the saturated zone flow model abstraction with climate and infiltration models. It should be noted that this simple scaling approach to account for climate change ignores the effects of climate-induced water table rise on saturated zone flow paths. This simplification is supported by analyses showing climate-induced water table rise should not have a significant effect on calculated flow paths (Bechtel SAIC Company, LLC, 2003b, Section 6.4.5). An independent analysis of water table rise on saturated zone flow paths (Winterle, 2003) is consistent with this conclusion.

Effective porosity, which affects the ground water velocity for a given specific discharge, is considered in the saturated zone radionuclide transport abstraction. Ranges of effective porosity values for volcanic tuffs and alluvium units are stochastically sampled in the Monte Carlo analyses used to develop 200 realizations of saturated zone radionuclide transport for total system performance assessment. The sampled distribution for the effective porosity of volcanic tuff includes a range of values from 10^{-5} to 0.1. The porosity of alluvial units is sampled from a truncated normal distribution that ranges from 0.0 to 0.30 (Bechtel SAIC Company, LLC, 2003c, Table 6-8).

Several features, events, and processes have been excluded from the abstraction of flow paths in 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). DOE has indicated (Bechtel SAIC Company, LLC, 2003c, Section 6.2) that the features, events, and processes screening arguments will be updated in support of a potential license application.

In summary, the model domain, numerical grid discretization, and calibration approach used in the abstraction of saturated zone flow paths appear to be sufficient to predict flow paths from the potential repository area to the compliance boundary. The saturated zone flow model represents flow system features and boundary conditions that may affect predicted flow paths and ground-water specific discharges. The integration of the flow model with the radionuclide transport model allows consideration of factors that affect ground-water velocity, including specific discharge, effective porosity, and effects of climate change. The integrated saturated zone flow model also allows consideration of anisotropic permeability and preferential flow within structural features, and the resulting effects on the location where flow paths transition from volcanic tuff to alluvium.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.8.5), is sufficient to expect that the information necessary to assess flow paths in the saturated zone with respect to system description and model integration will be available at the time of a potential license application.

5.1.3.8.4.2 Data and Model Justification

Data and analyses used to justify the conceptual framework, process model development, and model abstraction for saturated zone flow paths are summarized in several DOE documents (Bechtel SAIC Company, LLC, 2003a–c).

Justification of the underlying Hydrogeologic Framework Model is provided in a report (U.S. Geological Survey, 2001) that describes the conceptual foundation for the hydrostratigraphy of the site-scale three-dimensional flow model. Available hydrogeologic data used to develop the Hydrogeologic Framework Model include the Geologic Framework Model (CRWMS M&O, 2000d), 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 evaluated the Luckey, et al. (1996) conceptual model and found it provided an adequate basis for a ground-water flow model, with the exception of uncertainty in properties of the alluvial aquifer system and location of the tuff-alluvium interface (NRC, 1999). This uncertainty has recently been reduced by drilling and logging activities at several new well locations. A summary of this recent information was provided by DOE (Bechtel SAIC Company, LLC, 2003a, Appendix G) to justify the treatment of uncertainty in the location of the tuff-alluvium interface in the abstraction of saturated zone flow and transport. The constraints provided by this additional well data indicate that the saturated zone flow paths should comprise between 1 to 10 km [0.62 to 6.2 mi] of the total flow distance to the compliance boundary, depending on the effect of horizontal anisotropy on flow paths and the location at which the flow paths transition from tuff to alluvium.

The DOE model considers a range of values for horizontal anisotropy for the permeability of saturated volcanic tuff in a defined region of the model downgradient from Yucca Mountain (Bechtel SAIC Company, LLC, 2003a, Appendix E). The anisotropic nature of volcanic tuff is supported by several factors. These factors include the presence of several predominantly north-striking faults and fracture orientations and interpretations of drawdown in observation wells during a long-term pumping test at the C–Holes Complex. Additionally, the DOE model report (Bechtel SAIC Company, LLC, 2003b) notes that model calibration error was slightly improved for an alternative model using a 5:1 horizontal anisotropy ratio to account for preferential permeability with a north-south orientation. Thus, the inclusion of horizontal anisotropy is consistent with available site data.

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, 2000g), (ii) estimates of recharge from analysis of stream flows in Fortymile Wash (Savard, 1998), and (iii) regional ground-water-specific discharges predicted by the Death Valley Regional Groundwater Flow Model (D'Agnese, et al., 1997). Lateral recharge, estimated from the Death Valley Regional Groundwater Flow Model, accounts for the vast majority of ground water inflow to the site-scale saturated zone model. As previously mentioned, NRC noted (Reamer, 2000) the Death Valley Regional Groundwater Flow Model has been significantly modified and refined since the abstraction of saturated zone flow paths. DOE agreed to provide information to address the change to the regional flow model, however, that information was not available at the time of this status assessment.

Water level data collected in Yucca Mountain wells (U.S. Geological Survey, 2000) indicate areas of moderate and high hydraulic gradients west and north of Yucca Mountain. East and southeast of Yucca Mountain, the hydraulic head and the hydraulic gradient reflected in water levels are significantly lower than those to the west and north. Lower water levels in wells east of the Solitario Canyon fault support the conceptual model of eastward flow directly beneath Yucca Mountain that gradually turns southward in the vicinity of Fortymile Wash. The calibrated saturated zone site-scale flow model reproduces this moderate gradient in a manner consistent with available site data.

DOE interprets the moderate hydraulic gradient as caused by a low permeability zone in the area of the Solitario Canyon fault. This interpretation is supported by wells drilled on Yucca Mountain that indicate low permeability just east of the Solitario Canyon fault. For example, transmissivity estimates for the volcanic tuffs in Wells USW 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 appear to increase 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).

The cause of the large hydraulic gradient north of Yucca Mountain is less certain, however, observations from wells USW WT–24 and USW G–2 suggest the low permeability of the Calico Hills unit, which dips below the water table in this area, could restrict flow and cause higher hydraulic heads to the north. Model results suggest the cause of large hydraulic gradient is not important to determining ground-water flow paths and specific discharges downgradient from Yucca Mountain (Bechtel SAIC Company, LLC, 2003b, Section 6.4.1).

The calibrated site-scale saturated zone flow model also emphasizes the need to reproduce an upward vertical hydraulic gradient between the Paleozoic carbonate aquifer and the overlying volcanic tuff. Data to support the existence of this upward gradient come from Wells UE–25p#1, USW H–1, USW H–3, and NC–EWDP–2DB. Hydraulic heads in UE–25p#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 Well UE–25p#1 are separated by the lowermost volcanic confining unit (Luckey, et al., 1996). Well USW H–1 does not penetrate to the carbonate aquifer, but reaches the lower portion of the lowermost volcanic confining unit where observed heads are approximately 50 m [164 ft] greater than in the overlying tuff aquifer (e.g., Graves, et al., 1997). Similarly, hydraulic potentials in Well USW H–3 are nearly 30 m [98.4 ft] higher in the lower interval than in the upper interval. Well NC–EWDP–2DB, located in southern Fortymile Wash, is only the second well in the vicinity of Yucca Mountain to penetrate the carbonate aquifer. Data from this well also indicate hydraulic potentials are higher in the Paleozoic carbonates than in the overlying tuff and alluvial aquifers.

DOE has water level data from 115 wells to provide calibration targets for the saturated zone site-scale flow model (Bechtel SAIC Company, LLC, 2003b, Table 13). These calibration points are distributed throughout the model domain, both horizontally and vertically, but are present in greater density in the area of potential flow paths. To achieve calibration, permeability values of hydrogeologic units and hydrologic features are adjusted to match hydraulic potentials inferred from the water-level data. Adjustment of the permeability values is constrained within ranges of values based on the judgment of model developers (Bechtel SAIC Company, LLC, 2003b, Table 14). The constraints on permeability values for the calibrated model generally are consistent with permeability estimates obtained from aquifer pumping tests. Given the limited number of pumping tests that have been 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 (e.g., see Bechtel SAIC Company, LLC, 2003b, Figures 37 and 38).

The use of ground-water specific discharge scaling factors to account for future wetter climate conditions is supported by comparison with other modeling analyses. The scaling factor of 3.9 for the glacial-transition climate is based on an analysis performed with the Death Valley regional flow model (D'Agnese, et al., 1999). The ratio of glacial-transition infiltration in the unsaturated zone model to the present-day infiltration also is approximately 3.9 (CRWMS M&O, 2000g). Based on this correspondence, DOE assumes the unsaturated zone infiltration ratio provides a reasonable estimate of the specific discharge ratio for the saturated zone. Accordingly, the scaling factor of 2.7 used for the monsoon climate represents the ratio of predicted unsaturated zone infiltration for monsoon conditions to present-day infiltration.

The DOE abstraction of saturated zone flow and transport treats saturated alluvium as a homogenous porous medium. Data from cuttings and core samples from Nye County wells, as well as an independent study of Fortymile Wash channel sediments (Ressler, 2001), however, suggest the valley-fill alluvium in the Fortymile Wash area is heterogenous with significant contrasts in porosity and permeability. Such heterogeneity could result in channelization of flow into relatively fast-moving pathways, which is an uncertainty that should be considered in the performance assessment abstraction. DOE justifies the treatment of alluvium as homogenous by stating the potential for preferential pathways in alluvium is implicitly included in the saturated zone transport model through the range of uncertainty in the effective porosity values (Bechtel SAIC Company, LLC, 2003a, Appendix B; 2003c, Section 6.5.2.3). The NRC review of the treatment of uncertainty of effective porosity is discussed in the following Section 5.1.3.8.4.3.

In summary, representations of flow system features and boundary conditions that may affect flow paths or ground-water specific discharges are reasonably based on supporting data. The model calibration approach relies on a sufficient number of documented observations in locations in and around areas of predicted flow paths. Factors such as effective flow porosity and potential climate changes that can affect ground water flow velocity along predicted flow paths are reasonably based on supporting data. Permeability values and anisotropy ratios for hydrogeologic units, which can affect the location where flow paths transition from volcanic tuff to alluvium, also are reasonably based on supporting site data.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.8.5), is sufficient to expect that the information necessary to assess flow paths in the saturated zone with respect to data being sufficient for model justification will be available at the time of a potential license application.

5.1.3.8.4.3 Data Uncertainty

Uncertainty in present-day ground water specific discharge is a result of uncertainties in the calibrated model permeabilities and in the prescribed boundary and recharge conditions that lead to the distribution of ground-water flow throughout the model domain. This uncertainty is addressed in the site-scale saturated zone flow abstraction by developing steady-state flow fields using a range of scaling factors for permeability values and recharge rates. Scaling all permeability and recharge rates by the same proportion throughout the model domain results in proportional increases or decreases in specific discharge throughout the model domain while maintaining the same model calibration. The 200 sets of flow fields developed for stochastic sampling in the radionuclide transport abstraction include the use of 5 values for specific discharge scaling factors: 1/30, 1/3, 1.0, 3, and 10 (Bechtel SAIC Company, LLC, 2003c, Table 6-8). Previously, the abstraction of saturated zone flow paths used only factors of 1/10, 1.0, and 10 based on a range of estimates obtained from expert elicitation (CRWMS M&O, 1998). The basis for the revised distribution is not clear, and DOE has agreed to provide additional information to support the uncertainty distribution for ground-water specific discharge estimates (Agreement USFIC.5.02), however, the requested additional information was not available at the time of this status assessment.

Data uncertainty related to horizontal anisotropy also is reflected in the sets of flow fields developed from the saturated zone site-scale flow model as input for the radionuclide transport abstraction. These flow fields are developed using 12 different values for the horizontal anisotropy ratio, ranging from 0.05 to 20 (Bechtel SAIC Company, LLC, 2003c, Table 6-8). Anisotropy ratios less than one represent preferential east-west permeability; ratios greater than one represent preferential north-south permeability. Ninety percent of the probability weighting for performance assessment is given to horizontal anisotropy ratios greater than 1.0, which is consistent with the predominant orientation of fractures and faults in the region. The analyses DOE uses to develop this stochastic uncertainty distribution make use of available site data, interpretations from the long-term aquifer pumping test at the C-Holes Complex, and an analysis of the effects of the horizontal anisotropy ratio on the site-scale flow model calibration (Bechtel SAIC Company, LLC, 2003a, Appendix E). The flow path modeling results DOE provides suggest the range of uncertainty considered for horizontal anisotropy produces significant variability in the location where flow paths transition from volcanic tuff to alluvial aquifer systems. The propagation of the stochastic distribution of flow paths into the flow and transport abstraction for performance assessment indicates this important parameter uncertainty is appropriately considered in performance assessment analysis.

Uncertainty in effective porosity, which affects the ground-water velocity for a given specific discharge, is considered in the saturated zone radionuclide transport model. Ranges of effective porosity values for volcanic tuffs and alluvium units are stochastically sampled in the Monte Carlo analyses used to develop 200 realizations of saturated zone radionuclide transport for total system performance assessment. 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 included in the radionuclide transport abstraction using a range of values from 10^{-5} to 10^{-1}, with 75 percent of the probability distribution given to values between 10^{-4} and 10^{-2} (Bechtel SAIC Company, LLC, 2003c, Section 6.5.2.5). Based on a previous assessment of effective porosity (Farrell, et al., 2000), this range is considered to bound the uncertainty of this parameter.

Uncertainty in effective porosity of alluvium is included in the radionuclide transport abstraction by stochastic sampling from a truncated normal distribution with a mean value of 0.18, a standard deviation of 0.051, a lower bound of 0.0 and an upper bound of 0.30 (Bechtel SAIC Company, LLC, 2003c, Section 6.5.2.3). This effective porosity distribution for alluvium is based mainly on a study of hydraulic characteristics of alluvium within the North American Basin and Range Province (Bedinger, et al., 1989). The upper bound of the distribution is based on a site-specific total porosity estimate from Well NC–EWDP–19D and a study of alluvium porosity in Frenchman Flat on the Nevada Test site (Burbey and Wheatcraft, 1986). A single corroborative site-specific effective porosity estimate of 0.1 was obtained from a single-well tracer test at Well NC–EWDP–19D (Bechtel SAIC Company, LLC, 2003c, Section 6.5.2.3). Estimates of alluvium effective porosity gathered during the saturated zone expert elicitation (CRWMS M&O, 1999) also are presented by DOE as corroboration. Staff agree these points of corroboration are generally consistent with the alluvium porosity uncertainty distribution developed for the radionuclide transport abstraction. It is not clear, however, if this uncertainty distribution implicitly includes the uncertainty regarding potential effects of heterogeneity, which could produce channelized flow, as suggested by DOE (Bechtel SAIC Company, LLC, 2003a, Appendix B; 2003c, Section 6.5.2.3). DOE agreed to provide additional information to justify the range of uncertainty considered for effective porosity of alluvium (Agreement RT.2.01). The necessary additional information was not available, however, at the time of this status assessment.

Uncertainty in the location where flow paths transition between volcanic tuff and alluvium is accounted for stochastically in the abstraction of saturated zone flow and transport. The tuff–alluvium 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] and an east-west extent of approximately 5 km [3.1 mi] (Bechtel SAIC Company, LLC, 2003c, Figure 6-8). The northern and western boundaries of the alluvial zone are varied to account for the uncertainty in geometry of tuff–alluvium interface beneath the water table. DOE explains (Bechtel SAIC Company, LLC, 2003c, Section 6.5.2.2) uncertainty in the northern extent of the alluvial uncertainty zone is bounded by the location of Well UE–25 JF#3 in which the water table is below the tuff–alluvium contact and by Well NC–EWDP–10S, in which the water table is above the tuff–alluvium contact. Consistent with this observation, the northern portion of the alluvial uncertainty zone extends from just south of Well UE–25 JF#3 to just north of Well NC–EWDP–10S. The geometry of the western edge of the tuff–alluvium transition is constrained by Wells NC–EWDP–10S, NC–EWDP–22S, and NC–EWDP–19D. These wells form a south-southwest-trending line in which the water table is above the tuff–alluvium contact. Consistent with data from these wells, the western portion of the alluvial uncertainty zone begins just west of the line defined by these wells. Outcrops of volcanic bedrock to the west constrain the western edge of the alluvial uncertainty zone.

In summary, the DOE abstraction of flow paths in the saturated zone reasonably allows consideration of the range of effects that uncertainties in the representation of flow system features and boundary conditions have on modeled flow paths and ground-water specific discharge estimates. Data uncertainty in the parameters that affect predicted ground-water flow velocity along predicted flow paths and directions, such as effective flow porosity and horizontal anisotropy ratio, are included in the model abstraction. The abstraction of flow paths in the saturated zone also allows consideration of the range of uncertainty in the location where flow paths transition from volcanic tuff to alluvium.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.8.5), is sufficient to expect that the information necessary to assess flow paths in the saturated zone with respect to data uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.8.4.4 Model Uncertainty

One model uncertainty in the DOE approach is the extent to which changes in transport pathways could result from a climate-induced water table rise. Climate-induced changes to flow paths are not considered in the site-scale saturated zone flow abstraction. A previous water table rise, a few tens of meters, has been inferred by the diatomite deposits south of Yucca Mountain, near Highway 95. Effects of such a water table rise might include changes in locations where the water table transitions from the tuff to the alluvial aquifer. Documentation of the site-scale flow model contains an analysis in which the flow model is adapted to include the effects of estimated water table rise (Bechtel SAIC Company, LLC, 2003b, Section 6.4.5). The adapted model suggests greater portions of alluvium would be present below the water table along potential flow paths in the event of a water table rise. Greater flow distance through alluvium instead of through volcanic tuff would slow transport of radionuclides; thus, from this perspective, not including water table rise in the flow model abstraction can be considered conservative. The DOE analysis also includes an assessment of the effect of water table rise on ground-water specific discharge and concludes the maximum estimated water table rise could result in approximately a factor of four increase in ground-water specific discharge. This estimate is consistent with the use of a specific discharge multiplier of 3.9 to account for the glacial-transition climate state. Based on analyses provided by DOE, the exclusion of water table rise from the flow model abstraction is justified. An analysis of potential water table rise using an independently developed site-scale saturated flow model (Winterle, 2003) also shows a rise in the water table ranging from nearly 30 to 150 m [100 to 500 ft] for the area of interest does not significantly affect ground-water flow paths from Yucca Mountain.

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 has led the NRC staff to consider an alternative conceptual model wherein the deep volcanic tuffs and carbonates are hydraulically well connected 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 potential repository overlies a portion of the moderate hydraulic gradient area. Because hydraulic heads in this moderate gradient area are similar to those in the deeper carbonate aquifer, it is conceivable potential releases of contaminants from the potential repository could enter a flow system connected to the regional carbonate aquifer system. DOE provides a modeling analysis that assumes the low-permeability zone along the Solitario Canyon fault diminished with depth, thereby allowing a significant hydraulic connection between the regional carbonate and volcanic tuff aquifers below Yucca Mountain (Bechtel SAIC Company, LLC, 2003a, Appendix D). Results of the DOE modeling, using both the original flow model and the alternative flow model, show radionuclide contamination reaching the west side of the Solitario Canyon fault would be transported southward in the tuff and carbonate aquifer, but eventually would be transported eastward across the Solitario Canyon fault, then move in an east-southeast direction, ultimately converging at nearly the same location along the compliance boundary. This modeling analysis provides a reasonable basis for the DOE conclusion that the effects on total system performance are expected to be minor of both reducing the depth of the Solitario Canyon fault and of initiating some flow paths west of the fault zone.

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. DOE is engaged in data collection in the alluvial aquifer related to the Nye County Drilling Program. DOE treats the uncertainty in alluvial sediment flow properties by stochastically varying the effective porosity value in performance assessment calculations. NRC has requested (Agreement USFIC.5.05) DOE provide hydrogeologic cross sections of the alluvial basin that include recent Nye County well data. These requested cross sections and supporting documentation and discussions of alluvial basin stratigraphy have been provided by DOE (Bechtel SAIC Company, LLC, 2003a, Appendix B), however, the staff review of this information was not complete at the time of this status assessment.

Another alternative conceptual model, proposed by scientists working for the State of Nevada, is the potential for seismically or geothermally activated perturbations of the saturated zone flow system. As supporting evidence, the State of Nevada scientists cite 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; Dublyansky, et al., 2001). It is important to note, however, the State of Nevada researchers have not provided details explaining the mechanism by which seismic or geothermal events could trigger water table rise of several hundred meters over such a large area. Several previous reviews have shown water table changes from earthquakes are transitory and of a limited extent (Arnold and Barr, 1996; Gauthier, et al., 1995; Carrigan, et al., 1991). The University of Nevada, Las Vegas, recently completed a 2-year study of the fluid inclusions designed to determine the ages and temperatures of secondary mineralization at Yucca Mountain (Wilson and Cline, 2002a,b). The conclusion of that independent study is the fluid inclusion data support a conceptual model wherein two-phase fluid inclusions were formed by descending meteoric water that infiltrated a cooling volcanic tuff sequence, became heated, and precipitated secondary minerals within the unsaturated zone. The Wilson and Cline study does not attempt to explain how the unsaturated zone at Yucca Mountain was able to remain hot for several million years after the last tuffs were erupted. Following the publication of the reports by Wilson and Cline, a three-part report, funded by the State of Nevada, was published by TRAC Corporation (Szymanski and Harper, 2002; Szymanski, et al., 2002; Dublyanski, et al., 2002). This report challenges the independent conclusions reached by the University of Nevada group, but provides no details explaining the mechanism by which seismic or geothermal events could trigger such a sustained water table rise for such a large scale. The NRC staff is assessing available information to determine what, if any, additional information may be needed. DOE agreed (Agreement ENFE.2.03) to provide the updated screening argument for the decision to exclude the upwelling of hot water from consideration in performance assessment models.

The DOE documentation of the site-scale saturated zone flow model also includes analyses to evaluate the potential significance of several alternative conceptual models (Bechtel SAIC Company, LLC, 2003b, Table 8). These analyses include alternative conceptualizations of the degree of vertical anisotropy, the cause of the large hydraulic gradient north of Yucca Mountain, and different interpretations of water level data. These analyses indicate it is not necessary to propagate these alternative conceptual models forward into the performance assessment abstraction because they have no significant effect on ground-water flow paths or velocities beyond the range of uncertainty already considered.

In summary, DOE has considered numerous alternative conceptual models for saturated zone flow that are consistent with available site data. Viable alternative conceptualizations have been excluded based on appropriate levels of detailed analyses. DOE analyses suggest effects of a climate-induced water table rise on saturated flow paths would have minimal effect on flow paths beyond that already considered in the abstraction.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.8.5), is sufficient to expect that the information necessary to assess flow paths in the saturated zone with respect to model uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.8.4.5 Model Support

The documentation of the site-scale saturated zone flow model contains a detailed summary of data and analyses used for model support (Bechtel SAIC Company, LLC, 2003a, Appendix D; 2003b, Section 7).

Another source of model support is data from newly drilled Nye County wells that were not used in the model calibration. Water level data from these wells (Bechtel SAIC Company, LLC, 2003a, Table D–3) show residual differences between model-calculated water levels are similar in magnitude to residual errors obtained for wells used in the calibration process. Of the newer Nye County wells, only data from Wells NC–EWDP–19D and NC–EWDP–19P are located along likely flow paths from Yucca Mountain and not used for model calibration. These two wells monitor deep and shallow portions of the alluvial aquifer, respectively, at essentially the same location. The observed water level reported for Well NC–EWDP–19D is 0.4 m [1.3 ft] greater than the calculated water level; the difference is 5.7 m [19 ft] for Well NC–EWDP–19P. This magnitude of error is generally consistent with the range of errors reported for wells along Yucca Mountain flow paths used in the flow model calibration process. The largest differences between calculated and observed water levels generally occur in near hydrogeologic features that result in steep hydraulic gradients. In addition, model calibration errors are generally normally distributed among positive and negative values and, thus, are free from bias.

Model support provided by DOE also includes a plot of model-calculated hydraulic heads at selected points along a transect that follows projected flow paths from Yucca Mountain to the compliance boundary (Bechtel SAIC Company, LLC, 2003a, Figure D–5). This plot of calculated hydraulic heads versus distance is compared with a plot of observed heads versus distance. In these plots, the slope of the lines between observation points is an indicator of the approximate hydraulic gradient along the projected flow path. The comparison of plots shows that, along most of the projected flow path from Yucca Mountain, calculated hydraulic gradients are in good agreement with the hydraulic gradients inferred from observed differences in hydraulic heads between well locations. In this comparison of inferred hydraulic gradients, head data from five of the six wells used in the plot also are used in the model calibration. Because a model calibrated to match individual water level observations does not guarantee the model will reasonably reproduce hydraulic gradients between wells, knowledge that the modeled hydraulic gradients are in good agreement with the gradients inferred from observations provides a measure of confidence beyond that gained by simply achieving a good model calibration.

For most hydrogeologic units represented in the flow model, the calibrated permeability values are within the range of values reported from *in-situ* testing. This consistency is to be expected, however, because, although *in-situ* permeability estimates are not used as calibration targets, they are used to guide constraints on the range of permeability values considered for each hydrogeologic unit during the calibration process. The only new permeability estimates reported by DOE are those obtained from the Alluvial Testing Complex at Well NC–EWDP–19D. The calibrated model permeability for alluvium at this location was one order of magnitude greater than the permeability estimated from a single-hole test and was 19 percent greater than the permeability estimated from a cross-hole test at this location. In general, the permeability estimate from the larger-scale cross-hole test can be considered more reliable for estimating aquifer permeability at the scale of flow model grid than the estimate obtained from the single-hole test. The calibrated permeability assigned to the model for alluvium can be considered within the range of uncertainty typically ascribed to pumping test results.

Additionally, the fact that the calibrated permeability value is greater than the estimates from the Alluvial Testing Complex would conservatively favor higher estimates of ground-water specific discharge and velocity.

Geochemical data also are cited by DOE as a source of support for saturated zone flow abstraction (Bechtel SAIC Company, LLC, 2003b, Section 7.3). This analysis of ground water chemistry indicates ground water chemistries can be divided into geochemical ground water types that trend along generally north-south orientations that are broadly consistent with flow paths predicted by the flow model. The summary of hydrochemical data trends provided by DOE suggests ground waters from the Crater Flat, Yucca Mountain, and Jackass Flat areas converge in the Northern Amargosa Valley area. This interpretation is broadly consistent with flow path predictions that result from the saturated zone flow model.

A model of thermal transport developed from the site-scale flow model is also presented as model support (Bechtel SAIC Company, LLC, 2003b, Section 7.4). This analysis considers both a conduction-only model and a model of coupled conduction and ground water advection. The conduction-only model shows simulated ground-water temperatures are largely influenced by thickness of the unsaturated zone. Higher temperatures correspond to the relatively thick unsaturated zones under Yucca Mountain in the central model region and under the Calico Hills in the northeastern model region. This study, therefore, does nothing to improve confidence in the ground-water flow paths and specific discharge calculated with the saturated zone flow model. Interestingly, residual errors in matching temperature observations increased after coupling ground water advection to the thermal conduction model. This increase in error might seem to suggest ground-water flow rates predicted by the flow model are not validated by the temperature data. In actuality, however, the reason for the increased residual error is that ground-water flow is coupled to a calibrated conduction-only model, and the thermal properties and boundary conditions are not recalibrated after including advective processes. Hence, as currently developed, the coupled model of thermal conduction and advection neither validates nor invalidates ground-water flow fields predicted by the DOE saturated zone flow model.

DOE also provided an analysis of ground-water residence time using C-14 data as a line of evidence to support the abstraction approach (Bechtel SAIC Company, LLC, 2003a, Appendix F). Interestingly, C-14 data do not show a clear decrease in activity from north to south along likely flow pathways, which suggests ground water may be affected by recharge and ground-water mixing along the entire flow path. The mixing of ground water of various ages, combined with significant uncertainty in the locations and compositions of recharge source areas, and the degree of calcite dissolution during water-rock interactions make it difficult to obtain reliable estimates of ground-water residence times. This difficulty is reflected in the broad range, from 0 to 10,000 years, estimated for ground-water residence time based on differences in C-14 ages between the area below the potential repository area and the accessible environment at the compliance boundary. Although ground-water traveltimes predicted by the DOE models fall within this range, this fact provides little additional confidence in the models because the range is so broad. The DOE site-scale saturated zone flow model for Yucca Mountain includes spatially variable recharge rates at Yucca Mountain, the higher elevation areas to the north, and in Fortymile Wash. This inclusion of ground-water recharge is broadly consistent with the interpretation of recharge and mixing along flow paths.

In summary, no one source of model support provides complete confidence in predicted flow paths and ground-water specific discharge. The sum of available model support information is, however, generally consistent with the abstraction of saturated zone flow paths. Modeled flow paths, ground-water velocities, and locations of flow path transitions from tuff to alluvium are supported by objective comparisons with site data, including observations not used for model calibration or development. The calibrated saturated zone flow model reasonably minimizes residual errors between model calculations and calibration targets; and residual calibration error is free from bias.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.8.5), is sufficient to expect that the information necessary to assess flow paths in the saturated zone with respect to model abstraction output being supported by objective comparisons will be available at the time of a potential license application.

5.1.3.8.5 Summary and Status of Key Technical Issue Subissues and Agreements

The likely flow paths through the saturated zone are through volcanic tuff and porous alluvium. The extent of the flow path through alluvium is important because of the potential capability of alluvial materials to retard radionuclide transport. In the discussion of risk insights in Appendix D, the length of the flow path through the saturated alluvium is assigned medium significance. Current agreements between DOE and NRC related to this aspect of flow paths in the saturated zone are, therefore, also of medium significance, and agreements that pertain to other aspects of the flow paths through the saturated zone are anticipated to be of lower significance with regard to performance. Aspects of performance related to retardation of radionuclide transport along flow paths through the saturated zone are considered in Section 5.1.3.9.

Table 5.1.3.8-1 provides the status of all key technical issue subissues referenced in Section 5.1.3.8.2 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 review methods discussed in Section 5.1.3.8.4. Note the status and detailed agreements pertaining to all the key technical issue subissues are provided in Table 1.2-1 and Appendix A.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (e.g., specified testing or analyses), indicates that information necessary to begin a technical review will likely be available at the time of a potential license application.

Table 5.1.3.8-1. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreement*
Unsaturated and Saturated Flow Under Isothermal Conditions	Subissue 2—Hydrologic Effects of Climate Change	Closed	None
Unsaturated and Saturated Flow Under Isothermal Conditions	Subissue 5—Saturated Zone Flow and Dilution Processes	Closed-Pending	USFIC.5.01 through USFIC.5.14

Table 5.1.3.8-1. Related Key Technical Issue Subissues and Agreements (continued)			
Key Technical Issue	**Subissue**	**Status**	**Related Agreement***
Unsaturated and Saturated Flow Under Isothermal Conditions	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 through 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
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 review methods.

Note: Key Technical Issue Agreement GEN.1.01 pertains to multiple integrated subissues, as well as some specific issues related to this integrated subissue.

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5.1.3.9 Radionuclide Transport in the Saturated Zone

5.1.3.9.1 Description of Issue

The radionuclide transport in the saturated zone model abstraction addresses features and processes that would affect movement of radionuclides in the saturated zone from the area beneath the potential repository site at Yucca Mountain to the proposed compliance boundary approximately 18 km [11 mi] downgradient from Yucca Mountain. The rate radionuclides migrate through the saturated zone depends on the water flow rate, the nature of the geologic materials through which the water travels—fractured volcanic rock or porous alluvium—and the water chemistry and mineralogy of the system. Figure 5.1.3.9-1 illustrates the relationship between the radionuclide transport in the saturated zone model abstraction and the flow paths in the saturated zone model abstraction (Section 5.1.3.8). The overall organization and identification of all the integrated subissues are depicted in Figure 1.1-2. DOE has described and documented the technical bases and its approach to modeling saturated zone transport in numerous reports prepared to support the potential license application (Bechtel SAIC Company, LLC 2003a–e) and the previous site recommendation (CRWMS M&O 2000a–o). The technical basis for abstraction of radionuclide transport in the saturated zone is summarized in Bechtel SAIC Company, LLC (2003b). Implementation of the abstraction in Total System Performance Assessment–Site Recommendation is described in CRWMS M&O (2000b,c). DOE recently updated the abstractions in Bechtel SAIC Company, LLC (2003f).

This section documents the current NRC understanding of the model abstractions developed by DOE to incorporate radionuclide transport in the saturated zone into its total system performance assessment. This section is focused on those aspects most important to waste isolation based on the risk insights gained to date, including Appendix D. The assessment presented is limited to examining if data gathered and methodologies developed by DOE are likely to be adequately documented for the staff to undertake a detailed technical review. This assessment is not a regulatory compliance determination review of a potential license application.

5.1.3.9.2 Relationship to Key Technical Issue Subissues

This Radionuclide Transport in the Saturated Zone Integrated Subissue incorporates subject matter previously captured in the following 12 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)

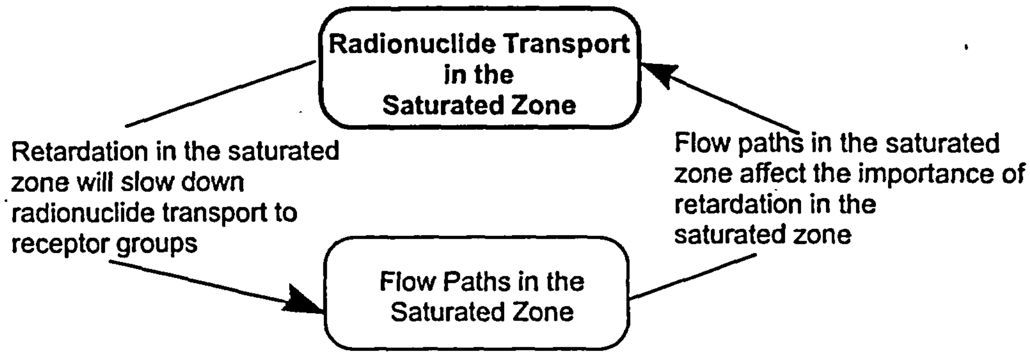

Figure 5.1.3.9-1. Diagram Illustrating the Relationship Between the Radionuclide Transport in the Saturated Zone and Flow Paths in the Saturated Zone Integrated Subissues. Material in Bold Is Identified in the Text.

- 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)
- 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 would provide to NRC. 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.

5.1.3.9.3 Importance to Postclosure Performance

Risk insights pertaining to radionuclide transport in the saturated zone indicate that retardation in the saturated zone is of high significance to waste isolation. Matrix diffusion in the saturated zone and the effect of colloids on transport in the saturated zone are assigned medium significance. The details of the risk insights ranking are provided in Appendix D. DOE identifies radionuclide delay through the saturated zone at Yucca Mountain as a principal factor of the postclosure safety case (CRWMS M&O, 2000d). One aspect of risk informing the NRC review is to determine how this integrated subissue is related to the DOE repository safety strategy. As described in Revision 4.0 of 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 (Bechtel SAIC Company, LLC, 2003f). 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. Some radionuclides are commonly associated with colloids, which can be transported in ground water. Thus, the processes that control colloid-facilitated transport of radionuclides also must be considered.

DOE investigated the importance of saturated zone transport through robustness and neutralization analyses (CRWMS M&O, 2000b,d). The degraded barrier analysis, in which 5th percentile values are used for parameters that positively promote delay of radionuclides in the saturated zone and 95th percentile values for parameters that positively promote transport in the saturated zone, suggests modest sensitivity (CRWMS M&O, 2000d) of dose to the saturated zone transport barrier. The similarity of the degraded barrier analysis 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 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). An independent NRC performance assessment sensitivity analysis has concluded retardation in the saturated zone is important, based on much higher modeled doses that result from removal of retardation from the analysis (NRC, 1999b; Mohanty, et al., 2002). In particular, neptunium retardation has been shown to have a significant dose effect (NRC, 2001b, 1999b). Sensitivity analyses using mean parameter values from the NRC TPA Version 4.1 code (Mohanty, et al., 2002, Section 3.3.6) suggest that, for the basecase, radionuclide retardation in the saturated zone provides a substantive delay in the release of radionuclides. Assuming no retardation at all for plutonium, americium, and thorium in both the unsaturated and saturated zones increases the expected ground water dose by one to three orders of magnitude for a 100,000-year simulation period. Assuming no matrix diffusion results in a peak expected dose that is approximately 450 years earlier and 50-percent higher than the basecase simulations (Mohanty, et al., 2002, Section 3.5.3).

DOE also examined the role of the saturated zone as a barrier using neutralization analyses (Bechtel SAIC Company, LLC, 2002). In these analyses, radionuclide transport in the saturated zone is demonstrated to be a potentially significant contributor to waste isolation. Similar barrier neutralization analyses were conducted using the NRC TPA Version 4.1 code (Mohanty, et al., 2002, Section 6.4.1). These analyses demonstrated suppression of the saturated zone as a repository component results in a 900-percent increase in peak expected dose.

In establishing its risk insights baseline (Appendix D), the NRC staff determined the significance of several aspects of radionuclide transport through the saturated zone to repository performance. Specifically, retardation of radionuclides in the saturated alluvium (and associated flow path length through the alluvium; see Section 5.1.3.8) has been assigned high significance to waste isolation, while matrix diffusion and the effect of colloids on radionuclide transport in the saturated zone are assigned medium significance to waste isolation. The following assessment of the DOE characterization and performance assessment abstraction of radionuclide transport in the saturated zone is conducted at a level of detail appropriate to the assigned degree of significance. DOE prepared a technical basis document and supporting reports (Bechtel SAIC Company, LLC 2003b–f) that summarize the saturated zone flow and transport model abstraction for the potential license application in December 2004. Not all the supporting references have been released to the public, however, and, therefore, these have not been considered by NRC in this assessment.

5.1.3.9.4 Technical Basis

NRC has developed a review plan (NRC, 2003) consistent with the acceptance criteria and review methods found in previous issue resolution status reports. A status assessment of the DOE approaches for including radionuclide transport in the saturated zone in total system performance assessment abstractions is provided in the following subsections. The assessment is organized according to the five review methods identified in Section 2.3: (i) Model Integration (including system description), (ii) Data and Model Justification, (iii) Data Uncertainty, (iv) Model Uncertainty, and (v) Model Support.

NRC previously reviewed the DOE abstraction approach for radionuclide transport in the saturated zone (CRWMS M&O, 2000a–c) after the DOE publication of the viability assessment (DOE, 1998) and after the DOE publication of the site recommendation (CRWMS M&O, 2000b). With exception of modifications to several parameter distributions, the general DOE approach for the abstraction of saturated zone radionuclide transport has not changed substantially since the site recommendation (Bechtel SAIC Company, LLC, 2003b,f).

5.1.3.9.4.1 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 (Bechtel SAIC Company, LLC, 2003f). Particle transport pathways are calculated based on spatially variable ground water flux vectors (flow fields) derived from the site-scale saturated zone flow model (Bechtel SAIC Company, LLC, 2003e). The influences of macro-scale dispersion, matrix diffusion, and adsorption of radionuclides to mineral surfaces (sorption) are incorporated through the use of a residence-time transfer function 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 traveltimes 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 sampling the probability distribution function of the particle residence time. 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 (Bechtel SAIC Company, LLC, 2003g; CRWMS M&O, 2000e).

The residence-time transfer function for the fractured tuff portion of the saturated zone 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 (Bechtel SAIC Company, LLC, 2003g). 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 process on the radionuclide transport simulations (Bechtel SAIC Company, LLC, 2003f,g). Also, it should be noted, neglecting radionuclide sorption on fracture surfaces is a conservative approach.

The saturated zone radionuclide transport component of total system performance assessment is coupled to the input of the unsaturated zone and to the output to the biosphere using the convolution integral method (Bechtel SAIC Company, LLC, 2003c,g). 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 (Bechtel SAIC Company, LLC, 2003f). The convolution integral method is computationally efficient and rests on the key assumptions of linear behavior and steady-state saturated zone flow conditions (Bechtel SAIC Company, LLC, 2003g). Release of radionuclides from the unsaturated zone into the saturated zone is assumed to occur at a point source near the water table (Bechtel SAIC Company, LLC, 2003f). The point source location is randomly sampled from one of four source regions that generally represent preferential flow pathways in the unsaturated zone flow model (Bechtel SAIC Company, LLC, 2003f).

DOE relies on linear sorption isotherms and represents all noncolloidal retardation processes using the sorption coefficient (K_d) (Bechtel SAIC Company, LLC, 2003f). A lumped parameter such as K_d does not allow explicit consideration of different processes that might affect radionuclide sorption and retardation, and care must be taken to ensure the validity of the approach is not overextended. Although transport of the radionuclide mass is distributed between colloids and dissolved components in the total system performance assessment model abstraction, aqueous speciation and other geochemical effects on sorption are considered indirectly through a K_d probability distribution function. For the site recommendation, sorption coefficients for the radionuclides of interest are selected based on an initial informal expert elicitation, although the specific constraints on some transport parameters were modified, particularly uranium, neptunium, and plutonium (Wilson, et al., 1994; CRWMS M&O, 2000g; Triay, et al., 1997). In response to radionuclide transport-related agreements (Reamer, 2000a), DOE modified the basis for sorption coefficient distributions (Bechtel SAIC Company, LLC, 2003c,f,g). Sorption parameter probability distribution functions are constrained assuming that water from the saturated volcanic tuff (Well J–13) and the Paleozoic aquifer (UE–25p#1) bound the chemistry of the ground waters at Yucca Mountain. Experimental results and process-level sorption modeling are used to delineate sorption probability distribution functions on two rock types: tuff and alluvium (Bechtel SAIC Company, LLC, 2003c,f,g). 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 (Bechtel SAIC Company, LLC, 2003f). DOE agreed to provide the technical basis for its transport parameter distributions (Reamer, 2000a) and provides updates in Bechtel SAIC Company, LLC (2003c, Attachments I and II; 2003f, Attachments I and II).

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 according to a one-dimensional transport model employed directly in total system performance assessment rather than the offline, three-dimensional model employed for radionuclides in general (Bechtel SAIC Company, LLC, 2003f,g). 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 (Bechtel SAIC Company, LLC, 2003f).

Colloidal transport in the saturated zone is handled, as elsewhere in total system performance assessment, with two types of radionuclide attachment—reversible and irreversible (Bechtel SAIC Company, LLC, 2003a,b,d,f,h). Colloids with irreversibly attached radionuclides are modeled as solutes, with a retardation factor applied specifically to the fractured tuff and alluvial aquifers to simulate the effects of nonpermanent filtration (Bechtel SAIC Company, LLC, 2003d); matrix diffusion of irreversible colloids in the saturated zone is conservatively neglected (Bechtel SAIC Company, LLC, 2003d,f). Reversible colloidal transport is modeled using the K_c factor, representing equilibrium sorption of aqueous radionuclides onto colloids (Bechtel SAIC Company, LLC, 2003a,d,g,h). Values for K_c are partitioned into three groups, depending on the radionuclides, and two substrates or colloid types. The three groups are (i) plutonium; (ii) cesium; and (iii) americium, protactinium, and thorium. The two substrates are iron oxide and smectite colloids (Bechtel SAIC Company, LLC, 2003d,h). Inclusion of reversible sorption to colloids lowers the effective diffusion coefficient D_e and the sorption coefficient K_d for the radionuclide (Bechtel SAIC Company, LLC, 2003d,f), enhancing advective transport. DOE agreed to provide the technical basis for selecting radionuclides for saturated zone transport modeling via reversible and irreversible colloid attachments (Reamer, 2000a) and provides an update in Bechtel SAIC Company, LLC (2003h), which is currently being reviewed in detail.

For site recommendation, DOE used arguments based on low probability, low consequence, or both to exclude numerous features, events, and processes from the total system performance assessment abstraction of radionuclide transport in the unsaturated zone. The screening arguments are outlined in Bechtel SAIC Company, LLC (2003d,f,g) and the features, events, and processes in CRWMS M&O (2001). Scenario analysis and the NRC assessment of the DOE screening arguments are provided in Section 5.1.2 of this report. In several cases, the screening arguments for exclusion of a particular feature, event, and process are appropriate. In other cases, however, the DOE argument is incomplete at this time. Also, in some cases, DOE has not identified a feature, event, or process as either included or excluded. DOE agreed

(Reamer, 2001) to address concerns relating to the technical basis for its screening of features, events, and processes.

DOE has screened the occurrence of far-field nuclear criticality in the unsaturated zone from its total system performance assessment based on low probability of occurrence within 10,000 years (CRWMS M&O, 2000m). This low probability is based on no waste package failures before 10,000 years; no fissile material is released, and there is no accumulation before 10,000 years through radionuclide transport in either unsaturated or saturated zones. The DOE screening arguments are discussed in Section 5.1.2.2 of this report.

In summary, the current DOE abstraction (Bechtel SAIC Company, LLC, 2003b,f) describes processes relevant to performance of the saturated zone barrier at Yucca Mountain. Processes that affect radionuclide transport including retardation, changes in water chemistry, mineralogy, matrix diffusion, colloidal transport, radioactive decay, and process coupling are considered, although in some cases only implicitly. The effect and importance of these processes differ in the fractured tuff units and the porous alluvium. In fractured tuffs, radionuclides are transported through the fractures and may diffuse into the surrounding matrix. If the radionuclides diffuse into the matrix, they also may be sorbed within the matrix of the rock (Bechtel SAIC Company, LLC 2003b). In the alluvium, because the effective porosity of the alluvium is considerably greater than that of the fractured tuff, the transport velocity in the alluvium is reduced greatly in comparison with that of the tuff (Bechtel SAIC Company, LLC, 2003b). The saturated zone radionuclide transport abstraction is closely linked to the saturated zone flow abstraction to account for these effects.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.9.5), is sufficient to expect that the information necessary to assess radionuclide transport in the saturated zone with respect to system description and model integration will be available at the time of a potential license application.

5.1.3.9.4.2 Data and Model Justification

The most recent DOE approach to transport parameter development is provided in Bechtel SAIC Company, LLC (2003b,f,g). These documents present a systematic technical basis for the K_d values and distributions of americium, cesium, neptunium, plutonium, protactinium, radium, strontium, thorium, and uranium. The K_d distributions are based on experimental data from the DOE program, and the effects of variability in geochemistry and mineral surface area are characterized for the long-lived actinides using a surface complexation modeling approach (Bechtel SAIC Company, LLC, 2003g, Attachment I). Separate sorption coefficient distributions for volcanic tuff and alluvium are determined. For volcanic tuffs, the sorption coefficient distributions are based on results of experiments on devitrified and zeolitic tuff samples (Bechtel SAIC Company, LLC, 2003g). For alluvium, only neptunium and uranium have supporting site-specific experimental evidence; the remaining radionuclide distributions are based on data from devitrified tuff samples. The limited range of geochemical conditions examined by the experiments is supplemented with a surface complexation modeling approach that allows interpolation and extrapolation of sorption coefficients for the range of chemistries applicable to the saturated zone (Bechtel SAIC Company, LLC, 2003g).

The alluvial flow path is a source of uncertainty in modeling radionuclide transport in the saturated zone (Bechtel SAIC Company, LLC, 2003e,f). DOE agreed to provide evidence from its site characterization program, including work on Early Warning Drilling Program Wells, the Alluvium Testing Complex, and related laboratory studies, to ensure data on transport properties of the alluvium are sufficient to support a potential license application (Reamer, 2000a). Recent reports provide updates of data sources used to support the alluvium transport properties (Bechtel SAIC Company, LLC, 2003b,d,f,g); however, supporting data for the alluvium are still sparse. Data to constrain the lateral extent and depth of alluvium that may occur along the saturated zone flow path are derived from the geological logging of Early Warning Drilling Program wells. There are only four drill holes available to define the extent of the alluvium along the final 8 km [5 mi] of the predicted saturated flow path (Bechtel SAIC Company, LLC, 2003f) (Section 5.1.3.8). The effective porosity used for saturated alluvium in the site recommendation is based on nonsite-specific data (CRWMS M&O, 2000h,g). Since then, one field measurement has been conducted at Well NC–EWDP–19D1. The measured value (0.10) is at the low end of the current probability distribution (0–0.30, with a mean of 0.18) used for effective porosity in the alluvium (Bechtel SAIC Company, LLC, 2003f). The effective porosity measurement was part of a single-well tracer test conducted at the Alluvium Testing Complex. Testing at the complex was cancelled after denial of tracer and water use permits (Bechtel SAIC Company, LLC, 2003f). Thus, no field-scale transport tests are available to confirm DOE estimates for alluvium transport properties.

In fractured tuffs, advective transport occurs within fractures; therefore, the effective fracture spacing and porosity are important for describing the advective velocity of dissolved constituents. Major flowing fracture zones (termed flowing intervals) are generally spaced meters to tens of meters apart, while fractures themselves may be more closely spaced. Radionuclides transported through the fractures may diffuse into the surrounding matrix and also may be sorbed within the matrix of the rock (Bechtel SAIC Company, LLC, 2003b). The analytical solution of Sudicky and Frind (1982), used to develop the residence-time transfer function, requires estimation of several parameters, including fracture aperture, mean fracture spacing (flowing interval spacing), linear ground water velocity within the fracture, rock matrix porosity, rock matrix and fracture retardation factors, and the effective matrix diffusion coefficient (Bechtel SAIC Company, LLC, 2003g). 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 (Bechtel SAIC Company, LLC, 2003f,g). 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 and the Nevada Test Site (Bechtel SAIC Company, LLC, 2003g). Field testing consists of cross-hole tracer tests within the Prow Pass tuff and Bullfrog tuff intervals of the C-Wells Complex, which shows tracers with differing diffusion coefficients are attenuated differently, with greater attenuation of the solute with a higher diffusion coefficient, as qualitatively predicted by the conceptual model (Bechtel SAIC Company, LLC, 2003f,g; Reimus, et al., 1999).

Data obtained from flow-meter surveys of several wells in the Yucca Mountain area are used to estimate a statistical distribution of the spacing between flowing intervals in the saturated volcanic tuffs (Bechtel SAIC Company, LLC, 2003f). 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 is significant variability in the amount of water produced by the various features identified as flowing intervals: some features are associated with fracture zones, others with permeable rock matrix—yet, the features are treated equally with regard to flowing interval spacing. Also, the flowing interval spacing parameter is used to support a conceptual model of flow through a series of parallel fractures; however, there is considerable variability in the strike directions and dips of the identified flowing features. Finally, the spacing between flowing intervals is 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 is used to infer the effective fracture (flowing interval) aperture for the residence-time transfer function approach. No new data are available to update the flowing interval spacing distribution (Bechtel SAIC Company, LLC, 2003f). DOE describes a sensitivity analysis about the effect of flowing interval spacing on radionuclide breakthrough in Reamer (2000a). As the spacing increases, separation of the breakthrough curves decreases, so the breakthrough curves for spacing of 50 m [160 ft] and 100 m [330 ft] are coincident. The flowing interval spacing of 21 m [69 ft] used by DOE results in a radionuclide breakthrough near the conservative limit of behavior.

The radionuclides subject to colloidal transport in the DOE total system performance assessment are identified in the inventory abstraction (CRWMS M&O, 2000k; Bechtel SAIC Company, LLC, 2003a,h). The selection of radionuclides is appropriately based on considerations of dose, inventory, and mobility. DOE agreed (Reamer, 2000a) to document how radionuclides are identified for colloidal transport in the total system performance assessment and provides this updated information in Bechtel SAIC Company, LLC (2003a,h). In the total system performance assessment model abstraction of the natural barrier system (i.e., the unsaturated and saturated zones combined), radionuclides are considered as either reversibly sorbed or irreversibly sorbed on colloidal particles (Bechtel SAIC Company, LLC, 2003a). In its analysis of colloidal transport, DOE includes plutonium, americium, thorium, protactinium, and cesium for the reversible model. These radionuclides are potentially major components of the inventory at 10,000 years. For the irreversible model, only plutonium and americium are included. Uranium and neptunium are not included in the colloidal transport abstraction because they are highly soluble and weakly sorbing for the conditions expected at Yucca Mountain (Bechtel SAIC Company, LLC, 2003a,d,h).

Americium and plutonium may be reversibly or irreversibly sorbed onto colloids. In general, the majority of americium and plutonium is irreversibly sorbed (90–99 percent of the colloid mass), while the remainder is reversibly sorbed (1–10 percent of the total colloid mass) (Bechtel SAIC Company, LLC, 2003a). Radionuclides reversibly sorbed to the colloid phase are permitted to desorb and resorb to immobile matrix minerals as determined by a sampled sorption distribution known as K_c. The K_c parameters are based on experimental data from a variety of sources (Bechtel SAIC Company, LLC, 2003h). Data for americium sorption to colloids is applied to the K_c values for americium, protactinium, and thorium because of a lack of data for protactinium and thorium (Bechtel SAIC Company, LLC, 2003h). DOE has not provided data to justify the reversible colloid attachment parameter appropriately to account for the effects of this process. The irreversibly sorbed colloids are divided into a fast component that travels unretarded through the fracture network to the water table and a slow component, subjected to retardation. The retardation factor, R_c, used for the slow irreversible colloids in fractured tuff is based on laboratory experiments and field-scale experiments conducted at the C–Wells Complex (Bechtel

SAIC Company, LLC, 2003a,h). R_c values for the saturated alluvium are based on laboratory experiments with microspheres and nonsite-specific field experiments with bacteriophages (Bechtel SAIC Company, LLC, 2003d). The DOE sensitivity analyses suggest the transport of both irreversible and reversible colloids is delayed significantly in the saturated zone.

Applicability of the microsphere results rests on assumptions regarding size distributions of microspheres versus colloids. DOE agreed to justify that microspheres can be used as analogs for colloids (e.g., equivalent ranges in size and charge) and provide constraints on colloid transport model parameters (Reamer, 2000a). DOE also agreed to use sensitivity analyses to constrain colloid transport parameters when modeling reversible and irreversible attachments and the effects of colloid transport on radionuclide transport in the saturated zone model abstraction (Reamer, 2000a). Recent DOE reports provide information to address these issues (Bechtel SAIC Company, LLC, 2003a, Appendix E; 2003b, Appendix M).

In summary, DOE uses a mix of laboratory data, field data, and process modeling results to provide a technical basis for parameters that describe the transport of dissolved radionuclides; however, limited field data exists for the alluvium. 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. DOE relies on expert judgment to establish parameter distributions for numerous parameters (e.g., K_c and R_c). Information regarding the expert judgment process is not yet available to allow a reviewer to evaluate the origins of the judgments (NRC, 1996). DOE agreed (Reamer, 2000a) to provide the documentation explaining the technical basis used to support expert judgment and the DOE approach.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.9.5), is sufficient to expect that the information necessary to assess radionuclide transport in the saturated zone with respect to data being sufficient for model justification will be available at the time of a potential license application.

5.1.3.9.4.3 Data Uncertainty

DOE uses stochastic approaches to identify and constrain data uncertainty in its model abstraction of radionuclide transport in the saturated zone (Bechtel SAIC Company, LLC, 2003b,f). The data uncertainty is represented in the performance assessment model abstraction by using probability distributions. During performance assessment calculations, the distribution is sampled in multiple realizations used to generate statistics of the estimated dose to the receptor. Appendix D indicates uncertainty in those parameters most important for waste isolation is related to sorption in the saturated alluvium, matrix diffusion in the fractured tuff, and colloid-facilitated radionuclide transport.

NRC had questions about how DOE characterized the spatial and stratigraphic variations in transport parameters of the alluvial aquifer (NRC, 2000a) at the time of site recommendation. DOE has agreed to accomplish further alluvium characterization to better define parameter variability (Reamer, 2000a). DOE recently submitted reports that update the technical bases for alluvial transport parameters (Bechtel SAIC Company, LLC, 2003b,f).

Sorption processes can provide a significant delay of the transport of radionuclides through the saturated zone. In particular, saturated alluvium can delay the breakthrough of even weakly sorbing radionuclides, such as neptunium, so the majority of calculated breakthrough may be delayed beyond 10,000 years (Bechtel SAIC Company, LLC, 2003g). Previous sensitivity studies show traveltimes through the saturated zone are sensitive to the values of sorption coefficients used for nuclides like neptunium (Reamer, 2000a). DOE agreed to provide additional information to support the sorption coefficient parameters used for the alluvium in Reamer (2000a). Risk insight information developed by NRC (Appendix D) also indicates neptunium is the most significant radionuclide affected by sorption in alluvium. DOE provides an update to the technical bases used to determine sorption coefficient distributions used for the saturated alluvium in Bechtel SAIC Company, LLC (2003g, Appendix I). Sorption coefficients for neptunium in alluvium are derived from laboratory experiments conducted using site-specific alluvium samples. Although water chemistries used in the experiments are limited in range, these chemistries are representative of water chemistries at the sample sites (Bechtel SAIC Company, LLC, 2003b). The neptunium sorption coefficient distribution is based on results of experiments that excluded fine particle sizes {<75 µm [<2.95 × 10^{-3} in]} (Bechtel SAIC Company, LLC, 2003h, Appendix I). The distribution is a piecewise-uniform distribution with a mean value of 6.3 mL/g [10.9 in^3/oz], a minimum of 1.8 mL/g [3.1 in^3/oz], and a maximum of 13 mL/g [22.5 in^3/oz]; the range between 5 and 95 percent probability is represented by a uniform segment from 4.0 to 8.7 mL/g [6.9 to 15.1 in^3/oz]. Sorption coefficients for uranium in the saturated alluvium also are derived using results of laboratory experiments on site-specific materials; however, alluvium sorption coefficients for the remainder of the sorbing radionuclides are based on laboratory experiments using devitrified tuff. The sorption parameter ranges for the actinides (americium, neptunium, plutonium, thorium, and uranium) also are supported by surface complexation modeling using the computer code PHREEQC (Parkhurst and Appelo, 1999). This process modeling is calibrated against experimental data external to the DOE program and is used to investigate the effects of observed variability in geochemistry and mineralogy. Process modeling has not been used to support the parameter distributions for cesium, protactinium, radium, and strontium (Bechtel SAIC Company, LLC, 2003h). Use of process modeling to extend the limited chemical conditions considered in the batch experiments with crushed tuff has provided a stronger technical basis for the upper and lower limits, and the upper limit is conservative (less than) the observed sorption values. Although the upper and lower limits of the K_d cumulative distributions are based on experimental data supported by process modeling, shapes of the distributions are assigned through expert judgment.

The evaluation of uncertainty of flow path lengths in tuff and alluvium has been incorporated into the saturated zone transport model by identifying an alluvium uncertainty zone and then abstracting it as a polygonal region assigned radionuclide transport properties representative of the alluvium (Bechtel SAIC Company, LLC, 2003b). The dimensions of the polygonal region, in particular, the northern and western boundaries, are randomly varied for the multiple realizations used in probabilistic assessment of uncertainty. The flow path lengths in the alluvium and fractured tuffs are justified using field data and analyses. The uncertainty is bounded by the locations of Early Warning Drilling Program wells that have penetrated saturated alluvium and outcrops or well penetrations of volcanic tuffs to the north and west. Because of a relative lack of data, a uniform distribution is used for the sampled boundary location distribution (Bechtel SAIC Company, LLC, 2003f). When combined with predicted flow paths from the potential repository, the minimum expected alluvium flow path length is approximately 2 km [1.25 mi]. As discussed in Section 5.1.3.9.4.2, only one site-specific measurement has been made for effective porosity of the saturated alluvium. Total porosity measurements and nonsite-specific data are used to estimate the range and distribution of effective porosity (Bechtel SAIC Company, LLC, 2003f).

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 three parameters: the effective diffusion coefficient, the effective flowing interval spacing, and the flowing interval porosity. 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. For fractured volcanic tuffs, the largest variability in the effective diffusion coefficient is caused by differences in lithology (Bechtel SAIC Company, LLC, 2003f). The DOE analyses suggest selecting a model to determine the range of effective diffusion coefficients, based on measured porosities and permeabilities, that yields a range of values similar to the laboratory-derived values for Yucca Mountain tuffs (Bechtel SAIC Company, LLC, 2003f). Using this approach, DOE estimates a range of possible values for effective diffusion coefficients in volcanic tuffs from 10^{-8} to 10^{-6} cm^2/s [10^{-9} to 10^{-7} in^2/s]. To ensure the effective diffusion coefficient is not overestimated, the range is scaled down to account for ionic charge and size of ions not measured in the laboratory experiments. A cumulative distribution is calculated using the mean porosity and permeability values of relevant volcanic hydrostratigraphic units (Bechtel SAIC Company, LLC, 2003f). This approach reasonably encompasses the uncertainty of this parameter.

Flowing interval spacing and flowing interval porosity combine to produce the flowing interval aperture, an important uncertainty needed to calculate matrix diffusion. Smaller values for effective flowing interval spacing would result in predictions of more rapid matrix diffusion. Analyses are 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 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] (Bechtel SAIC Company, LLC, 2003f). This wide range of values, which is unchanged from the site recommendation, reasonably encompasses the uncertainty of flowing interval spacing and, given the highly fractured nature of the volcanic tuffs beneath Yucca Mountain, does not appear overly optimistic. It should be noted the effective flowing interval spacing is used only as a transport parameter that affects the rate of matrix diffusion; it does not affect modeled ground water fluxes or flow velocities. The flowing interval porosity probability distribution has been modified to incorporate new information from gas tracer tests in unsaturated tuff in the Exploratory Studies Facility (Bechtel SAIC Company, LLC, 2003f). The upper and lower bounds of the distribution remain the same at $\log_{10}$ values of -1.0 and -5.0 (Bechtel SAIC Company, LLC, 2003f), while the distribution shape is shifted to place more weight in the middle of the distribution range compared with the uniform distribution used in the site recommendation (Bechtel SAIC Company, LLC, 2003f).

Many of the parameters (e.g., the colloid partitioning coefficient, K_c) used in the models have very limited support from site characterization or laboratory data. As discussed in Section 5.1.3.9.4.2, 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 (Bechtel SAIC Company, LLC, 2003d,f). In the saturated zone, R_c is defined for the tuff aquifer on the basis of one site-specific field test and numerous laboratory tests; no field-scale site-specific data are available for the alluvial aquifer (Bechtel SAIC Company, LLC, 2003d). A combination of microspheres, silica, and natural montmorillonite colloids are used in the laboratory tests for both tuff and alluvium (Bechtel SAIC Company, LLC, 2003d). For fractured volcanic rocks, the probability distribution for R_c is determined by fitting the results of the field and laboratory tests to find the filtration rate constant, weighting the results to favor the field-scale tests, and truncating the lower end of the distribution (Bechtel SAIC Company, LLC, 2003d). The DOE rationale for truncating the distribution is that the model used to fit colloid breakthrough curves was insensitive to large variations in its fitting parameters for breakthrough curves with tails having low colloid concentrations. In such cases, nearly any value of R_c (including values approximately equal to 1) could produce a reasonable fit, although some intermediate value coupled with an appropriate filtration constant provided the best least-squares fit. Because the R_c values below 6 were the most poorly constrained, the distribution was truncated at this value. The final cumulative probability distribution has relatively arbitrary minimum and maximum values, which are selected primarily because of a lack of relevant data. The R_c probability distribution for saturated alluvium is constructed in a similar fashion except no site-specific field-scale tests are available. Instead, a field-scale study of bacteriophage attachment and detachment is used to constrain the distribution (Bechtel SAIC Company, LLC, 2003d). Again, the final cumulative probability distribution is large to accommodate a large degree of uncertainty, and the upper and lower bounds are not well constrained by supporting data (Bechtel SAIC Company, LLC, 2003d). Microspheres used in the testing had diameters between 190 nm [7.5×10^{-6} in] and 640 nm [2.5×10^{-5} in] (Bechtel SAIC Company, LLC, 2003d); these values are large compared with a typical size range in colloids from 1 to 450 nm [4×10^{-8} 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 agreed to provide additional justification for the use of microspheres as analogs for colloids (Reamer, 2000a) and has provided an update in Bechtel SAIC Company, LLC (2003d).

The K_c parameter, used to simulate reversible colloid attachment by lowering the radioelement K_d, is based on data from numerous laboratory experiments conducted with americium and plutonium (Bechtel SAIC Company, LLC, 2003a,h). These distributions are determined separately from the K_d distributions for the rock matrix. They are supported by experimental data and process modeling studies external to the DOE program (EPA, 1999; Honeyman and Ranville, 2002). The K_d values are represented by probability distributions for sorption onto smectite and iron oxyhydroxides (Bechtel SAIC Company, LLC, 2003h). Because both the sorption coefficient and the colloid concentration are sampled from parameter distributions, the K_c parameter also is sampled. Because of the potential for sorption onto the immobile rock matrix, repository performance is not sensitive to the parameters that control reversibly sorbed colloids, except at the highest ranges of the K_c range (high sorption coefficient, high colloid concentration). The concentration of wasteform colloids is determined from the in-drift colloid concentration model abstraction (Bechtel SAIC Company, LLC, 2003h), while the natural colloid concentrations, ranging from 0.001 to 200 ppm, are based on concentrations measured in wells from the Yucca Mountain vicinity (Bechtel SAIC Company, LLC, 2003a). The uncertainty in ground water colloid concentrations is represented using a cumulative distribution function based on field measurements from the saturated zone in the Yucca Mountain area and at the Idaho National Engineering and Environmental Laboratory (Bechtel SAIC Company, LLC, 2003h). DOE has not used any data, site-specific or nonsite-specific, to demonstrate 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 (Reamer, 2000a).

In summary, DOE uses stochastic approaches to identify and constrain data uncertainty in its model abstraction on radionuclide transport in the saturated zone. The uncertainty represented by the parameter distributions is based on a combination of laboratory and field data, supported by process modeling. In various cases, however, the technical bases for the parameter distributions used to describe data uncertainty are not 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 the NRC guidance (NRC, 1996) and its own quality assurance program. DOE agreed previously (Reamer, 2000a) to provide technical support demonstrating appropriate handling of data uncertainty, including sensitivity analysis.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.9.5), is sufficient to expect that the information necessary to assess radionuclide transport in the saturated zone with respect to data uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.9.4.4 Model Uncertainty

Sorption of radionuclides is modeled through the sorption coefficient, K_d, which is obtained assuming a linear isotherm relationship. Neglecting sorption processes has a significant effect on the estimated breakthrough of radionuclides from the saturated zone to the accessible environment (Bechtel SAIC Company, LLC, 2003g). For moderately sorbing radionuclides, such as neptunium, an assumption of zero sorption reduces the median saturated zone breakthrough time from greater than 10,000 to 705 years. It is reasonable to expect that some sorption will occur, especially in the porous saturated alluvium. An important controlling factor determining the value of sorption coefficients is the chemistry of ground water. Although the chemical variation of waters used in laboratory experiments to derive sorption values for alluvium is asserted by DOE to bracket the range of variation near Yucca Mountain, several factors, including the range of CO_2 and oxidation-reduction potential are insufficiently considered. Probability distributions for sorption coefficients are assumed to bound the entire range of geochemical conditions present in the saturated zone at Yucca Mountain. Temporal variations in geochemistry and the potential presence of complexing agents and microbial populations are said to be included within the distributions. Probability distributions for sorption coefficients also assume oxidizing conditions exist in saturated zone ground waters (Bechtel SAIC Company, LLC, 2003g). Recently, DOE used surface complexation modeling to investigate the effects of geochemistry and mineral surface area and to provide constraints on the parameter distributions (Bechtel SAIC Company, LLC, 2003f, Attachment I). The variability in geochemical conditions is appropriate (e.g., pH from 6 to 9), and the derived K_d distributions are bounded by the modeling results.

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 (Bechtel SAIC Company, LLC, 2003b,f). 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 also is asserted conservatively bounded by the K_d approach (Bechtel SAIC Company, LLC, 2003f).

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 (Reamer, 2000b). DOE provided results of a sensitivity analysis for saturated zone performance in recent documents (Bechtel SAIC Company, LLC, 2003f,g). Cases are considered in which radionuclide breakthrough is compared for matrix diffusion with matrix sorption, matrix diffusion with no-matrix sorption, and no-matrix diffusion. Results indicate that the no-matrix diffusion case produces shorter traveltimes, but the effect is mitigated by the presence of the saturated alluvium (Bechtel SAIC Company, LLC, 2003g). Matrix diffusion is more important for flow paths with intermediate and long travel times, while the saturated alluvium provides a significant delay for flow paths with short transport times (Bechtel SAIC Company, LLC, 2003g).

For saturated zone colloid transport modeling, DOE addresses model uncertainty chiefly by adopting each of two distinct attachment modes—reversible and irreversible (Bechtel SAIC Company, LLC, 2003a). DOE has used a limited amount of site-specific information to develop parameter distributions that reflect the uncertainty in colloid transport parameters. The colloidal transport model provides sensitivity studies that suggest colloid transport through the saturated zone is significant only for fast irreversible colloids not allowed to be retarded during fracture transport. The studies also show retardation of irreversible colloids provides significant delay in transport through the saturated zone (Bechtel SAIC Company, LLC, 2003g). The use of R_c to describe irreversible colloid transport through the saturated zone implicitly assumes colloid filtration and detachment rates are fast relative to ground water travel times. An analysis using Damköhler numbers (rate constant multiplied by representative residence times) indicates this assumption is valid for greater than 94 percent of the irreversible colloid mass (Bechtel SAIC Company, LLC, 2003d). However, that portion of the colloid mass assigned as fast irreversible colloids is not well supported by site characterization data, and there is no objective evidence the assigned values are bounding (Bechtel SAIC Company, LLC, 2003d,g).

DOE has not provided adequate justification for its selection of colloid transport parameters. Such analyses do not address adequacy of the model itself. DOE should show, for example, that neglect of kinetic adsorption and desorption effects will not result in an underestimate of the effects on performance of the reversible attachment. DOE agreed to obtain such data in the future (Reamer, 2000a). More generally, DOE agreed to perform sensitivity analyses on the importance of colloidal transport that will address, in part, adequacy of parameter uncertainty ranges to account for model uncertainty (Reamer, 2000a). DOE provided information to address the sensitivity analyses in Bechtel SAIC Company, LLC (2003a), which is currently in review.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.9.5), is sufficient to expect that the information necessary to assess radionuclide transport in the saturated zone with respect to model uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.9.4.5 Model Support

DOE applied a broad approach to supporting aspects of the conceptual model used in the saturated zone transport abstraction. Specific validation exercises include quantitative comparison of calculated traveltimes from the repository to the compliance boundary with those derived from field water chemistry data and qualitative comparison of calculated flow paths to flow paths suggested from analysis of field water chemistry data (Bechtel SAIC Company, LLC, 2003g). Additional confidence building is provided by comparison to analog sites, model data, and external publications.

Numerical results from the saturated zone transport model are compared to ground water traveltimes inferred from measurements of C-14 activity in water samples taken from wells along the predicted flow path (Bechtel SAIC Company, LLC, 2003g). This comparison indicates the median model calculated traveltime of 705 years is well within the C-14 estimated range of 84–3,600 years. Variation in transport parameters, such as flowing interval spacing, flowing interval porosity, and specific discharge indicates the range of uncertainties in these parameters produces traveltimes bounded by the C-14 estimate. Similarly, trends in the geochemical composition of saturated zone ground waters correspond to the direction and width of the inferred flow paths produced by the saturated zone abstraction (Bechtel SAIC Company, LLC, 2003g).

No site-specific field data are available to confirm sorption in the saturated alluvium (Bechtel SAIC Company, LLC, 2003f). Multiple laboratory studies have been conducted and the sorption model is conceptually supported based on evidence from other sites; however, there is not sufficient site-specific evidence to support the model. Single-well injection tracer tests conducted at the Alluvium Testing Complex provide confirmation of the single continuum approach to flow modeling in the alluvium (Bechtel SAIC Company, LLC, 2003f). Confidence in the conceptual model approaches also has been derived from analog studies at the Nevada Test Site. The Nevada Test Site studies focus on radionuclide transport during a scale of years to decades. Consistent with the conceptual model, those nuclides found to be most mobile in the Nevada Test Site studies are those assigned sorption coefficients of zero, including technetium and iodine (Bechtel SAIC Company, LLC, 2003g). Those radionuclides with high sorption coefficients, such as cesium and plutonium, are found relatively immobile. Migration of highly sorbing nuclides is associated with colloid-facilitated transport. Studies of transport in saturated alluvium, limited by time available for measurement, confirm at least the lower range of sorption coefficients for cesium, plutonium, and strontium (Bechtel SAIC Company, LLC, 2003g).

The available C-Wells Complex tracer test results provide strong evidence that matrix diffusion, matrix sorption, and colloidal transport occurs in the saturated volcanic tuffs along flow paths from Yucca Mountain (Bechtel SAIC Company, LLC, 2003g; CRWMS M&O, 2000a). The C-Wells information also provides qualitative information regarding diffusion of tracers with different effective diffusion coefficients and supports a fracture flow model use of dual porosity. Diffusion coefficients interpreted from the C-Wells test overlap the distribution range used in the saturated zone transport abstraction (Bechtel SAIC Company, LLC, 2003g).

Tests at the C-Wells Complex also are used to support the transport model for radionuclides irreversibly attached to colloids. The fractional recovery of microspheres observed during C-Wells testing, however, also could be affected by the settling process and colloid instability associated with the tracer mix. Although the filtration model is conceptually viable, there is little or no site-specific support for the parameter distributions used. Likewise, there is no site-specific field data available to support the reversible colloid model or its parameter distribution. Reversible colloid attachment is supported primarily by external literature.

Sorption breakthrough curves produced by the three-dimensional process-level transport model are used to evaluate performance of the simplified transport abstraction and the one-dimensional transport model used for calculating transport of decay chain radionuclides (Bechtel SAIC Company, LLC, 2003f). A constant mass input of radionuclide is applied at the unsaturated zone upstream boundary, and the predicted breakthrough of the abstraction is compared with the process-level model results. For both abstraction models, the breakthrough curves show excellent agreement with the site-scale model (Bechtel SAIC Company, LLC, 2003f). An exception occurs for earliest breakthrough of fast case transport when the abstraction slightly underpredicts breakthrough. In this case, the discrepancy originates from the differing timesteps used in the models, which affects the first timestep in the fast transport case. A second comparison of the abstraction model with the three-dimensional process model included a check of mass balance transported through the model domain. After a simulation of 100,000 years, 98.1 percent of the input mass had been transported through the abstraction model compared with 98 percent for the process-level model. Results of the model testing indicate the abstraction is appropriate for the range of uncertainty incorporated through the input parameters, and the abstraction functions as intended for both sorbing and non-sorbing species (Bechtel SAIC Company, LLC, 2003f)

The residence-time transfer function method for coupling matrix diffusion to the particle-tracking transport is compared with predictions from analytical solutions and other numerical models (CRWMS M&O, 2000f,g). 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). 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). 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 is also reported in CRWMS M&O (2000f). For the site recommendation, correct implementation of the saturated zone radionuclide transport abstraction is addressed by checking that model inputs are correctly selected, that parameter functions are calculated properly, that relationships between unsaturated zone and saturated zone outputs correctly reflect the intended saturated zone behavior (e.g., more-sorbing radionuclides are delayed relative to less-sorbing radionuclides), and that ingrowth of radioactive daughters is simulated (CRWMS M&O, 2000c, Figures 6-176 to 6-181). The verification exercises checked both the one-dimensional and three-dimensional transport models (Section 5.1.3.9.4.1), and included colloidal species.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.9.5), is sufficient to expect that the information necessary to assess radionuclide transport in the saturated zone with respect to model abstraction output being supported by objective comparisons will be available at the time of a potential license application.

5.1.3.9.5 Summary and Status of Key Technical Issue subissues and Agreements

Table 5.1.3.9-1 provides the status of all key technical issue subissues, referenced in Section 5.1.3.9.2, 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 review methods discussed described (Section 5.1.3.9.4). Note the status and the detailed agreements pertaining to all the key technical issue subissues are provided in Table 1.2-1 and Appendix A.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (e.g., specified testing or analyses) indicates that information necessary to begin a technical review will likely be available at the time of a potential license application.

Table 5.1.3.9-1. 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 through 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

Table 5.1.3.9-1. Related Key Technical Issue Subissues and Agreements (continued)			
Key Technical Issue	**Subissue**	**Status**	**Related Agreement***
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 review methods. Note: Key Technical Issue Agreement GEN.1.01 pertains to multiple integrated subissues, as well as some specific issues related to this integrated subissue.			

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5.1.3.10 Volcanic Disruption of Waste Packages

5.1.3.10.1 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) establishment of magma flow paths to the surface, and (v) effect of sustained magma flow on engineered barrier performance and possible waste package and high-level waste disaggregations. Transition to the Airborne Transport of Radionuclides Integrated Subissue (Section 5.1.3.11) 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 (Section 5.1.3.2). The relationship of this integrated subissue to other integrated subissues is depicted in Figure 5.1.3.10-1. The overall organization and identification of all the integrated subissues are depicted in Figure 1.1-2.

The DOE description and technical basis for the volcanic disruption of waste packages abstraction are summarized in Bechtel SAIC Company, LLC (2003a), and five supporting analysis and model reports (Bechtel SAIC Company, LLC, 2003b–f). 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. Because supporting analysis and model reports recently were provided by DOE, some revised topical areas that did not pertain to prior agreement issues were not reviewed in detail for this report.

5.1.3.10.2 Relationship to Key Technical Issue Subissues

The Volcanic Disruption of Waste Packages Integrated Subissue incorporates subject matter previously captured in the following seven 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)
- 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)

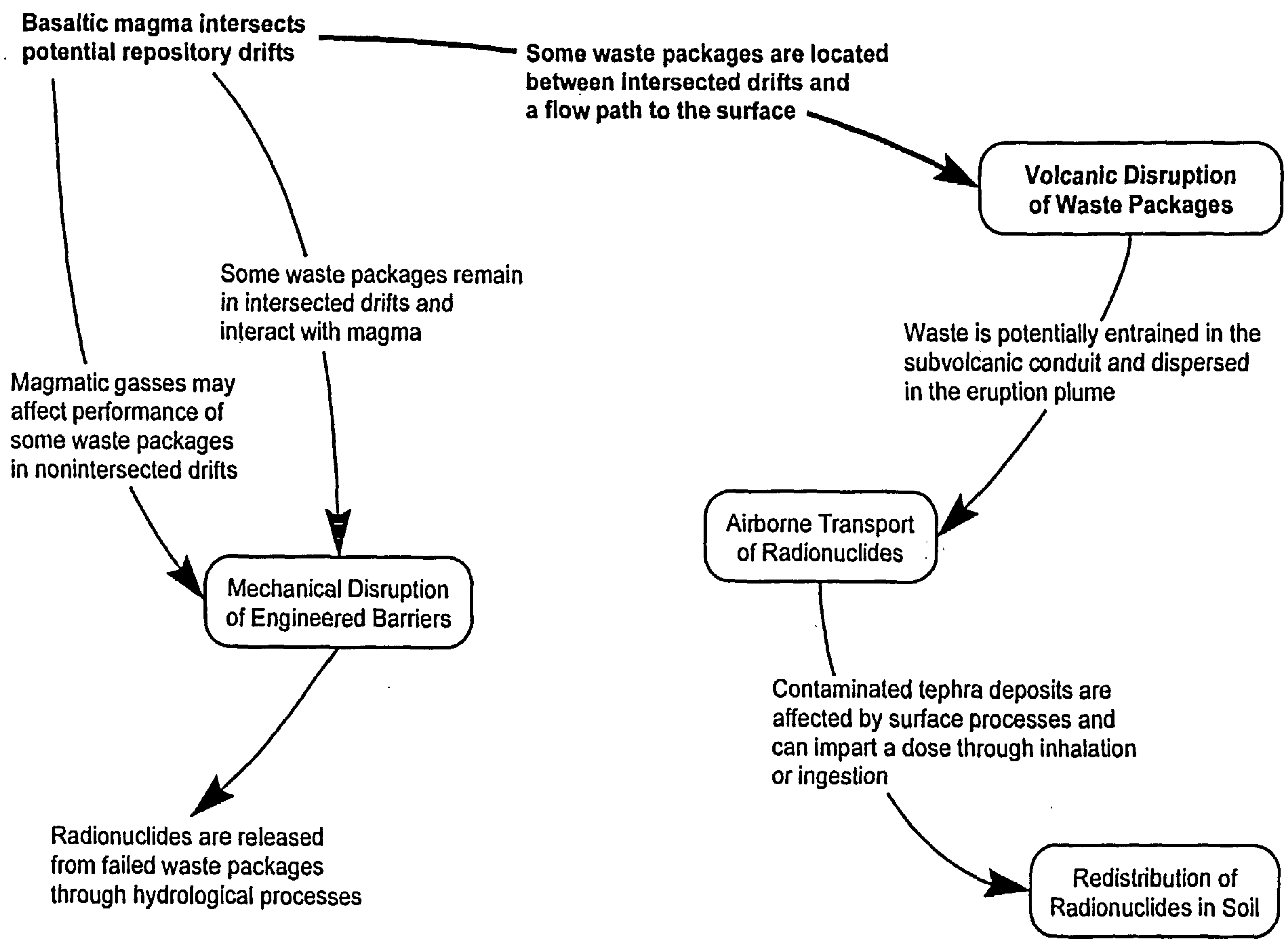

Figure 5.1.3.10-1. Diagram Illustrating the Relationship Between Volcanic Disruption of Waste Packages and Other Integrated Subissues. Material in Bold Is Identified in the Text.

- 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 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 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.

5.1.3.10.3 Importance to Postclosure Performance

Risk insights pertaining to volcanic disruption of waste packages indicate that the probability of igneous activity and the number of waste packages affected by an eruption are of high significance to waste isolation. The number of waste packages damaged by intrusion is assigned medium significance. The details of the risk insights ranking are provided in Appendix D. The probability of igneous activity is evaluated in Section 5.1.2.2, and the number of waste packages damaged by intrusion is evaluated in Section 5.1.3.2.

The DOE model results (Bechtel SAIC Company, LLC, 2002; CRWMS M&O, 2000a) indicate igneous activity is a natural process that could cause a significant number of waste package failures and thus result in a dose to the receptor during the regulatory period of interest. The NRC sensitivity analyses (Appendix D; Mohanty, et al., 2002) indicate the volcanic disruption of waste packages has a high significance to total system performance assessment results. This level of significance arises because the consequences from extrusive igneous activity (i.e., volcanism) are directly proportional to the number of waste packages intersected by a subvolcanic eruption conduit. Typical subvolcanic conduits are on the order of 5–50 m [16–164 ft] in diameter (e.g., Bechtel SAIC Company, LLC, 2003bc; NRC, 1999), although conduit diameters as large as 150 m [492 ft] may occasionally occur (Bechtel SAIC Company, LLC, 2003b). In addition, some physical conditions could potentially result in horizontal flow of magma along a drift, with a conduit forming some lateral distance away from the point of initial intrusion intersection (Woods, et al., 2002). Although conditions for this horizontal flow pathway now appear less likely to occur (Bechtel SAIC Company, LLC, 2003b), if this process occurred, it could affect a significantly larger number of waste packages than a simple vertical conduit. Damage to waste packages intersected by a subvolcanic conduit likely occurs from the high thermal and mechanical stresses created by a basaltic magma during an eruption (e.g., Bechtel SAIC Company, LLC, 2003c). Although detailed process models for these effects have not been developed, available information indicates the current waste package design would not provide the physical integrity necessary for waste isolation after direct entrainment in an erupting volcanic conduit (Bechtel SAIC Company, LLC, 2003c; NRC, 1999).

Although direct volcanic disruption has the potential to entrain and transport waste directly to the location of the reasonably maximally exposed individual, analyses used to demonstrate compliance with licensing requirements must factor the likelihood of a potential disruptive event into the performance calculations to determine a probability-weighted dose. As discussed in Section 5.2.2.2., most DOE estimates for the annual probability of igneous disruption at the potential repository site range from on the order of 10^{-10} to 10^{-8} (e.g., Bechtel SAIC Company, LLC, 2003f). In contrast, other annual probability estimates generally range from on order of 10^{-8} to 10^{-7} (e.g., NRC, 1999) to values as high as 10^{-6} using Bayesian methods (e.g., Ho and Smith, 1997). None of these probability models, however, has considered current uncertainties in the number and age of past igneous events (e.g., Hill and Stamatakos, 2002). Using a range of alternative conceptual models, Hill and Stamatakos (2002) describe how these uncertainties may have negligible to order of magnitude effects on the igneous activity probability estimate. Because the probability of igneous activity is directly proportional to the risk from potential igneous activity, these unaccounted for uncertainties may result in negligible to order of magnitude effects on current risk estimates for volcanic disruption of waste packages. The NRC staff is evaluating additional information provided in Ziegler (2003) to address current concerns regarding consideration of existing uncertainties in the DOE probability estimate.

5.1.3.10.4 Technical Basis

Basaltic magma can be thought of as a hot, pressurized fluid with higher viscosity than most other geologic fluids. Volcanoes form where magma rises from depth through a hydrofracture-type ascent process, which is controlled by the fluid pressure in the magma system and the distribution of stress in the surrounding rock (e.g., Lister and Kerr, 1991). Although the mechanisms of magma ascent are generally understood, local-scale variations in magma pressure or rock stress produce complexities in evaluating these ascent processes (e.g., Rubin, 1995, 1993). Introduction of subsurface engineered systems into a magma ascent pathway would further complicate the assessment of flow processes, due to significant perturbations in the distributions of ambient stress and fluid pressure. Based on independent analyses, the NRC staff has concerns subsurface repository systems could affect magma ascent processes and result in more deleterious effects than captured by the initial DOE models (Hill and Connor, 2000; NRC, 1999).

Some important staff concerns are addressed by DOE igneous activity models in CRWMS M&O (2000a). These concerns focus on the need to support previous assumptions regarding the limited potential for volcanic disruption of waste packages. Most importantly, these DOE analyses assume waste packages would fail if intersected by an erupting subvolcanic conduit and all contained high-level waste would be available for entrainment and subsequent atmospheric dispersal (CRWMS M&O, 2000a). Additionally, these DOE models include a significant reduction in high-level waste particle size during volcanic disruption, and all modeled eruptions were assumed to have violent strombolian dispersal characteristics (CRWMS M&O, 2000a). Nevertheless, preliminary models by Woods, et al. (2002), NRC (1999), and Woods and Sparks (1998), suggest the flow characteristics of magma into potentially intersected drifts could be more rapid and energetic than abstracted in CRWMS M&O (2000a). To address these concerns, DOE agreed to provide additional modeling support for magma-repository interactions, including evolution of potential magma flow paths through the duration of an igneous event (Reamer, 2001). Based, in part, on these agreements, the Volcanic Disruption of Waste Packages Integrated Subissue currently is considered closed-pending.

NRC developed a review plan (NRC, 2003) consistent with the acceptance criteria and review methods found in the previous issue resolution status reports. A status assessment of the DOE approaches for including volcanic disruption of waste packages in total system performance assessment abstractions is provided in the following subsections. This assessment is organized according to the five review methods identified in Section 2.3: (i) Model Integration (including system description), (ii) Data and Model Justification, (iii) Data Uncertainty, (iv) Model Uncertainty, and (v) Model Support.

5.1.3.10.4.1 Model Integration

Risk insights pertaining to the volcanic disruption of waste packages indicate the model for the number of waste packages entrained in a subsurface conduit makes the most significant contribution to risk calculations for possible radiological releases by extrusive volcanic processes. An important component of this model is the response of potential magma flow processes to the presence of subsurface drifts.

The current DOE approach to evaluating volcanic disruption of waste packages (Bechtel SAIC Company, LLC, 2003a) is derived from a series of conceptual models, which abstract a range of complex physical processes associated with potential subsurface igneous activity.

- Ascending basaltic magma intersects one or more subsurface drifts. Although the stress field around a drift may be perturbed by thermal-mechanical effects resulting from waste emplacement, DOE assumes vertical magma ascent always occurs, and the ascent pathway is unaffected by potential stress redistribution effects (Bechtel SAIC Company, LLC, 2003b).

- Magma flows into the intersected drifts caused by the pressure gradient between the intruding magma and the essentially atmospheric conditions in the drifts. Flow rates are on the order of 10 m/s [22 mi/hr] for magmas containing relatively low abundance of exsolved volatiles (Bechtel SAIC Company, LLC, 2003b).

- Most of the ascending magma is diverted into the intersected drifts, which completely fill with magma within approximately 5 minutes of initial intersection. Once the intersected drifts are filled, magma continues to rise along the initial vertical plane of ascent (Bechtel SAIC Company, LLC, 2003b).

- 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 (Bechtel SAIC Company, LLC, 2003a,b).

- All waste packages directly intersected by the subvolcanic conduit are assumed to fail from the adverse mechanical and thermal conditions in the erupting conduit (Bechtel SAIC Company, LLC, 2003c).

Current DOE models account for more physical processes than were considered in previous DOE models (e.g., CRWMS M&O, 2000a,b) in response to agreements reached between NRC and DOE on potential magma-drift interactions (Reamer, 2001). For example, the DOE volcanic disruption of waste packages model no longer relies on the presence of debris plugs to restrict magma flow in potentially intersected drifts (CRWMS M&O, 2000a,b). In contrast, current DOE models (Bechtel SAIC Company, LLC, 2003b) evaluate potential flow conditions for drifts containing waste packages or piles of rubbly backfill at the ends of intersected drifts. These models assume the magma behaves as an incompressible fluid with Newtonian behavior, which appears consistent with the degassing processes thought to occur during the initial stage of potential magma ascent (Woods, et al., 2004; Bechtel SAIC Company, LLC, 2003b). Using these assumptions, the DOE models conclude a potentially intersected drift would fill with magma in 1–5 minutes following intersection (Bechtel SAIC Company, LLC, 2003a,b). Similar results are obtained by Lejeune, et al. (2002) using experimental analogs for flow of degassed magma into open drifts.

The possible emplacement of high-level waste in potential repository drifts at Yucca Mountain will release heat into the surrounding rock. Although active ventilation is planned to remove much of the heat before repository closure (Bechtel SAIC Company, LLC, 2003e), some heating of drift walls will occur following planned closure. This heating will likely result in thermal-mechanical effects on the surrounding rock, which can affect the magnitude and orientation of crustal stress surrounding the drift. Current DOE analyses evaluate more realistic physical conditions and coupled thermal-mechanical processes than were considered in earlier models (e.g., CRWMS M&O, 2000b). These current analyses indicate potential stress redistribution effects from heating will be restricted to within approximately 10 m [33 ft] of the drifts, and significant variations in wall-rock properties (e.g., fracture density and orientation) will likely result in complex stress redistribution patterns (Bechtel SAIC Company, LLC, 2003e). Independent analyses (Smart, 2004) also suggest these DOE models may underestimate the amount of strain that can be accommodated by wall rock around a drift, which would further reduce the magnitude of potential stress redistribution because of heating. Although the theoretical deflection of ascending dikes away from heated drifts is sometimes alluded to by DOE (e.g., Bechtel SAIC Company, LLC, 2003a), current DOE models for potential magma-repository interactions assume any rising magma will not be deflected away from potential repository drifts because of thermal-mechanical effects from possible waste emplacement (Bechtel SAIC Company, LLC, 2003b).

The processes that control the initial development of a subvolcanic conduit are poorly known. A common observation at basaltic scoria 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 style. A central vent then localizes along the fissure, with the eruption becoming more energetic and forming a dispersive scoria cone volcano (e.g., Fedotov, et al., 1984; Thorarinsson, et al., 1973). One explanation for this process is that a preferred vertical-flow pathway develops in 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 (Bruce and Huppert, 1990, 1989; Huppert and Sparks, 1985; Delaney and Pollard, 1982). 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 because calculations indicate magma will accelerate into the intersected drifts because of decompression effects (e.g.,Woods, et al., 2002; CRWMS M&O, 2000b; Woods and Sparks, 1998). Thus, streamlines for magma in the intersecting dike could focus on the drifts, with lower ascent velocities or possibly stagnation occurring in pillars between the drifts. The effect of focusing the vertical ascent of magma toward drifts may localize subsequent conduit formation in the drift.

Magma flow into potential repository drifts is evaluated in Bechtel SAIC Company, LLC (2003b), using a three-dimensional model that couples vertical flow of magma in a narrow dike to simplified rock-mechanical relationships. This model considers magma flow as a simple mass-transport process, with isotropic horizontal rock stress equivalent to half the gravitational (i.e., vertical) stress. Using these relationships and dike widths of 0.25–0.45 m [0.8–1.5 ft], Bechtel SAIC Company, LLC (2003b) concludes magma-flow streamlines would focus on the intersected drifts. These models also indicate magma in the pillars between drifts would possibly rise tens of meters above the level of the drifts, which is interpreted to favor the subsequent formation of conduits in pillars rather than at drifts (Bechtel SAIC Company, LLC, 2003a,b). The staff note other models in Bechtel SAIC Company, LLC (2003b) conclude the time necessary to fill a potentially intersected drift is shorter (i.e., 1–5 minutes) than the time calculated by the three-dimensional hydromechanical model as necessary to attain flow equilibrium into the drifts (i.e., 17 minutes). These models indicate potentially intersected drifts may fill with magma more rapidly than would allow magma to ascend and attain an equilibrium height above the drift. Nevertheless, independent analyses (Lejeune, et al., 2002; Woods, et al., 2002) support the basic conclusion of relatively rapid inflow of magma into potentially intersected drifts. The short amount of time necessary to completely fill a potentially intersected drift, however, does not appear sufficient to induce significant cooling effects in areas of low vertical velocity in the dike. Thus, conduit localization may not be affected by the transient effects of rapid magma flow into intersected drifts. Nevertheless, DOE does not use the results of these models to calculate the likelihood of conduit formation along a dike that potentially intersects a subsurface drift (Bechtel SAIC Company, LLC, 2003b,f).

The current DOE model for conduit formation is derived from statistical simulations of randomized conduit localization along a dike, empirical observations from Yucca Mountain region volcanoes, and an abstracted assumption regarding conduit localization on a drift (Bechtel SAIC Company, LLC, 2003f). Results of these analyses are convolved into a probability distribution function for the number of conduits forming within the potential repository, given a potential repository-penetrating subsurface igneous event. This approach, however, appears to be inconsistent with the methods used by DOE to calculate the probability of volcanic disruption. Bechtel SAIC Company, LLC (2003f) derives a 1.3×10^{-8}/yr average probability of new volcano formation at the potential repository site, based on a 1.7×10^{-8}/yr average probability of subsurface intrusion intersection from models in CRWMS M&O (1996). By definition (Bechtel SAIC Company, LLC, 2003f; CRWMS M&O, 1996), a volcanic event includes formation of a subsurface magma conduit. The probability distribution function for the number of conduits forming within the potential repository, however, includes a 22-percent average likelihood that no volcanic conduit will form during a simulated volcanic event (i.e., Bechtel SAIC Company, LLC, 2003f, p. 107). The probability distribution function for conduit formation should have a minimum value of one conduit, not zero conduits, because the event probability used by DOE already is conditional on one volcanic conduit forming.

Previous DOE models do not provide an adequate technical basis to conclude potential magma ascent would remain localized in a single vertical intrusion following possible drift intersection (CRWMS M&O, 2000b). Based on potentially significant variations in bedrock thicknesses over drifts, and the distribution of possible rock fractures, Woods, et al. (2002) hypothesize magma might emerge from a drift along a different vertical pathway than used to ascend from depth. This hypothesis is supported by a simple mechanical model that assumes the pressure needed to dilate an existing vertical fracture is a function of the overlying lithostatic load (e.g., NRC, 1999). Bedrock thicknesses overlying the potential repository range from 200 to 300 m [656 to 984 ft]. Assuming 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 could 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] lithostatic load during ascent. If the drift fills with magma and begins to repressurize, hydrofracturing and breakout through the drift roof might be 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, 2000b). In this situation, magma could flow horizontally through the drift between the initial intersection point and the final breakout point, potentially entraining more waste packages than intersected by a simple vertical conduit.

DOE provides additional detailed analyses in Bechtel SAIC Company, LLC (2003b) to evaluate thermal-mechanical processes associated with potential breakouts from magma-filled drifts at locations away from the point of initial intrusion intersection. These analyses compare rates of dike-tip propagation along a vertical fracture of original intersection and rates for vertical fractures located away from the point of initial intersection. Fracture propagation rates are evaluated using simplified rock mechanical models for hydrofracture processes, which appear to reasonably abstract magma propagation processes. Bechtel SAIC Company, LLC (2003b) uses these analyses to conclude that if an intersected drift is wholly filled with magma, ascent rates along the original vertical plane of intersection would be approximately twice as rapid as along other vertical planes located elsewhere along the drift. The original plane of magma ascent is favored because lower effective fluid pressures result if magma has to travel horizontally along the drift, relative to continued vertical ascent. Bechtel SAIC Company, LLC (2003b) concludes the horizontal "dog-leg" scenario of Woods, et al. (2002) appears unlikely, relative to the scenario of continued vertical magma ascent along the original plane of intersection. Although a quantitative reduction in scenario likelihood is not specified in Bechtel SAIC Company, LLC (2003b), the analyses in Bechtel SAIC Company, LLC (2003b) support the conclusion for a relatively lower likelihood of occurrence for the horizontal "dog-leg" scenario of Woods, et al. (2002).

DOE concludes the combined thermal and mechanical effects resulting from potential exposure to basaltic magma in an erupting volcanic conduit are sufficient to damage waste packages to the extent that no further protection to the wasteform is provided (Bechtel SAIC Company, LLC, 2003c). In addition, the processes of waste package disaggregation are sufficiently energetic to induce breakage of the waste into particles having average diameters of 0.02 mm [8×10^{-4} in] (Bechtel SAIC Company, LLC, 2003a; CRWMS M&O, 2001). Available information supports the disaggregation of waste packages and associated waste upon entrainment in an erupting volcanic conduit (NRC, 1999).

In summary, DOE considers available information sufficient to conclude that if basaltic magma was to rise beneath the potential repository site, this magma could intersect one or more drifts. Because of the pressure differences between potential magma and drifts, magma would flow into the drifts until the drifts were filled. Magma would then most likely continue to rise along the original plane of vertical ascent and reach the surface at Yucca Mountain. This potential subsurface magma system could then localize a subsurface conduit in a manner similar to conduit localization at other basaltic scoria cone eruptions. Widening of the conduit through time would likely entrain waste packages, with the number of entrained waste packages determined by the total diameter of the conduit. Models for subsurface magma ascent and flow processes account for the general physical processes likely to occur during basaltic igneous events characteristic of the Yucca Mountain region. In addition, the DOE models evaluate changes in potential subsurface magma flow processes that are likely to result if subsurface repository structures are encountered. Based on this abstraction, the current DOE modeling approach appears to be a reasonable general representation of potential basaltic magma ascent and drift-interaction processes.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.10.5), is sufficient to expect that the information necessary to assess possible volcanic disruption of waste packages with respect to system description and model integration will be available at the time of a potential license application.

5.1.3.10.4.2 Data and Model Justification

Risk insights pertaining to the volcanic disruption of waste packages indicate the model for the number of waste packages entrained in a subsurface conduit makes the most significant contribution to risk calculations for possible radiological releases by extrusive volcanic processes. Abstraction of this model relies on accurate characterization of past igneous events in the Yucca Mountain region, and evaluation of possible changes in igneous characteristics resulting from complex interactions with engineered systems. Because there are few analogs for the effects of potential igneous events on engineered systems, abstraction of the performance assessment model will necessarily rely on indirect information.

Data on the characteristics of past basaltic igneous events in the Yucca Mountain region are developed primarily in Bechtel SAIC Company, LLC (2003d). Few of these data needed to support model abstraction are directly available from measurements of basaltic igneous features in the Yucca Mountain region because these models rely on data representing active igneous systems in the subsurface. Thus, important model parameters such as magma temperatures and ascent velocities must be derived from analog information or physical process models.

Conceptual models for potential interactions between rising basaltic magma and potential repository drifts are most sensitive to assumptions regarding the pressure and ascent rates of magma in an intrusion (e.g., Bechtel SAIC Company, LLC, 2003b). 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., Woods, et al., 2002). In contrast to previous models (CRWMS M&O, 2000b), current DOE models consider a range of magma overpressures that extend from slightly below to greater than twice the lithostatic pressure {i.e., to 15 MPa [2,176 psi] at 300-m [984-ft] depth}. Although magma pressure is not measured directly during an igneous event, this pressure is often calculated as 1–10 MPa [145–1,450 psi] greater than lithostatic pressure for shallow dikes (Rubin, 1993; Rogers and Bird, 1987; Delaney, et al., 1986). The pressure data used in Bechtel SAIC Company, LLC (2003b) appear consistent with a general understanding of fluid pressures in shallow basaltic magma systems.

Magma ascent rates are governed by pressure in the magma system and the magnitude and orientation of crustal stresses (e.g., Woods, et al., 2004; Bechtel SAIC Company, LLC, 2003b; La Femina, et al., 2003). Few data, however, are available to constrain likely magma ascent rates during basaltic igneous events. Fedotov, et al. (1976) reports seismic data for the 1975 Tolbachik, Russia, eruption that indicate an approximately 0.04-m/s [0.13-ft/s] magma ascent rate. Depths of initial earthquakes for the 1999 Cerro Negro, Nicaragua, eruption suggest magma ascent rates on the order of 2 m/s [6.6 ft/s] to the start of the eruption (La Femina, et al., 2003). Bechtel SAIC Company, LLC (2003d) uses a generalized model to calculate magma ascent rates on the order of 1–10 m/s [3.3–33 ft/s] as representative of potential basaltic igneous events in the Yucca Mountain region. Although the lower bound of this rate is higher than suggested by data in Fedotov, et al. (1976), any possible overestimation of ascent rates appears to result in more rapid magma flow into potentially intersected drifts. This effect would likely reduce the magnitude of possible cooling effects on renewed magma ascent during the initial stage of a potential igneous event (i.e., Bechtel SAIC Company, LLC, 2003b). However, lower ascent rates may favor the ascent of secondary dikes relative to ascent along the main dike. Because of limited data are available on magma ascent rates for basaltic igneous systems, and the possible effects of lower ascent velocities are not clear, the staff recommend DOE provide additional support for the magma ascent rates used in Bechtel SAIC Company, LLC (2003d).

Data appear sufficient to characterize basaltic igneous events at the level of detail necessary to support models for potential volcanic disruption of waste packages. Chemical and mineralogical compositions of magmas are derived from basaltic volcanoes in the Yucca Mountain region (Bechtel SAIC Company, LLC, 2003d). These data also are used to derive basic physical properties of basaltic magmas, such as temperatures and viscosities, which are the bases for model parameters in Bechtel SAIC Company, LLC (2003b). The volatile content of basaltic magma is an important parameter for several models in Bechtel SAIC Company, LLC (2003b–d) related to volcanic disruption of waste packages. Water is the most abundant magmatic volatile, although carbon dioxide can have important effects on magma vesiculation and ascent processes (e.g., Sparks, et al., 1994). Using a range of experimental data for basaltic magmas throughout the world, Bechtel SAIC Company, LLC (2003d) concludes water contents for Yucca Mountain region basalt are most likely in the range of 1–3 wt%, with abundances of 0–1 and 3–4 wt% having lower likelihoods of occurrence. Direct investigations on basalt in the Yucca Mountain region, however, show magmatic water contents are approximately 4 wt% (Nicholis and Rutherford, 2004; Luhr and Housh, 2002). Thus, models in Bechtel SAIC Company, LLC (2003b–d) do not appear to account for water contents representative of the Yucca Mountain basaltic magmas. An underestimation of magmatic water or total volatile contents will likely affect results of models for magma ascent and flow processes, because the decompression-induced expansion of volatiles will increase magma flow rates and enhance magma fragmentation effects. In addition, the presence of a significant volatile fraction may affect key DOE assumptions for modeling magma as an incompressible fluid (Bechtel SAIC Company, LLC, 2003b) or for the production of pyroclastic ejecta at older volcanoes (e.g., Bechtel SAIC Company, LLC, 2003d). Thus, additional justification appears needed to support the water contents used by DOE to model basaltic igneous processes in the Yucca Mountain region.

Following potential intersection of drifts by an igneous intrusion, magma is thought to likely ascend to the surface after the intersected drifts are filled by magma (e.g., Bechtel SAIC Company, LLC, 2003b). Magma flow will likely focus on a single vertical conduit to the surface in response to complex interrelationships between rock stress, magma flow dynamics, and cooling of the intrusion (e.g., Bruce and Huppert, 1989). Once established, this subvolcanic conduit will likely widen during the course of the ensuing volcanic eruption through a general process of wall-rock erosion (e.g., Bechtel SAIC Company, LLC, 2003d; Valentine and Groves, 1996). In performance assessment models, the diameter of the subvolcanic conduit determines the number of waste packages potentially entrained into the erupting magma. DOE uses conduit diameters of 1–150 m [3–492 ft] measured from a range of analog volcanoes that are thought to have eruptive volumes comparable to past eruptions in the Yucca Mountain region (Bechtel SAIC Company, LLC, 2003d). These data appear to be reasonable representations of possible subvolcanic conduit diameters for Yucca Mountain region basaltic volcanoes and support the conceptual model for conduit development during a potential basaltic volcanic eruption. The DOE models also address the possibility that more than one volcanic conduit could develop during a potential igneous event, based on an interpretation of Yucca Mountain region volcano characteristics in CRWMS M&O (1996).

In summary, sufficient data appear to be available to support the DOE conceptual models for the number of waste packages disrupted during potential igneous events. The characteristics of potential igneous events in the Yucca Mountain region appear to be reasonable interpretations of available data by DOE. Data for some important DOE conceptual models, such as subvolcanic conduits, are not directly available from observations at Yucca Mountain region volcanoes. In these instances, the technical bases used by DOE to develop information in support of these models appear traceable to analog information or interpretations of documented experimental investigations. These technical bases appear sufficient to support the DOE model abstractions for evaluation of the number of waste packages disrupted during potential igneous events.

While some information on possible volcanic disruption of the waste package with respect to data being sufficient for model justification may be available at the time of a potential license application, the staff is currently of the view that DOE should provide additional information on the DOE determination on how consideration of magma with higher volatile content (approximately 4 wt% water) may affect models of magma ascent and flow processes.

5.1.3.10.4.3 Data Uncertainty

Risk insights pertaining to the volcanic disruption of waste packages indicate the most important data uncertainty needs relate to support for models that evaluate the number of waste packages potentially entrained during a potential extrusive volcanic event. The number of waste packages directly intersected by a basaltic subvolcanic conduit is calculated using a range of conduit geometries derived from analog volcanoes (Bechtel SAIC Company, LLC, 2003d). This parameter range appears reasonable based on similarities in interpreted eruption characteristics between the analog volcanoes and volcanoes in the Yucca Mountain region (Bechtel SAIC Company, LLC, 2003d) and the independent observations at other analogous volcanoes (NRC, 1999; Doubik and Hill, 1999). The DOE models address the possibility that more than one volcanic conduit could develop during a potential igneous event. The range of potential conduits (0–13) is based on a general interpretation of Yucca Mountain region volcano characteristics in CRWMS M&O (1996) and Bechtel SAIC Company, LLC (2003d). DOE calculates the number of waste packages entrained during a potential volcanic event by multiplying the area of a potential conduit by the average waste package density in the repository footprint (Bechtel SAIC Company, LLC, 2003a), which is then multiplied by the number of conduits that form in each sampled event.

Although this DOE approach captures the uncertainties in the number and size of potential subvolcanic conduits, use of an average waste package density does not appear to capture the uncertainty in this value inherent in the conditional probability used by DOE to represent a volcanic event. Current DOE probability distributions account for localization of the subvolcanic conduit directly on a drift during half of all realizations (Bechtel SAIC Company, LLC, 2003f), with randomized conduit formation along the originating intrusion during the remainder of the realizations. Use of an average waste package density, however, essentially results in a randomized conduit location for each realization. This approach appears inconsistent with the event probability, in which half of all realizations should localize on a drift. For currently proposed designs, a drift has a higher waste package density per unit area than the average density of the entire repository, thus, the current DOE approach appears to underestimate the uncertainty in the number of waste packages potentially intersected by a subvolcanic conduit.

For the DOE process models relevant to evaluating the number of waste packages entrained during potential volcanic events, many of the important data ranges are derived from information collected at, or interpreted from, Yucca Mountain region volcanoes. Information on subsurface intrusion geometries is derived primarily from CRWMS M&O (1996), which accounts for a range of judgments regarding potential characteristics of these subsurface features. The uncertainties in these characteristics recommended for use in performance calculations (e.g., Bechtel SAIC Company, LLC, 2003a) appear consistent with the underlying data. Models for magma ascent and subsurface flow processes sample a range of physical parameters that generally accounts for expected variabilities and uncertainties in the Yucca Mountain region (Bechtel SAIC Company, LLC, 2003b–d), revised and updated in response to agreements reached between NRC and DOE on potential magma-drift interactions (Reamer, 2001).

All waste packages entrained in a potential volcanic conduit are assumed by DOE to be damaged to the extent that waste in these packages is fragmented and dispersed into the erupting magma (Bechtel SAIC Company, LLC, 2003c). This assumption is supported by data indicating a lack of resiliency for waste packages encountering the thermal and mechanical stresses characteristic of erupting basaltic volcanoes (Bechtel SAIC Company, LLC, 2003c). The uncertainties in these data are encompassed by the conservative assumption that waste packages are completely destroyed upon entrainment into an erupting subvolcanic conduit. DOE also concludes the thermal and mechanical stresses in an erupting subvolcanic conduit would be sufficient to fragment the entrained high-level waste to particles that range in diameter from 0.001 to 0.5 mm [4×10^{-5} to 2×10^{-2} in] (CRWMS M&O, 2001). This size distribution reasonably accounts for the uncertainty in wasteform response to basaltic volcanic conditions (NRC, 1999), as potentially all the entrained waste is available for subsequent airborne transport calculations.

In summary, most data ranges derived from basaltic igneous systems appear to adequately represent the uncertainty and variability in the characteristics of potential future igneous events in the Yucca Mountain region. Although alternative interpretations to some of the data ranges can be derived, the technical basis used by DOE is sufficiently transparent to permit independent review and evaluation of the possible significance of alternative interpretations of data uncertainties. Uncertainties in data supporting the DOE models for waste package and wasteform response to basaltic volcanic conditions appear adequately considered through the use of reasonably conservative assumptions regarding likely fragmentation during entrainment in an erupting subvolcanic conduit.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.10.5), is sufficient to expect that the information necessary to assess possible volcanic disruption of waste packages with respect to data uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.10.4.4 Model Uncertainty

Risk insights pertaining to the volcanic disruption of waste packages indicate the most important model uncertainty needs relate to support for conclusions regarding the pathways magma might take to the surface upon potential interaction with subsurface drifts. The current DOE model for the initial ascent of basaltic magma (Bechtel SAIC Company, LLC, 2003b) evaluates the vertical fracturing and dilation of country rock through elastic strain induced by the flow of an incompressible Newtonian fluid (i.e., magma). There are several important uncertainties in this modeling approach, which are addressed by Bechtel SAIC Company, LLC (2003b). Basaltic magma is assumed to be an incompressible Newtonian fluid during magma ascent. However, magmas within several kilometers of the surface will contain some fraction of gas bubbles as well as solid crystals. Although these phases will result in model uncertainties because of compressibility effects and non-Newtonian flow processes (e.g., Woods, et al., 2004), most models for magma ascent do not account for these uncertainties (e.g., Rubin, 1995). The DOE model for magma ascent also assumes wall rock is a homogeneous material with elastic strain response. Realistically, subsurface rock does not have homogeneous properties, and strain response is not purely elastic (Bechtel SAIC Company, LLC, 2003b). Nevertheless, fracture mechanics models commonly assume a homogeneous, elastic strain material to produce tractable models (e.g., Woods, et al., 2004; Rubin, 1995). The DOE magma ascent models also assume flow rates and magma viscosities are sufficiently low to stay in a laminar flow regime, which allows several useful approximations to be used in the numerical models (Bechtel SAIC Company, LLC, 2003b). These assumptions are reasonable and commonly used in magma ascent models (e.g., Woods, et al., 2004). Although current DOE models do not explicitly account for these uncertainties, DOE performs numerous model calculations that sample a reasonable range of important model parameters such as magma ascent velocity, temperature, and viscosity. Current DOE models for magma ascent represent assumptions commonly used to produce tractable numerical models. This approach was developed in response to agreements reached between NRC and DOE on potential magma-drift interactions (Reamer, 2001) and is a significant improvement to the models in CRWMS M&O (2000b), which did not explicitly evaluate important uncertainties in magma ascent processes.

DOE evaluates several alternative conceptual models for the ascent of magma by a hydraulic fracturing mechanism. These models use well-established analytical solutions to hydrofracture mechanics formulae (Bechtel SAIC Company, LLC, 2003b). Although these alternative conceptual models provide useful insights on fracture mechanics processes, these models do not appear suitable for use in fluid ascent models that involve subsurface withdrawal of fluid from the fracture system (i.e., magma flow into a drift). In addition, the details of some models are proprietary and thus not available for review. DOE concludes there are no alternative conceptual models that would provide a significantly different result than currently determined by Bechtel SAIC Company, LLC (2003b). The conclusion appears reasonable, as the models in Bechtel SAIC Company, LLC (2003b) indicate that magma ascending directly beneath a drift will intersect that drift.

The current DOE model for magma flow into potentially intersected drifts (Bechtel SAIC Company, LLC, 2003b) represents a complete revision of the models presented in CRWMS M&O (2000b). Previously, the DOE models relied on the formation of debris plugs to restrict the flow of magma into potentially intersected drifts (CRWMS M&O, 2000b). Current DOE models evaluate potential magma flow into drifts containing waste packages with uncertainties related to interactions between the waste packages and free-flowing magma. Because the potential flow of magma into drifts is a complex process, DOE uses insights from several types of numerical models to evaluate the possible effects of model uncertainty on the performance assessment. The most potentially significant uncertainty in these models involves the assumption that the magma is sufficiently degassed so rapid expansion of gas does not cause acceleration of the flow during decompression. By assuming magma is degassed, models in Bechtel SAIC Company, LLC (2003b) conclude flow rates into potentially intersected drifts will be on the order of 10 m/s [33 ft/s]. This result is consistent with experimental models developed by Lejeune, et al. (2002) for degassed magma. In contrast, fully coupled gas-magma models in Woods, et al. (2002) indicate a completely nondegassed magma could flow into potentially intersected drifts with velocities on the order of 100 m/s [328 ft/s]. Models in Woods, et al. (2002) use a series of simplifying assumptions to evaluate the possible effects of flow-induced shocks on waste package performance, which might result if decompression-induced magma flow is accompanied by high flow velocities down the drift. These models conclude that even if flow-induced shocks were to develop under optimized conditions, the magnitude of the shock overpressures are significantly below the strength of an intact waste package (Woods, et al., 2002). Although the uncertainty in extent of magma degassing will affect important model results for magma flow velocities, the uncertainties in these velocities do not appear to affect risk calculations significantly.

Current DOE models (Bechtel SAIC Company, LLC, 2003b) conclude if magma intersects subsurface drifts, magma ascent will essentially stop at or slightly above the level of the potentially intersected drifts. This condition occurs because the modeled flux of ascending magma is effectively captured by flow into the potentially intersected drifts. After a potentially intersected drift fills with magma, repressurization in the intrusion system will cause the magma to renew ascent along the original vertical plane of intersection. This conclusion is supported by application of the same hydrofracture model as used to evaluate initial ascent of magma. Uncertainties in this model are evaluated through reasonable variations in basic model parameters.

An alternative conceptual model for magma ascent at a location away from the point of initial drift intersection is proposed by Woods, et al. (2002). This model is based on consideration that topographic variations above a drift could potentially result in stress conditions more favorable for magma ascent at a location away from the point of initial drift intersection. The significance of this alternative model is horizontal flow paths along a drift could be significantly longer than 150 m [492 ft], which is the maximum diameter of subvolcanic conduits, and thus entrain more waste packages than modeled by a simple vertical conduit. DOE provides extensive evaluation of this alternative conceptual model in Bechtel SAIC Company, LLC (2003b). This evaluation uses the same hydrofracture model used to evaluate the initial ascent of magma. This model concludes that if a fracture were to occur at the distal end of a potentially intersected drift, magma could ascend through this fracture once the drift was filled. However, magma also would continue to simultaneously rise along the original plane of intersection. The rate of magma ascent along the distal fracture would be no more than half the ascent rate as modeled along the original plane of ascent because magma in the distal fracture is supported only by magma in the potentially intersected drift. In contrast, magma in the original vertical fracture is supported by magma from depth, which gives a larger effective fluid pressure to dilate the fracture. Thus, magma modeled along the distal fracture will likely cool more rapidly, and ascend much more slowly, than along the original plane of ascent. These effects will cause the magma along the original ascent fracture to reach the surface well before magma along the distal fracture can ascend far from the drift. The DOE model concludes magma ascent to the surface along a distal fracture appears highly unlikely, relative to continued ascent along the original vertical fracture. Based on the mechanical analysis presented in Bechtel SAIC Company, LLC (2003b), conditions for this alternative conceptual model currently appear less likely to occur than conditions for the model of continued ascent along the original plane of ascent.

In summary, uncertainty in the underlying DOE conceptual models appears adequately considered in the evaluation of potential volcanic disruption of waste packages. Although the model uncertainties are not quantified, the results of these uncertainties are used by DOE to support reasonable conclusions regarding rapid magma flow into intersected drifts and renewed ascent to the surface following potential drift intersection. In addition, DOE appears to have considered an appropriate range of alternative conceptual models that are currently available. Although these alternative conceptual models could possibly reduce the potentially adverse effects of volcanic disruption of waste packages, results of these alternative conceptual models apparently are not used to reduce these effects in the DOE performance assessment calculations.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.10.5), is sufficient to expect that the information necessary to assess possible volcanic disruption of waste packages with respect to model uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.10.4.5 Model Support

Risk insights pertaining to the volcanic disruption of waste packages indicate the most important information needs for model support relate to conclusions regarding the pathways magma might take to the surface after potential interaction with subsurface drifts. The significant model abstraction for volcanic disruption of waste packages is the determination of the number of waste packages potentially entrained by a subvolcanic conduit. The generalized DOE model abstraction is that rising basaltic magma can intersect a subsurface drift, fill the drift with magma, then resume ascent to the surface (i.e., Bechtel SAIC Company, LLC, 2003b). Once magma reaches the surface, subsurface flow localizes along a preferred vertical pathway and forms a conduit. This subvolcanic conduit can widen during the eruption to diameters of up to 150 m [492 ft] and potentially entrain waste packages intersected by the conduit. Waste packages entrained in an erupting subvolcanic conduit break apart and release fragmented high-level waste into the erupting magma where it is available for airborne transport in the eruption column.

Current DOE models for magma ascent and conduit formation (Bechtel SAIC Company, LLC, 2003b) appear generally consistent with models in the literature for these processes at other basaltic volcanoes (e.g., Doubik and Hill, 1999; Rubin, 1995; Bruce and Huppert, 1989; Delaney and Pollard, 1982). There is no known analogy, however, for the potential subsurface interaction between basaltic magma and engineered systems analogous to the potential repository at Yucca Mountain. Thus, the DOE models will need to be supported by comparison with detailed process models, rather than by comparison with empirical observations at analog systems.

The DOE model for subvolcanic conduit development is supported by several detailed process-level models, developed and revised in response to agreements reached between NRC and DOE on potential magma-drift interactions (Reamer, 2001). The process model for magma ascent uses detailed rock mechanics and fluid flow concepts to evaluate the general relationships between magma ascent and wall-rock strain (Bechtel SAIC Company, LLC, 2003b). This model supports the abstraction that magma rising beneath drifts would intersect those drifts along the vertical ascent pathway. In addition, this detailed model is used to support the conclusion that, following the potential inflow of magma into a drift, vertical magma ascent is most likely along the original plane of ascent relative to a new location located elsewhere along a drift (Bechtel SAIC Company, LLC, 2003b). A separate process model for potential magma flow into an intersected drift uses a detailed three-dimensional hydromechanical model for fluid flow from a vertical plane into a horizontal tube. This model supports the abstraction ascending magma will flow into an intersected drift and fill the drift with magma on the order of minutes (Bechtel SAIC Company, LLC, 2003b). In addition, this process model suggests renewed magma ascent is more likely in the pillars between drifts rather than above the potentially intersected drifts. The result suggests subvolcanic conduit localization might be more likely in pillars than in drifts, however, DOE does not incorporate this result into the performance calculations (Bechtel SAIC Company, LLC, 2003a,b). Results of the three-dimensional hydromechanical model are used to support the DOE conclusion the likelihood of subvolcanic conduit formation (i.e., number of waste packages entrained in potential volcanic events) should account for randomized conduit localization along the intrusion, as well as potential localization along a drift (Bechtel SAIC Company, LLC, 2003a).

In summary, the DOE models for the number of waste packages entrained during potential volcanic events represent a process that has no known reasonable analogy with basaltic volcanic systems. Thus, this model abstraction cannot be directly supported by empirical observations at analog volcanoes. These DOE models, however, are supported by detailed process-level models, which are consistent with models in the available literature and current understandings of potential basaltic volcanic processes.

Overall the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.10.5), is sufficient to expect that the information necessary to assess volcanic disruption of waste packages, with respect to model abstraction output being supported by objective comparisons will be available at the time of a potential license application. As noted in this section of the report, however, further information should be provided on how consideration of magma with higher volatile content (approximately 4 wt% water) may affect models of magma ascent and flow processes.

5.1.3.10.5 Summary and Status of Key Technical Issue Subissues and Agreements

Table 5.1.3.10-1 provides the status of all key technical issue subissues, referenced in Section 5.1.3.10.2, for the Volcanic Disruption of Waste Packages Integrated Subissue. The table also provides the related DOE and NRC agreements pertaining to the Volcanic Disruption of Waste Packages Integrated Subissue. The agreements listed in the table are associated with one or all five generic acceptance criteria discussed in Section 5.1.3.10.4. Note the status and the detailed agreements pertaining to all the key technical issue subissues are provided in Table 1.2-1 and Appendix A.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (e.g., specified testing or analyses), indicates that information necessary to begin a technical review will likely be available at the time of a potential license application.

Table 5.1.3.10-1. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreement*
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.2.18 IA.2.19 IA.2.20
		Closed	IA.2.05 IA.2.10
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 review methods.

NOTE: Key Technical Issue Agreement GEN.1.01 pertains to multiple integrated subissues, as well as some specific issues related to this integrated subissue.

5.1.3.10.6 References

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5.1.3.11 Airborne Transport of Radionuclides

5.1.3.11.1 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., Walker, 1993; Blackburn, et al., 1976). In the event of a volcanic eruption through the potential repository, high-level waste also may 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 potential repository at Yucca Mountain and the consequences of this activity for waste package integrity are discussed 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 Redistribution of Radionuclides in Soil. The relationship of this integrated subissue to other integrated subissues is depicted in Figure 5.1.3.11-1. The overall organization and identification of all the integrated subissues are depicted in Figure 1.1-2 of this report.

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) and Bechtel SAIC Company, LLC (2003a,b). 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–g). Because supporting analysis and model reports recently were provided by DOE, some revised topical areas that did not pertain to prior agreement issues were not reviewed in detail for this report.

5.1.3.11.2 Relationship to Key Technical Issue Subissues

The Airborne Transport of Radionuclides 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)

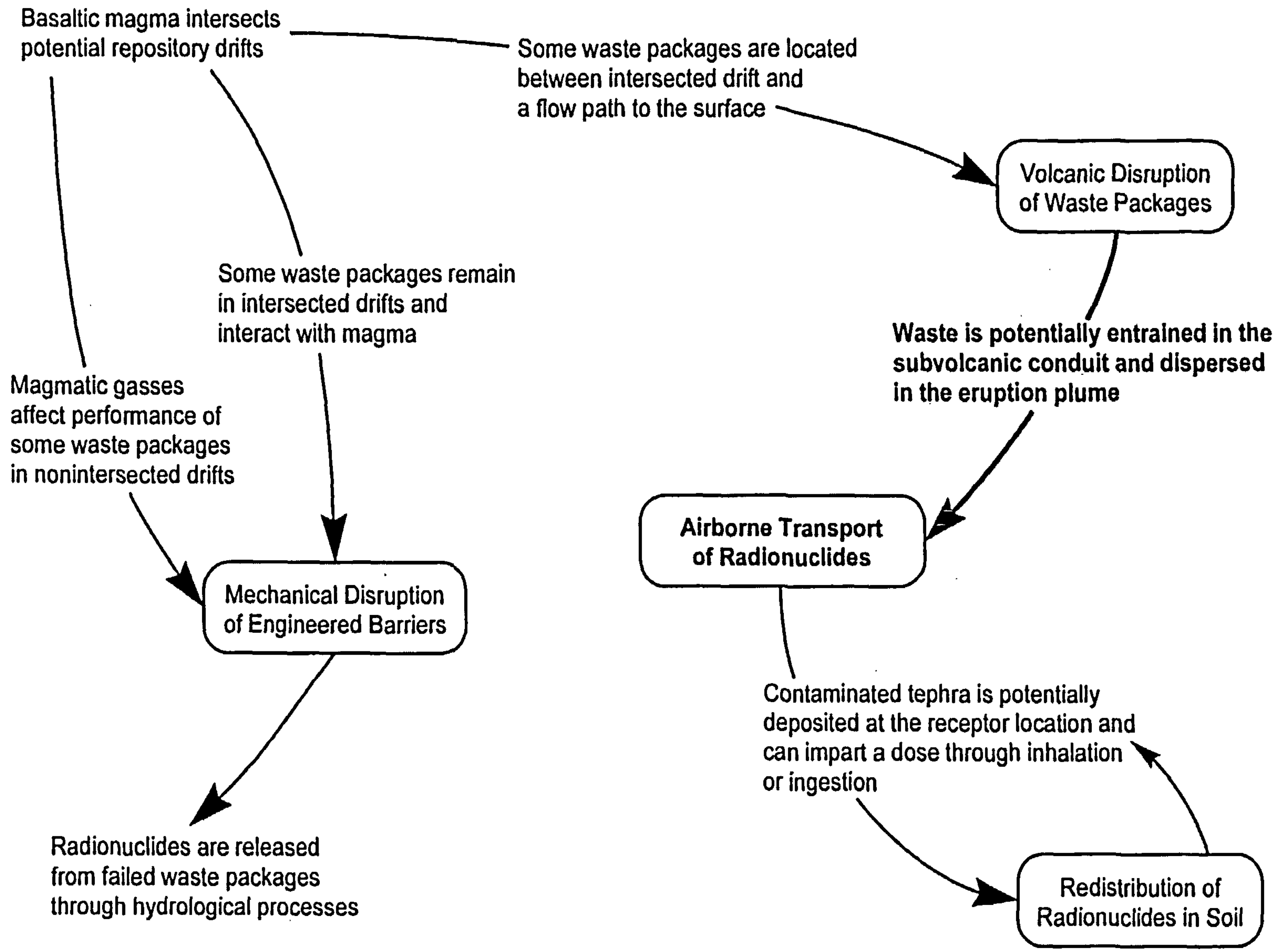

Figure 5.1.3.11-1. Diagram Illustrating the Relationship Between Airborne Transport of Radionuclides and Other Integrated Subissues. Material in Bold Is Identified in the Text.

- 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 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 issues subissues. The subsequent sections incorporate applicable portions of these key technical issue subissues, however, no effort was made to explicitly identify each subissue.

5.1.3.11.3 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. Appendix D identifies four topics with significance to waste isolation: (i) volume of ash produced by an eruption (medium significance), (ii) remobilization of ash deposits (medium significance), (iii) inhalation of resuspended volcanic ash (high significance), and (iv) wind vectors during an eruption (medium significance). These processes directly affect the amount of radionuclides potentially deposited at the reasonably maximally exposed individual location by volcanic eruption through the potential repository. Remobilization of ash deposits and the inhalation of resuspended volcanic ash are evaluated in Section 5.1.3.13. 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 CRWMS M&O (2000h) and Bechtel SAIC Company, LLC (2001a,b). As stated in Section 5.3 of 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 determines the magnitude of the overall mean annual dose from nominal and disruptive performances during the first 10,000 years."

5.1.3.11.4 Technical Basis

NRC developed a review plan (NRC, 2003) consistent with the acceptance criteria and review methods found in the previous issue resolution status reports. A status assessment of the DOE approaches for including airborne transport of radionuclides in total system performance assessment abstractions is provided in the following subsections. This assessment is organized according to the five review methods identified in Section 2.3: (i) Model Integration (including system description), (ii) Data and Model Justification, (iii) Data Uncertainty, (iv) Model Uncertainty, and (v) Model Support.

5.1.3.11.4.1 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 potential repository, high-level waste also may 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 by wind or surface water of the radionuclides and volcanic ash after initial deposition. Volcanic risk calculations are governed by the amount of contaminated particles inhaled by the receptor in the years following a potential volcanic eruption. Airbome transport and deposition of radionuclides in volcanic ash plumes must be modeled to estimate the dose consequences and risks associated with these phenomena.

A conceptual and mathematical model, implemented in the computer code known as ASHPLUME, has been developed for atmospheric dispersion and subsequent deposition of tephra from a potential eruption at Yucca Mountain, Nevada (Jarzemba, et al., 1997). ASHPLUME is used as a component of the DOE total system performance assessment model to assess hazards from possible volcanic activity at the Yucca Mountain site. DOE conducted a comparison of ASHPLUME model results to representative tephra-fall deposits (e.g., Bechtel SAIC Company, LLC, 2003b; CRWMS M&O, 2000a,g). Ash distribution patterns and depths predicted by the model are consistent with observations from analog sites. ASHPLUME uses the Suzuki (1983) model to abstract the thermo-fluid dynamics of ash dispersion in the atmosphere. The primary equation for calculating the areal density of ash deposition following a volcanic eruption is given in Jarzemba, et al. (1997) as

$$X(x,y) = \int_{\phi_{min}}^{\phi_{max}} \int_{0}^{H} \frac{5 f_z(z) f_\phi(\phi) Q}{8\pi C (t+t_s)^{5/2}} \exp\left\{ -\frac{5\left[(x-ut)^2 + y^2\right]}{8C(t+t_s)^{5/2}} \right\} dz\, d\phi \qquad (5.1.3.11\text{-}1)$$

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; $f_\phi(\phi)$ is a probability density function for grain size (particle diameter), 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 estimated as part of performance assessments (Connor, et al., 2001; CRWMS M&O, 2000a,c; Jarzemba, 1997).

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 the 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 diameter particles (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 $f_\Phi(\phi)$.

In summary, airborne transport and deposition of radionuclides in ash plumes must be modeled to estimate the dose consequences and risks associated with these volcanic phenomena. DOE demonstrates the ASHPLUME code, as implemented by DOE in CRWMS M&O (2000g), 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 provides the cumulative distribution functions for both the mean ash particle diameter used in its models and the ash-dispersion controlling constant. Subsequently, the recommended distribution incorporates the range of values that have been estimated from recent work on the Lathrop Wells Cone tephra sheet (Bechtel SAIC Company, LLC, 2003b). These values appear reasonable and, therefore, the NRC staff considers that DOE has information available to address this review method.

Overall, the available information is sufficient to expect that the information necessary to assess airborne transport of radionuclides with respect to system description and model integration will be available at the time of a potential license application.

5.1.3.11.4.2 Data and Model Justification

Risk insights pertaining to the airborne transport of radionuclides indicate the most important data and model justification needs are those used to estimate the volume of ash produced by an eruption and to analyze wind vectors during an eruption (Appendix D). Included in this insight is the basis for evaluating the range of eruption energetics used by DOE in the ASHPLUME simulations, the method for incorporating high-level waste into erupting tephra, and windfield characteristics used in ASHPLUME simulations of tephra and high-level waste dispersion. Each of these factors could significantly affect estimates of dose and risk at the receptor location.

The ASHPLUME model was first developed for use in the high-level waste program by Jarzemba, et al. (1997) and later modified by DOE (e.g., CRWMS M&O, 2000g). Most 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. In CRWMS M&O (2000a), analogous eruptions, including but not limited to the 1975 Tolbachik, Russia; 1943–52 Paricutin, Mexico; and 1850–1999 Cerro Negro, Nicaragua, and violent strombolian eruptions are cited as the sources of acceptable parameters for use in ASHPLUME. These data and the volcanological processes represented 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 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., Connor, et al., 2001, 2000; Hill, et al., 1998; Rosi, 1998; Sparks, et al., 1997; Carey, 1996; Woods, 1995). 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

$$\frac{dV}{dt} = \left[\frac{H}{1.67}\right]^4 \qquad (5.1.3.11\text{-}2)$$

and

$$V = \frac{dV}{dt} T \qquad (5.1.3.11\text{-}3)$$

These relationships provide a check on input parameters. It is crucial for DOE to track the mass flow rate together with the muzzle velocity at the vent for simulated eruptions in ASHPLUME to ensure 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). A technical basis is needed to ensure that mass flow and vent velocity regimes are sufficient to maintain such columns for all ASHPLUME simulations to avoid collapsed plumes. Currently, it appears some modeled events have mass flow rates and vent velocities too low to sustain such plumes (CRWMS M&O, 2000c).

The ASHPLUME code in CRWMS M&O (2000h) uses the erupted ash volume as a proxy for eruptive power in the computation of plume height estimates. The technical basis for inclusion of volume information from analog volcanoes is mostly described in Bechtel SAIC Company, LLC (2003b). Based on the estimated volumes of Quaternary basaltic volcanoes in the Yucca Mountain region, uniform distribution between 0.004 km^3 [0.001 mi^3] and 0.08 km^3 [0.02 mi^3] adequately captures the uncertainty associated with ash volume from a basaltic eruption at the potential repository (Bechtel SAIC Company, LLC, 2003b). There are no restrictions on the subsequent use of this distribution. In Appendix A of Bechtel SAIC Company, LLC (2003a), DOE states information needed to close Igneous Activity 2.03 additional information needed (AIN–1) (Reamer and Williams, 2000) is provided in Bechtel SAIC Company, LLC (2004). In addition, Appendix B of Bechtel SAIC Company, LLC (2003a) reports technical documentation for column height, wind speed, and wind direction is found in this analysis and model report. Bechtel SAIC Company, LLC (2004) is one of several reports scheduled for revision. This report was not available at the time of this status report, and review is ongoing. DOE has indicated that this report will aid validation of input parameters and provide a better understanding how tephra volumes have been used in ASHPLUME Version 2.0 to calculate column height and other eruptive characteristics.

CRWMS M&O (2000a) notes 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

$$\rho_c = \log\left(\frac{d^a_{\min}}{d^f}\right) \qquad (5.1.3.11\text{-}4)$$

where ρ_c is the incorporation ratio, 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), is to bound the particle size and density distribution for estimating the dispersion of contaminated waste. Jarzemba, et al. (1997) arbitrarily choose a value of $\rho_c = 0.3$ to illustrate the application of the model. The assumption that $\rho_c = 0.3$ is propagated through the total system performance assessment for site recommendation (CRWMS M&O, 2000c). That Jarzemba, et al. (1997) made this assumption about the incorporation ratio, as an example, is not a sufficient basis for DOE to make this assumption in a potential 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 (Reamer, 2001). The whole incorporation model, not just the ρ_c parameter as an incorporation ratio, must be justified. NRC review of Bechtel SAIC Company, LLC (2004) will examine model parameters and provide a better understanding how such an incorporation ratio has been used in ASHPLUME to quantify properly the complex process of high-level waste incorporation and transport.

Wind speed is a parameter that significantly affects tephra dispersion models for basaltic volcanoes (e.g., Hill, et al., 1998). Observations of the most violent strombolian basaltic eruptions show column heights reaching altitudes of 2–6 km [1–4 mi] above ground level. Although near-ground-surface wind data are available for the potential 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 uses wind speeds and directions obtained from near-surface stations (CRWMS M&O, 2000a,c). More recent analyses by DOE have begun to employ data sets that extend to higher altitudes (Bechtel SAIC Company, LLC, 2003a, 2001a,b), including the incorporation of meteorological data (National Oceanic and Atmospheric Administration, n.d.) from the Desert Rock airport near Mercury, Nevada.

A stratified windfield is incorporated into ASHPLUME by specifying variation in the windfield as a function of height, which is necessary to model the effects of stratified wind velocities and directions for eruptions (e.g., Glaze and Self, 1991). 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

$$u_{\text{avg}} = \frac{1}{Z}\sum_{k=0}^{N_k} u_k \Delta z_k \qquad (5.1.3.11\text{-}5)$$

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.

In summary, for the total system performance assessment for site recommendation (CRWMS M&O, 2000c), DOE uses wind speed data that was expanded on and more completely described in Bechtel SAIC Company, LLC (2001c). The NRC staff has noted that DOE models of a volcanic eruption through the potential repository may underestimate eruptive column heights (Reamer and Williams, 2000; Schlueter, 2000). Wind speed usually increases with altitude, and underestimating column heights could lead to selection of wind speed data inappropriately biased toward lower wind speeds. Use of low wind speeds in modeling a volcanic eruption through Yucca Mountain could lead to an incorrect ash distribution that, in turn, could affect the estimated dose to the reasonably maximally exposed individual (see Section 5.1.3.13). The analysis documented in Bechtel SAIC Company, LLC (2001b) shows use of the Desert Rock data increases probability-weighted mean annual doses by a factor of approximately two compared with earlier total system performance assessment–site recommendation values (CRWMS M&O, 2000c). DOE developed additional information related to the ASHPLUME parameters and determined ASHPLUME Version 2.0 is more appropriate for modeling atmospheric dispersal of contaminated ash than is ASHPLUME Version 1.4LV-dll (Bechtel SAIC Company, LLC, 2003a, Appendix B). The technical basis for inclusion of volume information from analog volcanoes is mostly described in Bechtel SAIC Company, LLC (2003b), with a summary listed in Bechtel SAIC Company, LLC (2003a, Appendix A). In Bechtel SAIC Company, LLC (2003b), DOE documented the range of tephra volumes and the basis for the range used to support the total system performance assessment–license application calculations, with additional information provided in Bechtel SAIC Company, LLC (2004). The NRC staff review of this report is ongoing. This DOE report also provides a better understanding how an incorporation ratio, ρ_c, is used in ASHPLUME to quantify properly the complex process of high-level waste incorporation and transport. To address concerns associated with the effects on the wasteform from interactions with magma and magmatic products (Bechtel SAIC Company, LLC, 2003a, Appendix D), DOE agreed to describe the method of high-level waste incorporation that will be used for the total system performance assessment–license application (Reamer, 2001).

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.11.5), is sufficient to expect that the information necessary to assess airborne transport of radionuclides with respect to data being sufficient for model justification will be available at the time of a potential license application.

5.1.3.11.4.3 Data Uncertainty

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.

The function for distribution of tephra and high-level waste in the vertical eruption column, β (beta), 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 (also known as the ash dispersion controlling constant). 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. An increased 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. Jarzemba, et al. (1997) employ a log-uniform distribution for β that has a minimum value of 0.01 and a maximum value of 0.5. In CRWMS M&O (2000c), β is limited to a range 0.01–0.5, or a range that limits the ascent of particles in the tephra column. Hill, et al. (1998), however, find β = 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 β = 10 is used by DOE to demonstrate the ASHPLUME code can reasonably replicate a natural eruption (i.e., the 1995 Cerro Negro eruption). The DOE documentation should explain the rationale for the change in the use of values for this parameter.

In summary, models or model abstractions that use parameter values, assumed ranges, probability distributions, and bounding values must be technically defensible and accountable to accurately depict the risk estimate. DOE has indicated that in response to agreements reached between NRC and DOE on igneous activity (Reamer, 2001; Reamer and Williams, 2000), it will provide revisions of reports to address the parameter discrepancies between CRWMS M&O (2000c) and CRWMS M&O (2000g).

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.11.5), is sufficient to expect that the information necessary to assess airborne transport of radionuclides with respect to data uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.11.4.4 Model Uncertainty

DOE notes 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). 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 Gas-Thrust (CRWMS M&O, 2000c), currently are not implemented, which presents 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. With this limitation, it is not possible to evaluate the direct effects of variations of some physical parameters (e.g., initial volatile content) to the expected dose to the reasonably maximally exposed individual. Because DOE demonstrated the ASHPLUME code can reasonably replicate analog eruptions (CRWMS M&O, 2000g), this concern has been addressed. Second, because ASHPLUME is an empirical model, it is difficult to gain confidence in the manner in which 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. DOE agreed, however, to conduct sensitivity studies (Schlueter, 2000). Third, there is a potential the repository engineered barrier 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 potential repository itself may modify flow conditions and, therefore, the eruptive characteristics (Reamer, 2001). 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.

The staff notes 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, the code is 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 the ensuing dispersion into the accessible environment (Hill and Connor, 2000). DOE considered the Gas-Thrust model, but concluded the parameter β has a similar effect (CRWMS 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 (CRWMS M&O, 2000g), the concern is alleviated regarding treatment of thermophysical properties within the eruption column.

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 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 {0.003 km^3 [0.0007 mi^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 affecting peak annual total effective dose equivalents for individuals located 20 km [12 mi] from a potential repository-penetrating volcanic eruption (Hill and Connor, 2000). Finally, changes in the physics of the eruption caused by the development of complex near-surface magma flow in the potential repository can be incorporated in total system performance assessment.

In summary, DOE has demonstrated the ASHPLUME code, as implemented, can reasonably replicate a natural analog eruption (e.g., CRWMS M&O, 2000g). It is recognized, however, the changes in physics of an eruption, because of interactions with the potential repository, may necessitate modifications to the code. This determination cannot be made until the analyses have been completed for the Volcanic Disruption of Waste Packages Integrated Subissue (Section 5.1.3.10), as agreed to in Reamer (2001). Also, the basis for the incorporation ratio, ρ_c, is the observation of xenoliths being incorporated into natural flows and eruptions; however, further evaluation by DOE is needed to determine if the incorporation ratio can be justified, and, if not, which alternative method should be used as a substitute (Reamer, 2001). Air and water transport of 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 any uncertainty in modeling air transport during the eruption. Ash redistribution and inhalation of resuspended ash is being evaluated in the Redistribution of Radionuclides in Soil Integrated Subissue (Section 5.1.3.13). Therefore, to get a reasonably accurate evaluation of the risk from a volcanic eruption, information about these three integrated subissues needs to be articulated and correlated. There are agreements in place in all three integrated subissues to address these concerns as they relate to model uncertainty (Reamer, 2001; Reamer and Williams, 2000; Schlueter, 2000).

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.11.5), is sufficient to expect that the information necessary to assess airborne transport of radionuclides with respect to model uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.11.4.5 Model Support

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 performed a similar analysis. As demonstrated in Figure 6 of CRWMS M&O (2000g), the ASHPLUME code, as implemented by DOE, also can reasonably replicate the 1995 Cerro Negro eruption. The NRC staff, 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) (Schlueter, 2000).

Questions remaining about use of the ASHPLUME model relate 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 development that allows exploration of the parameters governing these variations. Thus, many nuances of observed eruption columns and their deposits can now be understood by fundamental physical processes (e.g., Sparks, et al., 1997). Such an understanding is important for volcanic risk assessment related to the potential repository at Yucca Mountain because there are no observations analogous to the potential behavior of dense high-level waste particles in eruption columns, and no appropriate analogs have been identified. There also is considerable uncertainty how to simulate the entrainment and dispersal of high-level waste in eruption columns. Physically accurate eruption column models provide an opportunity to extend understanding of tephra plumes to encompass the potential 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 agreed to describe the methodology it will use in its models for waste incorporation, including possible particle aggregation (Reamer, 2001). The DOE response to the agreement items were not available at the time of this review.

In summary, DOE completely documented that the ASHPLUME code, as implemented by DOE, can reasonably replicate a natural basaltic volcanic eruption and agreed to provide the necessary information on high-level waste incorporation to demonstrate the code has a sound technical basis. It is recognized 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 processes of ash distribution and deposition will be necessary.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.11.5), is sufficient to expect that the information necessary to assess airborne transport of radionuclides with respect to model abstraction output being supported by objective comparisons will be available at the time of a potential license application.

5.1.3.11.5 Summary and Status of Key Technical Issue Subissues and Agreements

Table 5.1.3.11-1 provides the status of all key technical issue subissues, referenced in Section 5.1.3.11.2 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 review methods discussed in Section 5.1.3.11.4. Note the status and the detailed agreements pertaining to all the key technical issue subissues are provided in Table 1.2-1 and Appendix A.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (e.g., specified testing or analyses), indicates that information necessary to begin a technical review will likely be available at the time of a potential license application.

Table 5.1.3.11-1. 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.2.01 through 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 review methods.			
NOTE: Key Technical Issue Agreement GEN.1.01 pertains to multiple integrated subissues, as well as some specific issues related to this integrated subissue.			

5.1.3.11.6 References

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5.1.3.12 Concentration of Radionuclides in Ground Water

5.1.3.12.1 Description of Issue

The Concentration of Radionuclides in Ground Water subissue relates to estimating the effects of well pumping on the concentration of radionuclides in ground water. To limit speculation, this is to be a stylized calculation as described in NRC (1999a) and its implementation is constrained by requirements in 10 CFR Part 63. Relationship of this integrated subissue to other integrated subissues is depicted in Figure 5.1.3.12-1. The overall organization and identification of all the integrated subissues are depicted in Figure 1.1-2. The DOE description and technical bases for abstraction of concentration of radionuclides in ground water are documented in CRWMS M&O (2000a,b) and several analysis and model reports cited throughout the following sections. This section documents the current NRC staff understanding of the abstractions DOE used to incorporate concentration of radionuclides in ground water into its total system performance assessment. The assessment is focused on those aspects most important to repository safety based on the risk insights gained to date, including Appendix D. The scope of the assessment presented here is limited to examining if the data gathered and the methodology used by DOE are likely to be adequately documented for the staff to undertake a detailed technical review. This assessment is not a regulatory compliance determination review of a potential license application.

This section does not address potential repository performance relative to compliance with separate ground water protection standards because this is treated as a separate issue in NRC (2003). Discussions related to the separate ground water protection standard are contained in Section 5.1.4.3 of this report.

5.1.3.12.2 Relationship to Key Technical Issue Subissues

The Concentration of Radionuclides in Ground Water Subissue incorporates subject matter previously captured in the following six key technical issue subissues:

- Unsaturated and Saturated Flow Under Isothermal Conditions: Subissue 5—Saturated Zone Flow and Dilution Processes (NRC, 1999b)
- Radionuclide Transport: Subissue 3—Radionuclide Transport Through Alluvium (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)

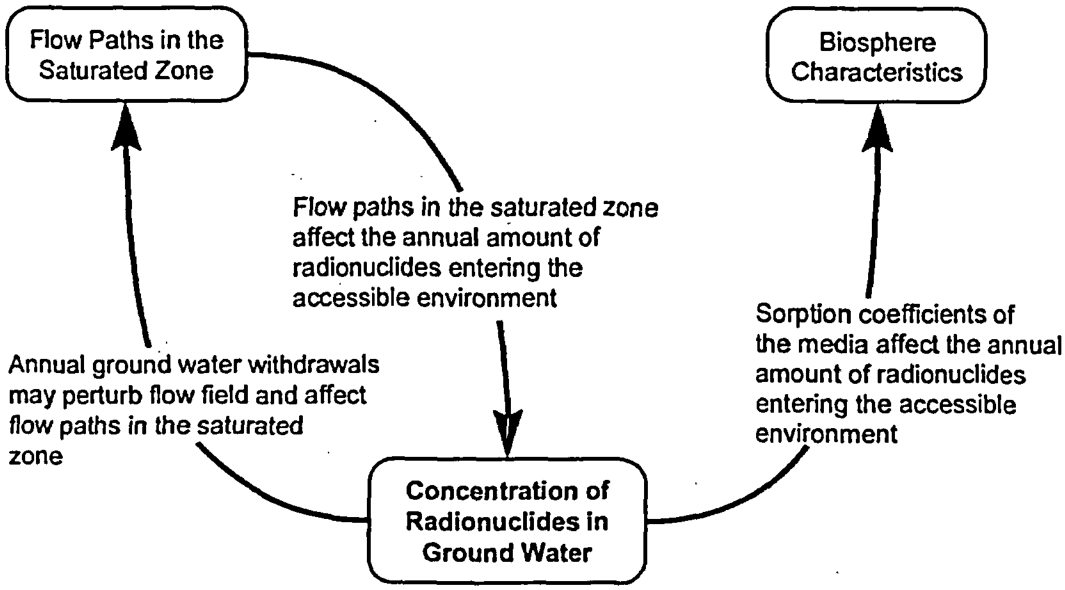

Figure 5.1.3.12-1. Diagram Illustrating the Relationship Between Concentration of Radionuclides in Ground Water and Other Integrated Subissues. Material in Bold Is Identified in the Text.

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 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.

5.1.3.12.3 Importance to Postclosure Performance

One aspect of risk informing the NRC staff understanding of postclosure repository performance (Appendix D) is to determine how this integrated subissue is related to the DOE repository safety strategy. Risk insights pertaining to the concentration of radionuclides in ground water indicate the well pumping model is of low significance to waste isolation. The details of the risk insights ranking are provided in Appendix D. The annual amount of radionuclides that enter the accessible environment is the result of the release and transport calculations in previously discussed model abstractions. The remaining parameters in the concentration calculation do not vary and, therefore, do not have any potential to increase or decrease the resulting concentration. For example, the annual water demand (i.e., pumping volume) is specified by regulation in 10 CFR Part 63 at 3.7×10^6 m^3 [3,000 acre-ft]. This prescribed approach constrains the significance of modeling radionuclide concentrations in ground water.

The importance of the concentration of radionuclides in ground water to the postclosure repository performance has been addressed several times during the last 6 years. Sensitivity analyses based on uncertainty in the pumping rate (producing variable plume capture) performed to support CRWMS M&O (2000c) indicated performance estimates were only slightly sensitive to dilution of radionuclides in ground water because of well pumping (CRWMS M&O, 2000c, Section 4.2.8). Based on that assessment, DOE did not consider dilution of radionuclides in ground water due to well pumping to be a principal factor in its postclosure safety case (CRWMS M&O, 2000c).

The total system performance assessment model for site recommendation adopted by DOE, which differs from the model used for the repository safety strategy report (CRWMS M&O, 2000c), assumes complete plume capture for radionuclides crossing the compliance boundary and subsequent dilution of the captured radionuclides in the pumped volume of water. The DOE sensitivity analyses incorporating this model indicate the calculated dose was directly affected by the pumping volume, and increases or decreases in the pumping volume produced a proportional reduction or increase in the calculated dose (CRWMS M&O, 2000d, Figure 5.2-16).

In the more recent postclosure analysis (Bechtel SAIC Company, LLC, 2002), DOE further modified its well capture abstraction in CRWMS M&O (2000d) to include complete mixing of the captured radionuclides in the annual water-use demand of 3.7×10^6 m^3/yr [3,000 acre-ft/yr] presented at 10 CFR Part 63. Note that in CRWMS M&O (2000d), all radionuclides reaching the compliance boundary are assumed to be captured. The annual water-use volume of 3.7×10^6 m^3/yr [3,000 acre-ft/yr] is less conservative than the approximately 2.5×10^6 m^3/yr [2,000 acre-ft/yr] previously used in CRWMS M&O (2000d) and results in lower mean annual dose estimates to the reasonably maximally exposed individual (Williams, 2001).

The calculation for estimating concentrations in ground water is constrained by requirements in 10 CFR Part 63 that specify the annual water demand and an annual dose limit. Currently, DOE assumes complete capture of the ground water plume. Additional analyses of the capture fraction, within the constraints of 10 CFR Part 63, are unlikely to produce significantly different results. The requirements limit the significance of modeling radionuclide concentrations in ground water.

5.1.3.12.4 Technical Basis

NRC developed a review plan (NRC, 2003) consistent with the acceptance criteria and review methods found in previous issue resolution status reports. A status assessment of the DOE approaches for including concentration of radionuclides in ground water in total system performance assessment abstractions is provided in the following subsections. This assessment is organized according to the five review methods identified in Section 2.3: (i) Model Integration (including system description), (ii) Data and Model Justification, (iii) Data Uncertainty, (iv) Model Uncertainty, and (v) Model Support.

5.1.3.12.4.1 Model Integration

To determine the concentration of radionuclides in ground water at the location of the reasonably maximally exposed individual, DOE assumes complete capture of all radionuclides reaching the compliance boundary. The total volume of water pumped at the location of the reasonably maximally exposed individual is 3.7×10^6 m^3/yr [3,000 acre-ft/yr]. To determine the concentration of radionuclides in ground water reaching the biosphere, DOE uses the Bechtel SAIC Company, LLC (2002) model to calculate the amount of each radionuclide species reaching the geosphere/biosphere interface in a given year. The amount of each radionuclide species reaching the geosphere/biosphere interface is converted to a concentration by diluting the total annual activity of the radionuclides into the specified annual water demand {3.7×10^6 m^3/yr [3,000 acre-ft/yr]}.

DOE assumes 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 ground water withdrawn.

In summary, available information for the saturated zone, from the saturated zone process model report (CRWMS M&O, 2000a) and the supporting analysis and model reports, is sufficient to (i) determine the concentration of radionuclides in ground water and (ii) determine the concentration of radionuclides in ground water in total system performance assessment analyses.

Overall, the available information is sufficient to expect that the information necessary to assess concentration of radionuclides in ground water with respect to system description and model integration will be available at the time of a potential license application.

5.1.3.12.4.2 Data and 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 required to estimate the concentration of radionuclides in ground water.

To support early estimates of the concentration of radionuclides in ground water, DOE estimated future ground water pumping rates based on a combination of data from a 1997 survey of ground water pumping in Nye County, Nevada (State of Nevada, 1997) and the 1990 census data (U.S. Census Bureau, 1999). These data were used to estimate a range of present-day, per-farm pumping rates. In those analyses, DOE assumed the size of the hypothetical farming community to be reasonably consistent with NRC (1999a), which indicated 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.

In Bechtel SAIC Company, LLC (2002), DOE fixed the annual volume of ground water pumped at the location of the reasonably maximally exposed individual to 3.7×10^6 m^3/yr [3,000 acre-ft/yr].

Overall, the available information is sufficient to expect that the information necessary to assess concentration of radionuclides in ground water with respect to data being sufficient for model justification will be available at the time of a potential license application.

5.1.3.12.4.3 Data Uncertainty

In Bechtel SAIC Company, LLC (2002), DOE adopted an annual water demand of 3.7×10^6 m^3/yr [3,000 acre-ft/yr] as prescribed in 10 CFR Part 63. As a result, no variation is generated in this abstraction because 10 CFR Part 63 sets the annual water demand at

3.7×10^6 m^3 [3,000 acre-ft], and all radionuclides in the plume are assumed to be captured by the pumping well.

Overall, the available information is sufficient to expect that the information necessary to assess concentration of radionuclides in ground water with respect to data uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.12.4.4 Model Uncertainty

The concentration of radionuclides in ground water and dose calculations for the safety case consider radionuclide capture and total ground water pumping as defined by the regulations for the potential high-level waste repository as well as NRC (1999a). 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/yr [3,000 acre-ft/yr]. As for radionuclide capture, DOE assumes 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 concentration of radionuclides in ground water and the dose is unaffected by the ground water pumping uncertainty. DOE does not consider changes in ground water demand in the future in Amargosa Valley in Bechtel SAIC Company, LLC (2002).

In addition, 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.

Overall, the available information is sufficient to expect that the information necessary to assess concentration of radionuclides in ground water with respect to model uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.12.4.5 Model Support

As indicated previously, the annual ground water demand is prescribed as 3.7×10^6 m^3/yr [3,000 acre-ft/yr] in 10 CFR Part 63. The DOE (CRWMS M&O, 2000a) and CNWRA (Winterle, 2003) site-scale-steady-state ground water flow modeling efforts for the region that includes Fortymile Wash and Yucca Mountain indicate there is sufficient ground water flow in the vicinity of the reasonably maximally exposed individual to meet the prescribed pumping rate during the 10,000-year period of performance.

Overall, the available information is sufficient to expect that the information necessary to assess concentration of radionuclides in ground water with respect to model abstraction output being supported by objective comparisons will be available at the time of a potential license application.

5.1.3.12.5 Summary and Status of Key Technical Issue Subissues and Agreements

The resolution status of this integrated subissue is based on the resolution status of each of the contributing key technical issue subissues referenced in Section 5.1.3.12.2, for the Concentration of Radionuclides in Ground Water. Table 5.1.3.12-1 provides the status of all key technical issue subissues. The table also provides the related DOE and NRC agreements pertaining to the Concentration of Radionuclides in Ground Water Integrated Subissue. The agreements listed in the table are associated with one or all five generic review methods discussed in Section 5.1.3.12.4. Note the status and the detailed agreements pertaining to all the key technical issue subissues are provided in Table 1.2-1 and Appendix A.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (e.g., specified testing or analyses)indicates that information necessary to begin a technical review will likely be available at the time of a potential license application.

Table 5.1.3.12-1. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreement*
Unsaturated and Saturated Flow Under Isothermal Conditions	Subissue 5—Saturated Zone Flow and Dilution Processes	Closed-Pending	None
Radionuclide Transport	Subissue 3—Radionuclide Transport Through Alluvium	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
*Related DOE and NRC agreements are associated with one or all five generic review methods.			

5.1.3.12.6 References

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———. "Biosphere Process Model Report." TDR–MGR–MD–000002. Rev. 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. Rev. 04 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000c.

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5.1.3.13 Redistribution of Radionuclides in Soil

5.1.3.13.1 Description of Issue

The Redistribution of Radionuclides in Soil Integrated Subissue addresses the movement of radionuclides following deposition on the ground, either through ground water irrigation or remobilization of volcanic tephra (i.e., ash) following an eruption. Movement of radionuclides is possible through redistribution of contaminated deposits by wind and water or leaching during rainfall and irrigation. 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 5.1.3.13-1. The overall organization and identification of all the integrated subissues are depicted in Figure 1.1-2. This section provides a review of the abstraction of redistribution of radionuclides in soil incorporated by DOE in its total system performance assessment.

The DOE description and technical basis for the redistribution of radionuclides in soil abstractions are summarized in Bechtel SAIC Company, LLC, (2003a) and six supporting analysis and model reports (Bechtel SAIC Company, LLC, 2003a–e). 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. Because supporting analysis and model reports were provided by DOE recently, some revised topical areas that did not pertain to prior agreement issues were not reviewed in detail for this report.

5.1.3.13.2 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 the subissue. The resolution status of this integrated subissue is based on the resolution status of each of the contributing key technical issue subissues.

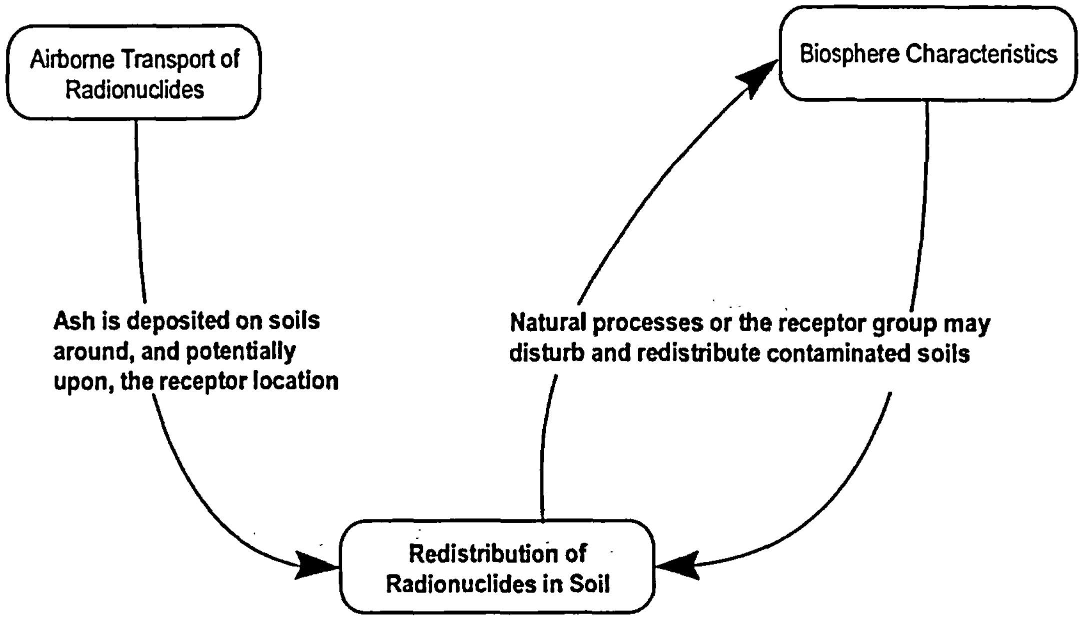

Figure 5.1.3.13-1. Diagram Illustrating the Relationship Between Redistribution of Radionuclides in Soil and Other Integrated Subissues. Material in Bold Is Identified in the Text.

The subsequent sections incorporate applicable portions of these key technical issue subissues, however, no effort was made to explicitly identify each subissue.

5.1.3.13.3 Importance to Postclosure Performance

Conceptually, aspects of the igneous activity exposure modeling related to this subissue include remobilization of tephra deposits and resuspension of fine-grained contaminated particles {i.e., <100 μm [0.004 in]} from these deposits to air with subsequent inhalation by the reasonably maximally exposed individual (hereafter, receptor). Following a potential volcanic eruption, a submillimeter-to-meters thick deposit of tephra could be deposited on hillslopes around Yucca Mountain that are part of the Fortymile Wash drainage basin. Remobilization processes focus on the erosion and surface transport of these tephra deposits in the Fortymile Wash drainage basin. Remobilized tephra is expected to follow a path similar to existing sediments (i.e., down the Fortymile Wash drainage during periods of overland water flow). In the currently active system, transported sediments begin to accumulate approximately several kilometers north of the receptor location, where the main Fortymile Wash drainage changes from a steep-sided channel to a broad, braided fan system. Existing sediment deposition continues south into the Amargosa Desert and overlaps the general area of the receptor location near the southern boundary of the Nevada Test Site (Bechtel SAIC Company, LLC, 2003d).

The significance of remobilization is that this process, through time, likely brings significant amounts of resuspendable particles into the general area of the receptor. Any initial tephra deposit at the receptor location will erode and become depleted in resuspendable particles through time, resulting in progressively lower inhalation doses in the years following a potential volcanic event. In contrast, additional fine-grained particles can be deposited in Fortymile Wash from remobilization processes. Surface winds can entrain fine-grained particles from the remobilized deposits, which can then be inhaled by the nearby receptor. Simple mass-balance scoping calculations (Hill and Connor, 2000; Hooper, 2004) indicate the accumulation rate of remobilized tephra likely exceeds the decay rate in airborne mass load from the original volcanic deposit at the receptor location. Thus, remobilization of tephra deposits may sustain airborne mass loads and associated inhalation doses for longer periods of time than indicated by simple decay relationships for original volcanic deposits (e.g., Bechtel SAIC Company, LLC, 2001; 2003b).

NRC modeling results (Appendix D) and Mohanty, et al. (2002) suggest remobilization and mass loading of ash are significant contributors to total system performance assessment results. The models and parameters used in these calculations include large uncertainties and continue to be refined (Hooper, 2004). Calculations used to bound the potential effects of remobilization assumed (i) a tephra deposit always occurs at the receptor location and (ii) mass loading does not decrease during the 10,000-year compliance period. These two assumptions resulted in an approximate factor of five increase in calculated risk, relative to basecase models that assume relatively rapid decay in airborne mass load following a potential volcanic event (Appendix D).

Past DOE model results (Bechtel SAIC Company, LLC, 2002; CRWMS M&O, 2000a) show igneous activity is a natural process that could cause a significant number of waste package failures and thus result in a dose to the receptor during the regulatory period of interest. To date, DOE has not documented a final model for remobilization in their total system performance assessment. Development of a remobilization model is, however, one acceptable method DOE could use to address Igneous Activity Key Technical Issue Agreement 2.17 (Reamer, 2001a). The DOE scoping analyses (Bechtel SAIC Company, LLC, 2002) suggest the significance of remobilization processes may be minor, however, these scoping calculations are limited by a lack of coupling between the remobilization rate and the mass loading decay rate (Bechtel SAIC Company, LLC, 2003b,f). DOE has refined its biosphere model and associated input parameters (Bechtel SAIC Company, LLC, 2003g), and has agreed to provide the technical basis for its remobilization models and associated results.

Risk insights indicate that the redistribution of radionuclides in soil is of low significance to waste isolation. The details of the risk insights ranking are provided in Appendix D. This section also includes the evaluation of the remobilization of ash deposits and the inhalation of resuspended volcanic ash, which are of medium significance (Appendix D). Redistribution of radionuclides also is addressed in modeling ground water releases from Yucca Mountain by consideration of soil leaching processes and the potential buildup of radionuclides in irrigated soils. Irrigation of agricultural fields through multiple growing seasons can lead to a buildup or washout of radionuclides in the soil, depending on the chemical properties of the radionuclides and soils. Leaching also can affect radionuclide soil concentrations from a potential volcanic event; however, the chemical properties of the key radionuclides contributing to dose for this scenario reduce the significance of this process (Appendix D). DOE (Bechtel SAIC Company, LLC, 2003a; CRWMS M&O 2000a) and NRC (Appendix D) reported low importance for ground water-related biosphere exposure pathways in sensitivity studies. The DOE assessments indicate that, for most radionuclides, buildup of radionuclides in the soil has a minor effect on the calculated dose conversion factors (CRWMS M&O, 2000b). These assessments show biosphere dose conversion factors increase by less than a factor of two for most radionuclides, even for buildup times on the order of thousands of years. More recent analyses on the effects of leaching on dose calculations using the new DOE biosphere model suggest similar results (Bechtel SAIC Company, LLC, 2003a). Section 5.1.3.14, Biosphere Characteristics, provides additional information on the significance of ground water pathway modeling in total system performance calculations. Prior to developing these system-level risk insights, staff reviews of the DOE documents included comments on the DOE leaching calculations that were subsequently resolved by the DOE updates to its models and documentation. Additional detailed discussion of these issues is provided in Schlueter (2004).

5.1.3.13.4 Technical Basis

NRC developed a review plan (NRC, 2003) consistent with the acceptance criteria and review methods found in the previous issue resolution status reports. A status assessment of the DOE approaches for including redistribution of radionuclides in soil in total system performance assessment abstractions is provided in the following subsections. This assessment is organized according to the five review methods identified in Section 2.3: (i) Model Integration (including system description), (ii) Data and Model Justification, (iii) Data Uncertainty, (iv) Model Uncertainty, and (v) Model Support.

5.1.3.13.4.1 Model Integration

For the volcanic event scenario, DOE uses a range of airborne mass loads to represent different activity levels of the receptor. Mass loads, however, are assumed to decay exponentially from levels representative of the first year after a potential eruption to a lower-level characteristic of preeruption conditions (Bechtel SAIC Company, LLC, 2003b,f). Using information primarily from analog areas, the DOE model indicates airborne mass loads would only be approximately 10 percent above preeruption levels within the first 10 years after most potential eruptions (i.e., Bechtel SAIC Company, LLC, 2003b). Only eruptions with calculated deposit thicknesses greater than 1 cm [0.4 in] at the receptor location would sustain elevated mass loads for a slightly longer time.

To support the conclusion that airborne mass loads would decrease exponentially after a potential volcanic eruption, DOE (Bechtel SAIC Company, LLC, 2003b) cites numerous studies conducted in various geographic locations. The DOE documentation, however, does not fully describe how the conditions affecting mass loading at these locations are analogous to conditions expected at the receptor location. Relative to current conditions at the receptor location, these areas are wetter, more vegetated, and have different soil and wind characteristics. Variations in these types of physical conditions strongly affect airborne mass loads above the deposit (e.g., Wiggs, 1997).

The wetter, more vegetated conditions in the areas studied by DOE appear capable of stabilizing or depleting the abundance of resuspendable particles relative to the arid, sparsely vegetated conditions at the receptor location. In addition, the studied areas are located away from the depositional basins of large drainage systems and, thus, do not have the potential to receive an influx of remobilized tephra following the volcanic event. Several studies in Bechtel SAIC Company, LLC (2003b), conducted in reasonably analogous areas (e.g., Anspaugh, et al., 1975), focused on the fixation of trace amounts of radionuclides by soil chemical processes rather than on decreases in fine-grained particulate abundances in relatively thick, contaminated deposits. Analyses in Bechtel SAIC Company, LLC (2003a–c) that support the DOE model of exponential decay in airborne mass load following a potential volcanic event do not consider significant differences in physical conditions between analog sites and the receptor location. Staff expect these differences in physical conditions to sustain elevated mass loads for longer periods of time than modeled in Bechtel SAIC Company, LLC (2003b).

One major assumption in the DOE model for changes in airborne mass load through time is that the additional influx of airborne particles from remobilized deposits is negligible (Bechtel SAIC Company, LLC, 2003b). This assumption arises, in part, through incorrect comparisons between sedimentary processes observed recently in the Fortymile Wash drainage system and processes likely to occur if appreciable amounts of easily redistributed volcanic tephra are deposited in this drainage system.

Current conditions in the Fortymile Wash drainage system are characterized by low sediment production and transport rates (e.g., DOE, 1993). The source area of the Fortymile Wash drainage system covers approximately 800 km^2 [309 mi^2] and includes the eastern slopes of Yucca Mountain. Although the Fortymile Wash depositional basin is approximately 130 km^2 [50 mi^2], most sedimentation during the last 1,000 to 10,000 years has been restricted to an approximate 24-km^2 [9-mi^2] area extending south from near the southern boundary of the Nevada Test Site. Deposition of 10^6–10^7 m^3 [3.5×10^7–3.5×10^8 ft^3] of loose tephra into this type of drainage system will strongly affect erosion and sediment transport rates (e.g., Segerstrom, 1950). A preliminary model that accounts for these effects shows minor to negligible amounts of tephra dilution would likely occur in the first decades following remobilization of a potential tephra deposit, with relatively large amounts of tephra being deposited in Fortymile Wash at or near the receptor location (Hooper, 2004). These observations and models do not support the DOE assertions in Bechtel SAIC Company, LLC (2003b,h) that the amount of remobilized tephra would be small, mixed with ambient sediment, and not significantly affect airborne mass loads for the receptor.

DOE has not documented its final model for the potential long-term redistribution of tephra in the Fortymile Wash drainage system. Preliminary analyses in Bechtel SAIC Company, LLC (2003a,b,h,i) evaluate dilution processes when trace amounts of tephra are released into active drainages around the 80,000-year-old Lathrop Wells volcano and examine some potential deposition and erosion sites in the depositional basin of Fortymile Wash. These analyses, however, do not relate current conditions in the Fortymile Wash drainage system to expected conditions and processes following potential deposition of a relatively extensive tephra-fall deposit. For example, hillslope erosion rates on a potential tephra deposit would likely increase significantly (e.g., Segerstrom, 1950), with remobilized tephra probably constituting the bulk of the transported sediment in the drainage system. Tephra grains are lower density and easier to suspend in flowing water than the sediment grains currently in Fortymile Wash. This change in grain density would affect posteruption sediment transport rates. Depositional patterns could change in response to increased sediment load. Thus, the staff believes current conditions in the Fortymile Wash drainage system may not be representative of the range of physical conditions likely to operate in the years following deposition of a potential volcanic tephra-fall deposit. DOE has agreed to provide to NRC the technical basis for the tephra redistribution model (Reamer, 2001a).

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.13.5), is sufficient to expect that the information necessary to assess redistribution of radionuclides in soil with respect to system description and model integration will be available at the time of a potential license application.

5.1.3.13.4.2 Data and Model Justification

Volcanic risk calculations are governed by the amount of contaminated particles inhaled by the receptor in the years following a potential volcanic eruption. Airborne mass loads are controlled by (i) soil moisture content; (ii) soil characteristics, such as grain size and mineralogy; (iii) vegetation cover; and (iv) local-scale meteorological conditions, such as wind speed and turbulence effects (e.g., Wiggs, 1997). Because these characteristics are site specific, the DOE analyses of data used to represent airborne mass loads in dosimetry calculations should evaluate significant differences in these characteristics between potential analog sites and the receptor location. Otherwise, erroneously high or low airborne mass loads may be used to represent the microenvironmental conditions specified in the DOE dose calculations (Bechtel SAIC Company, LLC, 2003d).

Airborne mass loads used by DOE (Bechtel SAIC Company, LLC, 2003b,f) primarily are based on data collected at various geographic areas outside the Yucca Mountain region. The DOE documentation, however, does not explain how the conditions affecting mass loading at these locations are analogous to conditions expected at the receptor location. The NRC staff previously evaluated the basis for the DOE mass load values and had questions on the relationship between analog sites used by DOE and specific conditions at the receptor location (Reamer, 2001b). These concerns focused on potentially significant differences between the receptor location and analog sites for (i) annual rainfall, (ii) soil morphology and composition, (iii) local meteorological conditions, (iv) amount and types of vegetation, and (v) types of surface disturbing activities. Information provided by DOE to address these questions (Bechtel SAIC Company, LLC, 2003a) is being assessed by NRC.

Some published data on airborne mass loads may not be suitable for use in exposure models such as the DOE environmental radiation model for Yucca Mountain (Bechtel SAIC Company, LLC, 2003a). This DOE model subdivides daily exposures into five discrete activity levels, each having specific exposure times. Periods of total outdoor exposure, which represent the periods of highest calculated dose, range from 0.9 to 8.7 hours per day (Bechtel SAIC Company, LLC, 2003d). Many outdoor mass loads cited in Bechtel SAIC Company, LLC (2003b), however, are for 24-hour daily averages. Because nighttime winds have lower velocities than daytime winds, nighttime mass loads are generally lower than daytime mass loads. Thus, mass loads derived from a daily average measurement will likely underestimate the mass load appropriate for several hours of exposure to daytime conditions. DOE has not yet addressed how average daily mass loads appropriately represent airborne mass loads for the specific exposure times used in the microenvironmental model (Bechtel SAIC Company, LLC, 2003a).

Appropriate data may not be directly available from the Yucca Mountain region to support models for tephra redistribution and resulting effects on airborne mass load through time. The last volcanic eruption in the Yucca Mountain region occurred 80,000 years ago, and the tephra-fall deposit from this volcano is almost completely removed by erosion (Bechtel SAIC Company, LLC, 2003i). Sediment transport processes represent equilibrium between low sediment production and erosion rates and episodic transport events involving coarse-grained bedload sediment (e.g., DOE, 1993). These conditions are not analogous to posteruption conditions in the years to perhaps centuries following a potential volcanic event at Yucca Mountain. Measurements of tephra dilution rates, or depositional and erosional patterns in the distal parts of the Fortymile Wash basin (Bechtel SAIC Company, LLC, 2003h), have questionable analogy to processes affecting potential tephra redistribution. Nevertheless, conclusions reached in Bechtel SAIC Company, LLC (2003a–c,e–i) regarding the insignificance of potential redistribution processes on airborne mass loads are based primarily on these data. DOE has agreed to provide its final model for the potential long-term redistribution of tephra in the Fortymile Wash drainage system (Reamer, 2001a).

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.13.5), is sufficient to expect that the information necessary to assess redistribution of radionuclides in soil with respect to data being sufficient for model justification will be available at the time of a potential license application.

5.1.3.13.4.3 Data Uncertainty

DOE propagates mass load parameter uncertainty in the Environmental Radiation Model for Yucca Mountain using stochastic sampling of triangular distribution functions (Bechtel SAIC Company, LLC, 2003a). In general, mass load parameter distributions for the volcanic disruption scenario are derived from measurements of airborne particle concentration made at locations thought to be analogous with the location of the receptor (Bechtel SAIC Company, LLC, 2003b,d). DOE concludes outdoor airborne mass loads in the year following a potential eruption are approximately twice the levels used to represent preeruption mass loads, however, this increase could be negligible to perhaps as high as a factor of five (Bechtel SAIC Company, LLC, 2003b). Indoor mass loads increase by a factor of two in the initial year following a potential volcanic eruption (Bechtel SAIC Company, LLC, 2003b).

For exposure times in an active outdoor environment, DOE samples airborne mass loads for total suspended particulates between 1 and 15 mg/m^3 [1×10^{-6} and 1.5×10^{-5} oz/ft^3]. This range appears reasonable based on data collected directly at a basaltic tephra-fall deposit for high levels of surface-disturbing activity (Hill and Connor, 2000). For exposure in an inactive outdoor environment, however, DOE samples a range of airborne mass loads from 0.05 to 0.3 mg/m^3 [5×10^{-8} to 3×10^{-7} oz/ft^3]. This range appears low, based on measurements of 0.1 to 1 mg/m^3 [1×10^{-7} to 1×10^{-6} oz/ft^3] for static to lightly disturbed conditions on a basaltic tephra-fall deposit (Hill and Connor, 2000). Part of this difference arises from the DOE assumptions that light surface-disturbing activities, such as walking, do not cause an increase in airborne mass load (Bechtel SAIC Company, LLC, 2003d). Additional data provided in, for example, Bechtel SAIC Company, LLC (2003b), show that light levels of surface-disturbing activity commonly result in elevated mass loads. Thus, DOE does not appear to consider an appropriate range of activities by the receptor in calculating the potential inhalation doses for time spent outdoors performing light levels of surface-disturbing activity (Bechtel SAIC Company, LLC, 2003d). This range of activity is not accounted for by the uncertainty in the airborne mass loads used in the volcanism inhalation dose calculations (Bechtel SAIC Company, LLC, 2003f). This concern was originally raised as part of Igneous Activity Key

Technical Issue Agreement 2.11 (Reamer, 2001b). Information provided by DOE to address this concern (Bechtel SAIC Company, LLC, 2003a) is being assessed by NRC.

Indoor mass loads have similar levels of significance as outdoor mass loads in the DOE dose calculations. This relationship arises because the receptor is a composite of four different population groups, each spending significantly more time indoors than outdoors (Bechtel SAIC Company, LLC, 2003d). Indoor mass loads are comparable to slightly lower than mass loads used by DOE for outdoor inactive conditions (Bechtel SAIC Company, LLC, 2003b). Thus, the longer calculated indoor exposure times offset the relative decreases in airborne mass loads and breathing rates. Current uncertainties in the range of airborne mass loads for indoor conditions do not appear to encompass the range of mass loads appropriate for the types of activities representative of the specific activities associated with the receptor. This concern was originally raised as part of Igneous Activity Key Technical Issue Agreement 2.11 (Reamer, 2001b). DOE has provided information to address this issue (Bechtel SAIC Company, LLC, 2003a).

Current conditions at the receptor location do not represent the range of physical conditions important to determine airborne mass load in performance calculations. Currently, the receptor location is described generally by the uninhabited areas within several kilometers of the Fortymile Wash drainage, along the southern boundary of the Nevada Test Site (Bechtel SAIC Company, LLC, 2003d). Performance calculations, however, assume a stylized individual (i.e., the receptor) will inhabit this area as part of a larger, surrounding community (10 CFR Part 63). Thus, a range of surface-disturbing conditions are part of this projected inhabitation, which reasonably could affect the resulting airborne mass loads. Thus, airborne mass loads recently measured near the receptor location (Bechtel SAIC Company, LLC, 2003b) do not represent the range of conditions expected to affect airborne mass loads in an area with surface disturbance.

To evaluate changes in the rate of decrease in airborne mass loads following a potential volcanic eruption, DOE is expected to evaluate the effects of tephra redistribution in the Fortymile Wash drainage system (Reamer, 2001a). Although DOE has not presented the details of a tephra redistribution model in Bechtel SAIC Company, LLC (2003a,b,h,i), data cited in Bechtel SAIC Company, LLC (2003b) to support this model rely heavily on analog studies. The analog areas presented in Bechtel SAIC Company, LLC (2003b) are not located where significant influx of redistributed tephra would be received through time. Thus, parameters derived from measurements of mass loads in these areas do not evaluate uncertainties associated with potential tephra redistribution processes through time. To account for these uncertainties, Bechtel SAIC Company, LLC (2003b) increases the minimum and mode in the parameter distribution for the amount of time after a potential volcanic eruption necessary for the airborne mass load to return to preeruption levels. Although this approach appears conservative when compared with the analog data, these data are of limited use because the analog sites do not consider the effects of potential tephra redistribution processes. As discussed in Section 5.1.3.13.4.1 of this report, the large potential amounts of tephra and low ambient sediment flux in the Fortymile Wash drainage system are not consistent with assertions in Bechtel SAIC Company, LLC (2003b) that tephra would be well mixed with other sediment and only affect airborne mass loads for a short amount of time. Thus, parameters used to derive the mass load decay function in Bechtel SAIC Company, LLC (2003b) have not considered the full range of uncertainty resulting from the potential effects of tephra redistribution processes in Fortymile Wash. DOE has agreed to provide additional information on this topic (Reamer, 2001a).

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.13.5), is sufficient to expect that the information necessary to assess redistribution of radionuclides in soil with respect to data uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.13.4.4 Model Uncertainty

DOE has not yet documented its final model for the potential long-term redistribution of tephra in the Fortymile Wash drainage system. Preliminary analyses in Bechtel SAIC Company, LLC (2003a,b,h,i) present only limited amounts of data used to support development of the unpublished redistribution model. The staff has identified some concerns (i.e., Section 5.1.3.13.4.1) that these data do not represent adequately the range of physical conditions likely to exist in the years following deposition of a potential volcanic tephra-fall deposit. DOE has agreed to provide to NRC the technical basis for this model (Reamer, 2001a).

NRC currently is evaluating the potential redistribution of contaminated tephra in the Fortymile Wash drainage system (Hooper, 2004). Although sediment erosion, transport, and deposition rates in arid regions are not well known, a sediment budget can be constructed to account for redistribution processes in the Fortymile Wash drainage system (Hooper, 2004). Important model sensitivities are the erosion rate and the thickness of potential tephra deposits within the watershed. Preliminary model results, however, indicate the flux of redistributed tephra in Fortymile Wash near the receptor location appears significantly higher than fluxes implied in Bechtel SAIC Company, LLC (2003a,b,h,i).

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.13.5), is sufficient to expect that the information necessary to assess redistribution of radionuclides in soil with respect to model uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.13.4.5 Model Support

DOE has not yet documented its final model for the potential long-term redistribution of tephra in the Fortymile Wash drainage system. Preliminary analyses in Bechtel SAIC Company, LLC (2003a,b,h,i) present only limited amounts of information regarding the unpublished DOE redistribution model. Staff has identified some concerns in Section 5.1.3.13.4.1 that this information does not characterize adequately the potential long-term tephra redistribution processes in the Fortymile Wash drainage system. DOE has agreed to provide to NRC the technical basis for this model (Reamer, 2001a).

Currently, the Fortymile Wash drainage system is characterized by low sediment production and erosion rates, with episodic floods depositing the bulk of the transported sediments close to the receptor location. Potential deposition of a tephra-fall deposit in parts of this drainage system may significantly increase sediment erosion, transport, and deposition rates (e.g., Hooper, 2004). Thus, current conditions in the Fortymile Wash drainage system are of limited utility for testing or supporting a model for potential tephra redistribution processes. Traces of tephra-fall deposits in the Yucca Mountain region are extensively eroded and do not provide useful insights on potential redistribution processes likely to occur in the years to centuries following a possible volcanic eruption. Tephra-fall deposits in nonarid areas (Hooper, 2004) redistribute according to site-specific erosion, transport, and deposition rates, which often are not analogous to arid land processes in the Yucca Mountain region. DOE should account for physical processes characteristic of the Fortymile Wash drainage system following a potential volcanic event to support its assumptions for decreases in airborne mass load following a potential volcanic eruption (e.g., Bechtel SAIC Company, LLC, 2003a,i). DOE should present and support a model that accounts for physical processes characteristic of the Fortymile Wash drainage system following a potential volcanic event. DOE has agreed to provide to NRC additional information in support of the technical basis for the redistribution model (Reamer, 2001a).

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.13.5), is sufficient to expect that the information necessary to assess redistribution of radionuclides in soil with respect to model abstraction output being supported by objective comparisons will be available at the time of a potential license application.

5.1.3.13.5 Summary and Status of Key Technical Issue Subissues and Agreements

Table 5.1.3.13-1 provides the status of all key technical issue subissues, referenced in Section 5.1.3.13.2, 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 review methods discussed in Section 5.1.3.13.4.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (e.g., specified testing or analyses), indicates that information necessary to begin a technical review will likely be available at the time of a potential license application. Note the status and the detailed agreements pertaining to all the key technical issue subissues are provided in Table 1.2-1 and Appendix A.

Table 5.1.3.13-1. 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.2.11 IA.2.14 IA.2.17
		Closed	IA.2.06 IA.2.07 IA.2.08 IA.2.12 IA.2.13 IA.2.15 IA.2.16

Table 5.1.3.13-1. Related Key Technical Issue Subissues and Agreements (continued)			
Key Technical Issue	Subissue	Status	Related Agreement*
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 TSPAI.3.36
	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 review methods.			

5.1.3.13.6 References

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5.1.3.14 Biosphere Characteristics

5.1.3.14.1 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 ground water 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 Characteristics Integrated Subissue includes the features, events, and processes that affect 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 (i.e., an individual, based on characteristics derived from local populations, that lives in the accessible environment directly above the area of highest radionuclide concentration in the ground water 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 ground water release and airborne fallout resulting from a postulated volcanic event.

The DOE description and technical basis for biosphere dose modeling are documented in Bechtel SAIC Company, LLC (2003a) and various supporting analysis and model reports. Because supporting analysis and model reports were provided by DOE recently, some revised topical areas that did not pertain to prior agreement issues could not be reviewed in detail for this report. Staff will continue to review existing reports and monitor any new DOE documentation, as necessary, in a manner consistent with the importance of the information to risk. Results of forthcoming reviews will be documented in future reports or meetings.

5.1.3.14.2 Relationship to Key Technical Issue Subissues

The Biosphere Characteristics Integrated Subissue is derived from the dose calculation component of the biosphere system (Figure 1.1-2). The relationships between the Biosphere Characteristics Integrated Subissue and other integrated subissues are illustrated in Figure 5.1.3.14-1. 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 addressed in the following key technical issue integrated subissues:

- Radionuclide Transport: Integrated Subissue 3—Radionuclide Transport Through Fractured Rock (NRC, 2000a)
- Igneous Activity: Integrated Subissue 2—Consequences of Igneous Activity (NRC, 1999a)
- Unsaturated and Saturated Flow Under Isothermal Conditions: Integrated Subissue 1—Climate Change (NRC, 1999b

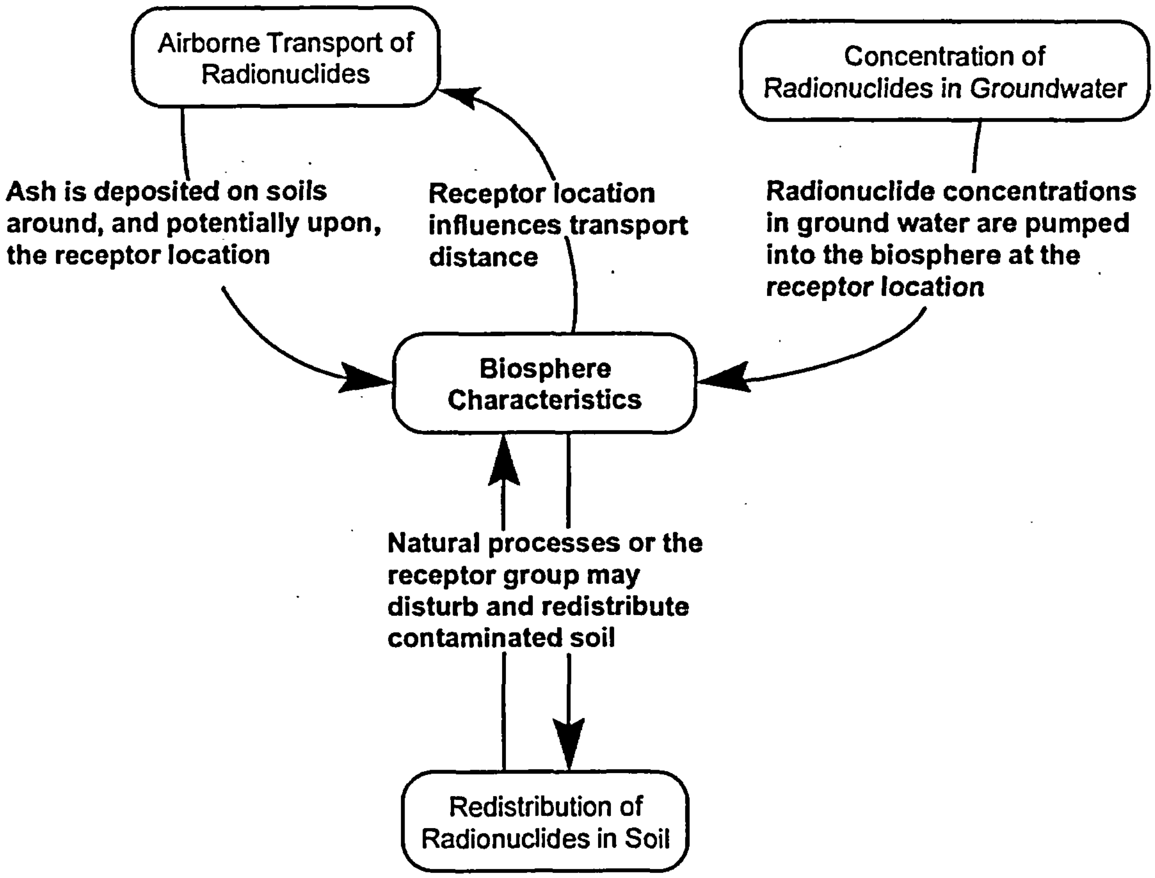

Figure 5.1.3.14-1. Diagram Illustrating the Relationship Between Biosphere Characteristics and Other Integrated Subissues. Material in Bold Is Identified in the Text.

- Unsaturated and Saturated Flow Under Isothermal Conditions: Integrated Subissue 2—Hydrologic Effects of Climate Change (NRC, 1999b)

- Unsaturated and Saturated Flow Under Isothermal Conditions: Integrated Subissue 3—Present Day Shallow Groundwater Infiltration (NRC, 1999b)

- Unsaturated and Saturated Flow Under Isothermal Conditions: Integrated Subissue 5—Saturated Zone Ambient Flow Conditions and Dilution Processes (NRC, 1999b)

- Total System Performance Assessment and Integration: Integrated Subissue 1—System Description and Demonstration of Multiple Barriers (NRC, 2000b)

- Total System Performance Assessment and Integration: Integrated Subissue 2—Scenario Analysis and Event Probability (NRC, 2000b)

- Total System Performance Assessment and Integration: Integrated Subissue 3—Model Abstraction (NRC, 2000b)

- Total System Performance Assessment and Integration: Integrated Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards (NRC, 2000b)

The key technical issue integrated 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 integrated subissue. The resolution status of this integrated subissue is based on the resolution status of each contributing key technical issue integrated subissue. The subsequent sections incorporate applicable portions of these key technical issue integrated subissues. Topical overlap exists between the Biosphere Characteristics Integrated Subissue and the Redistribution of Radionuclides in Soil Integrated Subissue. To facilitate organization of staff reviews and resulting documentation, biosphere modeling topics regarding remobilization, mass loading, and exposure times associated with the igneous disruptive event scenario are addressed in Section 5.1.3.13, Redistribution of Radionuclides in Soil.

5.1.3.14.3 Importance to Postclosure Performance

One aspect of risk informing the NRC review was to determine was how this integrated subissue is related to the DOE repository safety strategy. DOE initially determined 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 (2000a). This change in significance was attributed to the small variation in the mean values for biosphere dose conversion factors. The DOE and NRC performance assessment models both propagate a small and comparable amount of variation in the biosphere abstraction. DOE has documented the variability in biosphere model results (biosphere dose conversion factor distributions) for most radionuclides to be nearly a factor of two above and below the mean of the distribution (Bechtel SAIC Company, LLC, 2003a). Risk insights indicate that characterization of the biosphere is of low significance to waste isolation. The details of the risk insights ranking are provided in Appendix D.

5.1.3.14.4 Technical Basis

NRC developed a review plan (NRC, 2003) consistent with the acceptance criteria and review methods found in previous issue resolution status reports. A status assessment of the DOE approaches for including biosphere characteristics in total system performance assessment abstractions is provided in the following subsections. This assessment is organized according to the five review methods identified in Section 2.3: (i) Model Integration (including system description), (ii) Data and Model Justification, (iii) Data Uncertainty, (iv) Model Uncertainty, and (v) Model Support.

5.1.3.14.4 1 Model Integration

Although the overall significance of biosphere characteristics in total system performance calculations is ranked low, staff need to verify that system description and model integration are adequate to demonstrate compliance with specific biosphere requirements and support the DOE biosphere calculations.

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 5.1.2 of this report. Therefore, this section will concentrate on 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 10 CFR Part 63 requirements. Some important characteristics of the biosphere and dose receptor have been explicitly defined in 10 CFR Part 63 requirements. Although DOE is not required to justify characteristics of the biosphere and dose receptor defined explicitly in the regulation (e.g., drinking water consumption rate), supporting information is needed to define characteristics not explicitly defined in 10 CFR Part 63 (e.g., irrigation rates, food consumption, and outdoor activity).

Since the last staff review, DOE completely revised documents that describe the biosphere and dose receptor characteristics. The new documentation contains more detailed information on all aspects of the biosphere modeling, and improvements have been made to format and content. A general description of the biosphere and dose receptor is provided in Bechtel SAIC Company, LLC (2003a). More detailed technical information is provided in a series of analysis and model reports addressing specific aspects of the biosphere and dose receptor. In general, these reports provide an adequate system description 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 identified during prior reviews have been addressed by DOE or await resolution by the DOE response to remaining open 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. Detailed discussions of specific technical areas including support for establishing the characteristics 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.

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 the DOE documentation (Bechtel SAIC Company, LLC, 2003b) indicates demographic surveys of Amargosa Valley have been completed and documented, and the results are incorporated into the biosphere dose modeling as mean value parameters. 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 Bechtel SAIC Company, LLC (2003b), DOE documents the use of adult dosimetry in its application of dose coefficients from existing U.S. Environmental Protection Agency (EPA)

Federal Guidance reports (1993, 1988) that NRC uses and accepts for dose modeling. DOE also indicates the location of the dose receptor will likely be at the nearest location in the accessible environment to the south of Yucca Mountain to satisfy the 10 CFR Part 63 requirement that the reasonably maximally exposed individual live in the accessible environment above the highest concentration in the plume of contamination (Bechtel SAIC Company, LLC, 2003b).

The general description of the biosphere dose modeling provided in Bechtel SAIC Company, LLC (2003a) includes a dose receptor and biosphere intended to be consistent with the NRC regulations. The receptor is described as a hypothetical individual with dietary and lifestyle characteristics based on mean values of the Amargosa Valley population. The receptor is presumed to be exposed to radionuclide releases to ground water (nominal scenario) and air (for the disruptive volcanic event scenario). The reference biosphere is based on characteristics of Amargosa Valley that include a climate characterized as arid to semiarid (considering potential future climate evolution). Census data and results of a survey of local residents provide information on the lifestyle characteristics of people in the region. Alfalfa production and dairy farming are noted as primary agricultural activities in the area, although DOE reports additional food crops and residential gardening. Water for all uses in the area comes predominantly from local wells. Detailed information on local employment provides additional lifestyle characteristics. The staff believes that sufficient information on biosphere characteristics is documented 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 potential repository are transported to 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 contribute directly to exposure of the human dose receptor. This model includes transfer of radionuclides to soil, atmosphere, and flora and fauna (Bechtel SAIC Company, LLC, 2003a). 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 results from contact with contaminated air, water, food products (both plant and animal), and soil. Local practices, such as use of evaporative coolers and fish farming, have been included in the DOE exposure scenario. The staff previously identified an additional transport mechanism for the volcanic scenario involving redistribution of contaminated ash deposits. Redistribution in the biosphere is included in another integrated subissue (Redistribution of Radionuclides in Soil) and is addressed by an existing agreement (Section 5.1.3.13), 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 existing information. Results of 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 (Reamer, 2001).

Integration with related integrated subissues was evident from reviews of the DOE biosphere abstraction. Numerous biosphere modeling issues related to the igneous activity scenario are receiving technical input from Igneous Activity Integrated Subissue 2 (e.g., redistribution and mass loading). DOE included the effects of natural climate change on biosphere dose conversion factors in Bechtel SAIC Company, LLC (2003a). DOE 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 understood sufficiently 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 any major 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 that appears to be 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 found generally to be comprehensive for the biosphere. At the general conceptual model level, it is unlikely any additional features, events, or processes significant to the dose calculation will be identified after resolution of existing agreements. At a more detailed submodel level, some models may be optimized or updated; however, these modifications are not expected to change significantly the overall conceptual model of the biosphere.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.14.5), is sufficient to expect that the information necessary to assess biosphere characteristics with respect to system description and model integration will be available at the time of a potential license application.

5.1.3.14.4.2 Data and Model Justification

The overall significance of the Biosphere Characteristics Integrated Subissue in total system performance calculations is ranked low. This low ranking indicates staff will limit the depth of review to verify data, and models are adequately justified by focusing on key areas known to be important in the process level modeling. The DOE and NRC calculations indicate for those radionuclides that dominate ground water dose calculations, the drinking water consumption pathway contributes at least approximately half the dose [NRC Risk Insights Baseline Report (Appendix D)]. Because the regulations in 10 CFR Part 63 specify the use of 2 L/d [.5 gal/d] in the drinking water pathway dose calculation, the remaining half of the dose calculation (i.e., the nondrinking water pathways), influenced to a greater degree by parameter selection and variability, is emphasized in the staff review. The DOE biosphere calculations require a large number of parameter selections. Input parameters for the biosphere calculations are documented in analysis and model reports the NRC staff has reviewed at various levels of detail, depending on the importance to modeling results. Both DOE (CRWMS M&O, 2000b,c) 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, and the crop interception fraction. Other important biosphere parameters topically linked to soil redistribution processes applicable to the igneous disruptive event release scenario (e.g., mass loading, soil distribution coefficients, and exposure times) are discussed in Redistribution of Radionuclides in Soil (Section 5.1.3.13).

DOE selected a series of mathematical models for the biosphere dose modeling consistent with the key features, events, and processes included in the biosphere conceptual model for Yucca Mountain. A new biosphere model, Environmental Radiation Model for Yucca Mountain Nevada (ERMYN) (Bechtel SAIC Company, LLC, 2003c), has been developed to include most mathematical models used in the GENII-S dose modeling software program (Leigh, et al., 1993; Napier et al., 1988). Additional models have been included in ERMYN that address those Yucca Mountain features, events, and processes not considered in the GENII models. Extensive documentation and testing of the ERMYN biosphere model has recently been provided (Bechtel SAIC Company, LLC, 2003c). NRC has not identified any major problems with the mathematical models or justification; however, document reviews are ongoing and the DOE resolution of existing agreements may result in the use of new models for specific biosphere processes [e.g., redistribution (Section 5.1.3.13)].

Detailed biosphere parameter information is provided by DOE in a series of analysis and model reports. In general, these reports provide comprehensive documentation of the bases for selection of parameter values for the biosphere dose modeling. The following paragraphs provide results of staff reviews regarding the DOE approach to parameter justification for those parameters identified to be important in the process level biosphere modeling.

The DOE mean value consumption rates are supported by results of a stratified random sample survey of the local population (Bechtel SAIC Company, LLC, 2003b). The survey included the population residing within 84 km [52 mi] of Yucca Mountain (the communities of Amargosa Valley, Beatty, Indian Springs, and Pahrump). Information was collected on the consumption frequency of locally produced food and water, 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. Descriptions of the survey methodology, execution, and analysis of results in Bechtel SAIC Company, LLC (2003b) provide a transparent and traceable basis for the consumption rate parameter information.

In prior reviews of the DOE documents, the staff requested DOE provide additional documentation regarding the technical bases for selected parameter values for plant and animal transfer coefficients and for crop interception fractions. These requests were tracked as issue resolution agreements TSPAI.3.34 and TSPAI.3.35. DOE subsequently responded to these agreements (Bechtel SAIC Company, LLC, 2003a), and NRC later documented the resolution status (Schlueter, 2004). The following paragraphs briefly summarize the technical issues contained in the agreements and how the agreements were resolved by DOE.

Agreement TSPAI.3.34 requested the technical bases for selection of transfer coefficients that include plant and animal uptake (from soil) factors. The source information referenced by DOE in its response (Bechtel SAIC Company, LLC, 2003d) incorporated data obtained by a combination of available techniques, including a variety of laboratory and field studies. DOE supplemented the information with more site-specific information from local field and laboratory experiments conducted at the Nevada Test Site. The selection of transfer coefficients also was informed by other site-specific information where possible, including soil type and applicable crop types. The staff found documentation was sufficiently detailed to identify the data sources. The supporting documentation also was sufficiently detailed to allow the staff to reproduce the transfer coefficient estimates derived by the source data. Additional details of the staff review are documented in Schlueter (2004).

Agreement TSPAI.3.35 requested the technical bases for crop interception fractions used to estimate radioactive contamination in irrigation water deposited on plant surfaces. In responding to the agreement, DOE updated the model used to calculate the crop interception fractions using an experimentally derived process model (Bechtel SAIC Company, LLC, 2003a) that produces results consistent with the available laboratory and field studies [e.g., studies reported in Anspaugh (1987)]. Resulting mean values for crop interception fractions appear unlikely to underestimate interception of radionuclides by crops. Considering the available information and physical constraints of the parameter (range from 0 to 1.0), the staff found the calculated values were not likely to underpredict actual interception conditions in Amargosa Valley and considered the agreement satisfactorily resolved. Additional details of the staff review are documented in Schlueter (2004).

The aforementioned agreements were developed for issues where the initial DOE responses to staff concerns were incomplete. Some initial DOE responses to staff concerns were initially adequate and, therefore, did not result in the creation of agreements, yet included DOE action items to be completed in the future. The action items related to biosphere 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, Transfer Coefficient Analysis, to include methods for combining data based on individual crops to food groups and include a clarified definition of conservatism; and (iv) complete additional model validation for the GENII-S code (Leigh, et al., 1993). These items have been addressed or superseded by the new DOE documentation, therefore, staff will not continue to track them.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.14.5), is sufficient to expect that the information necessary to assess biosphere characteristics with respect to data being sufficient for model justification will be available at the time of a potential license application.

5.1.3.14.4.3 Data Uncertainty

The NRC Risk Insights Baseline Report (Appendix D) and DOE (Bechtel SAIC Company, LLC, 2003a) biosphere analyses both propagate input parameter uncertainties and variability in their respective biosphere dose calculations. The corresponding model output variation is similar in both models—approximately a factor of two above and below the mean value for biosphere dose conversion factors (Bechtel SAIC Company, LLC, 2003a). This output variability is low relative to other model abstractions that can contribute more than one order of magnitude variation in total system performance assessment results. Low variability contributes to the low significance ranking for ground water pathway biosphere calculations [NRC Risk Insights Baseline Report (Appendix D)] because input parameter changes do not produce large changes in calculated doses. Given the low variability propagated in the biosphere calculations, it is important staff verify DOE has documented the sources of uncertainty included or addressed in the biosphere calculations. Those biosphere parameters most likely to contribute significantly to uncertainty in model output are the aforementioned parameters identified in prior sensitivity studies.

As described in Bechtel SAIC Company, LLC (2003a), DOE propagates biosphere dose modeling input parameter variability and uncertainty by executing the ERMYN model (Bechtel SAIC Company, LLC, 2003c) interactively using vectors of sampled input parameters to generate corresponding vectors of radionuclide-specific biosphere dose conversion factors (i.e., annual dose per unit ground water concentration). The vectors of radionuclide-specific biosphere dose conversion factors are then randomly sampled for each realization of the total system performance assessment model. This approach to propagating biosphere variability is an improvement to the prior DOE approach, which involved random sampling from radionuclide-specific probability distributions of biosphere dose conversion factors. The prior DOE approach generated staff concerns regarding the potential to introduce bias. These concerns are documented in Agreement TSPAI.3.37.

DOE provided detailed documentation of the technical bases for selecting parameter distributions for important biosphere input parameters. Bechtel SAIC Company, LLC (2003d) describes the technical bases for selecting distributions of plant and animal uptake factors. Plant factor distributions, for example, were selected by calculating the geometric means and standard deviations of reported best estimate values from several source documents that summarized literature values. Truncated log normal distributions were then derived by calculating a 99-percent confidence interval using the calculated means and standard deviations. Establishing limits on the geometric standard deviations ensured calculated uncertainty ranges estimated by this approach fell within a reasonable range of uncertainty reported in the literature (2 to 10). The resulting distribution is characterized as representing the uncertainty in the generic composite value parameter rather than uncertainty in point estimates for specific crops. Although staff found the DOE derivation approach unconventional, the resulting parameter distributions fell within reasonable ranges found in available literature, and documentation was sufficiently complete to allow staff to understand fully and verify the input data sources, calculations, assumptions, and results. The DOE derivation of distributions for crop interception fractions was based on stochastic modeling to calculate a range of values. The documentation is sufficiently complete to address the bases for the calculated values. Results generally spanned the range of possible values for this parameter and appeared reasonable. Bechtel SAIC Company, LLC (2003c) provides complete tabulations of all biosphere parameter values and probability distributions.

The DOE documentation includes discussion of parameters correlated in the implementation of the ERMYN biosphere model (Bechtel SAIC Company, LLC, 2003c). These include correlations between soil distribution coefficients and soil to plant transfer factors and correlation of evaporative cooler airflow rate with water evaporation rate. At present, the staff has not identified any concerns regarding the documentation of correlations in the DOE biosphere model.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.14.5), is sufficient to expect that the information necessary to assess biosphere characteristics with respect to data uncertainty being characterized and propagated through the model abstraction will be available at the time of a potential license application.

5.1.3.14.4.4 Model Uncertainty

Current modeling by DOE (Bechtel SAIC Company, LLC, 2003a) and the NRC Risk Insights Baseline Report (Appendix D) suggests drinking water is the predominant exposure pathway for key radionuclides in ground water-based dose calculations. Because the biosphere component of the drinking water dose calculation is simple and constrained by regulatory requirements, staff do not expect use of alternative biosphere models would significantly change the magnitude of all-pathway dose estimates. Therefore, quantification of biosphere model uncertainty for a ground water release scenario does not appear to be necessary for a staff review of the DOE license application.

Biosphere dose modeling is a highly abstracted and idealized type of modeling. Many available models for biosphere dose calculations are based on similar conceptual models and mathematical representations. 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. To date, staff has not identified any risk significant biosphere submodels that warrant consideration of alternatives. Similarly, other than those issues addressed by related integrated subissues, staff has not identified any parts of the biosphere dose modeling where model uncertainty comparisons would help inform the review of the DOE safety case. An emphasis on propagation of parameter uncertainty is more appropriate for the type of modeling conducted for the biosphere.

To enhance model confidence, DOE conducted numerous model comparisons between the ERMYN model and other available biosphere models (Bechtel SAIC Company, LLC, 2003c). These comparisons are discussed further in the subsection on model support (Section 5.1.3.14.5).

Overall, the available information is sufficient to expect that the information necessary to assess biosphere characteristics with respect to model uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.3.14.4.5 Model Support

The DOE biosphere dose modeling abstraction consists of the biosphere dose conversion factor vectors, the approach for sampling these vectors for each realization, and the routine that multiplies estimated soil and ground water radionuclide concentrations by the sampled factors to calculate dose. The biosphere dose conversion factor vectors are generated by DOE from process modeling using the ERMYN model (Bechtel SAIC Company, LLC, 2003c). DOE has made comparisons to improve confidence that the modeling in the abstraction is being performed correctly. First, the ERMYN model equations were transferred to a spreadsheet to verify the calculations (Bechtel SAIC Company, LLC, 2003c). Extensive documentation of model validation activities and results also is provided in Bechtel SAIC Company, LLC (2003c). The validation approach is based on corroboration of the conceptual approach, mathematical representation, and comparison of results with five other available biosphere models, including the previous site recommendation biosphere model. Overall, the verification and validation activities provide confidence the models are operating as expected. Documentation of input parameters, assumptions, mathematical models, and results are sufficient to allow staff to verify or reproduce results, if necessary. The staff could not locate verification of the implementation of ERMYN in the total system performance assessment model, however, has indicated that this will be forthcoming in total system model documentation.

In summary, the nature of the abstraction provides a basis for comparisons with process model results. The DOE documentation includes comparisons that indicate that the new biosphere model is operating as expected.

Overall, the available information is sufficient to expect that the information necessary to assess biosphere characteristics with respect to model abstraction output being supported by objective comparisons will be available at the time of a potential license application.

5.1.3.14.5 Summary and Status of Key Technical Issue Subissues and Agreements

Table 5.1.3.14-1 provides the status of all key technical issue integrated subissues referenced in Section 5.1.3.14.2 for the Biosphere Characteristics Integrated Subissue. The table also provides the related DOE and NRC agreements pertaining to the Biosphere Characteristics Integrated Subissue. The agreements listed in the table are associated with one or all five generic review methods discussed in Section 5.1.3.14.4. Note the status and the detailed agreements pertaining to all the key technical issue integrated 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 (e.g., specified testing or analyses), indicates that information necessary to begin a technical review will likely be available at the time of a potential license application.

Table 5.1.3.14-1. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Integrated Subissue	Status	Related Agreement*
Radionuclide Transport	Subissue 3—Radionuclide Transport Through Fractured Rock	Closed-Pending	None
Igneous Activity	Subissue 2—Consequences of Igneous Activity	Closed-Pending Except IA.2.15 (closed)	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
	Subissue 2—Hydrologic Effects of Climate Change	Closed-Pending	None
	Subissue 3—Present Day Shallow Groundwater Infiltration	Closed-Pending	None

Table 5.1.3.14-1. Related Key Technical Issue Subissues and Agreements (continued)			
Key Technical Issue	**Integrated Subissue**	**Status**	**Related Agreement***
Unsaturated and Saturated Flow Under Isothermal Conditions	Subissue 5—Saturated Zone Ambient Flow Conditions 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 through TSPAI.2.04
	Subissue 3—Model Abstraction	Closed Except TSPAI.3.37 (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 review methods.			

5.1.3.14.6 References

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5.1.4 Demonstration of Compliance with the Postclosure Public Health and Environmental Standards

5.1.4.1 Demonstration of Compliance with the Postclosure Individual Protection Standard

5.1.4.1.1 Description of Issue

The DOE analysis of repository performance will be reviewed to ensure that it provides the required information and demonstrates compliance with the postclosure individual protection standard at 10 CFR 63.311. 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.

The NRC staff will evaluate the adequacy of the total system performance assessment presented in the potential license application to ensure the technical requirements at 10 CFR 63.114 are satisfied. If the performance assessment is determined to be adequate, the staff will determine whether there is a reasonable expectation, as defined in 10 CFR 63.304, that the repository will comply with the individual protection standard set out in 10 CFR 63.311. During the prelicensing period, only the first of these two evaluations is performed by the NRC staff.

This section documents the current NRC understanding of the DOE approach to demonstrating compliance with the postclosure individual protection standard by means of a performance assessment, as set out in 10 CFR 63.113(b) and 10 CFR 63.114. The assessment is focused on those aspects most important to repository safety based on risk insights gained to date from, for example, previous total system performance assessments, including independent analyses using the TPA Version 4.1 code (Mohanty, et al., 2002). The NRC review is limited to determining if the methodology developed by DOE is likely to be adequately documented for the staff to undertake a detailed technical review. This assessment is not a regulatory compliance determination review of a potential license application.

5.1.4.1.2 Relationship to Key Technical Issue Subissues

To adequately demonstrate compliance with the postclosure individual protection standard, an analysis of repository performance must appropriately incorporate scenarios into the total system performance assessment, properly conduct the total system performance assessment, and appropriately combine the results and compare them with the regulatory limits. This subissue is related to all key technical issue subissues because the DOE total system performance assessment must identify and incorporate scenarios and data analyses for conceptual model development and validation, which are the focal points of these key technical issues. Past reviews are 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
- Repository Design and Thermal-Mechanical Effects
- Unsaturated and Saturated Flow Under Isothermal Conditions
- Radionuclide Transport
- Total System Performance Assessment and Integration

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.

5.1.4.1.3 Importance to Postclosure Performance

This issue relates to the methodology used to calculate the performance of the potential repository system at Yucca Mountain and to compare the results of the DOE total system performance assessment with the regulatory requirements. Therefore, this issue directly relates to the determination of postclosure safety of the repository.

In addition to calculating the performance at Yucca Mountain for 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. The definition of performance assessment at 10 CFR 63.2 indicates that estimates of dose from disruptive events should be weighted by probability of occurrence when included in the calculation of dose to the reasonably maximally exposed individual.

5.1.4.1.4 Technical Basis

NRC developed a review plan (NRC, 2003) consistent with acceptance criteria and review methods found in the previous issue resolution status reports. A status assessment of the DOE approach to demonstration of compliance with the postclosure individual protection standard is provided in the following subsections of this report. This assessment is organized according to the three review methods identified in Section 2.2.1.4.1 of the review plan: (i) Appropriate Incorporation of Scenarios into the Total System Performance Assessment Results, (ii) Calculation of the Total Effective Dose Equivalent from the Repository System, and (iii) Credibility of the Total System Performance Assessment Results.

5.1.4.1.4.1 Appropriate Incorporation of Scenarios into the Total System Performance Assessment Results

The approach and technical basis for appropriately incorporating scenarios into the DOE total system performance assessment were documented in CRWMS M&O (2000a). Based on the results of features, events, and processes analysis, DOE concluded there are two disruptive event classes that could significantly affect repository performance: igneous activity, and seismically induced cladding failure. The probability of extrusive volcanism was 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 was 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 result in an estimate of the probability-weighted dose that can be compared with the 0.15-mSv/yr [15-mrem/yr] all-pathways dose standard in 10 CFR Part 63. DOE did 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, was not weighted by the probability of the nominal scenario class, which is slightly less than one, because the volcanism event class was excluded. The mean probability-weighted dose curve from the disruptive events was added to the conditional nominal case dose to calculate the total effective dose equivalent from the potential repository. The only concern with combining results of the nominal case and the igneous scenario is the same waste packages involved in the igneous event are also counted in the nominal case. Double counting is acceptable, however, 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 (Schlueter, 2000) 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 represented.

A similar process will be followed for the total system performance assessment license application, according to DOE (Bechtel SAIC Company, LLC, 2002a), although it is now expected that seismic vibration of cladding will be transferred from the nominal scenario class to a separate seismic scenario class.

The probability weight to be applied to each class will depend on the assumption of independence of the volcanic and seismic classes, which is still being reviewed by DOE.

Overall, the available information is sufficient to conclude that the information necessary to assess the incorporation of scenarios into the DOE total system performance assessment results will be available at the time of a potential license application.

5.1.4.1.4.2 Calculation of the Total Effective Dose Equivalent from the Repository System

The approach and technical basis for calculating the total effective dose equivalent from the repository system was documented by DOE in CRWMS M&O (2000a). DOE demonstrated the stability of its total system performance assessment results by plotting the time variation of mean dose from the repository system for different numbers of realizations. The NRC staff had concerns this approach was too qualitative and difficult to determine that the results were stable, especially when the dose histories were plotted on a logarithmic scale. The NRC staff found no indication that similar tests were performed for models that provided stochastic inputs 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 provided 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 the 100 transfer functions were sufficient to properly represent uncertainty in the saturated zone transport model. The NRC staff also had concerns because DOE had not provided a methodology to demonstrate its total system performance assessment results were stable with respect to numerical discretization of the model in CRWMS M&O (2000a).

Agreements were reached at technical exchanges (Schlueter, 2000; Reamer, 2001a) wherein DOE was requested to provide documentation describing the method to be employed to demonstrate that the overall results from the total system performance assessment were stable, both numerically and statistically (TSPAI.4.03). Results of the analyses will be provided later in a potential license application (or other appropriate documentation), and documentation of results of further analyses demonstrating numerical stability, with regard to spatial and temporal discretizations, will be provided in a potential license application according to Agreement TSPAI.4.04. DOE submitted a report (Bechtel SAIC Company, LLC, 2002a) that partly addressed the requirements of Agreement TSPAI.4.03 and was the equivalent of the methods and assumptions document promised in the agreement. That report outlined likely methods to be used for a potential license application, together with the likely content and structure of associated documents to show that discretization errors will be examined in both temporal and spatial representations of constituent models and the overall system. Hence, stable and convergent results will be obtained, and statistical convergence and stability of the constituent stochastic models and the overall system used for the Monte Carlo simulation will be examined (through varying sample seeds and sizes), and confidence intervals will be established (by parametric or nonparametric methods, where appropriate).

The NRC staff (Schlueter, 2003a), considered that by addressing the statistical measures DOE intended to use to support its arguments for stability and by starting to describe the components of a potential method, DOE has provided some of the information requested in Agreement TSPAI.4.03; however, further information is needed:

1. A description of the method that will be used to demonstrate numerical stability in the Total System Performance Assessment for the License Application [as indicated in the Bechtel SAIC Company, LLC (2002a) report, DOE has not yet decided on its approach]

2. Documentation that submodels (including those used to develop input parameters and transfer functions) are numerically stable, as requested in the original agreement

Based on the intermediate outputs available in CRWMS M&O (2000b), it appears sufficient information about intermediate outputs in the DOE total system performance assessment will be available to allow the NRC staff to understand how individual components or systems contribute to system performance. Concerns about the consistency between modeling individual components or systems have been documented in Sections 5.1.3.1–5.1.3.14 of this report. Results of the analysis in CRWMS M&O (2000a) seem to be consistent with the performance of individual systems or components.

A further agreement (TSPAI.4.0.2) was mentioned in Bechtel SAIC Company, LLC (2002b), however. That report did not explain how its documentation of the potential license application will justify the representation of distribution coefficients as uncorrelated and will not lead to an underestimation of risk. Subsequently, DOE submitted a report (Bechtel SAIC Company, LLC, 2003) in which Appendix I showed correlations had been derived for sorbing and nonsorbing elements. In a letter dated April 14, 2004 (Reamer, 2004a), the NRC staff concluded dose estimates in both the DOE and NRC performance assessments conducted to date were dominated at 10,000 years after closure by two nonsorbing radionuclides (technetium and iodine) and by weakly sorbing neptunium. Hence, correlations among transport parameters for sorbing radionuclides would be likely to have low overall significance with respect to risk. The information provided by DOE is sufficient to regard Agreement TSPAI.4.02 as complete.

Overall, the available information, along with key technical issue agreements between DOE and NRC, is sufficient to expect that the information necessary to assess calculation of the total effective dose equivalent from the repository system will be available at the time of a potential license application.

5.1.4.1.4.3 Credibility of the Total System Performance Assessment Results

Reasonable expectation of meeting the postclosure individual protection standard in 10 CFR 63.113(b), as required at 10 CFR 63.304, can only be achieved if the total system performance assessment code and its associated inputs have sufficient technical credibility. Risk insights can be gained from earlier performance assessments if they have a degree of credibility appropriate to the level of information and understanding available at the time. This subsection is concerned with the DOE methodologies to achieve a correct and defensible combination of individual model components and associated data to form the integrated system representation and its use in the probabilistic simulations likely to be undertaken for the envisaged license application. These individual components and data would need to be acceptable on the basis of staff reviews conducted in accordance with the 14 model abstraction subissues described in Sections 5.1.3.1–5.1.3.14 of this report. These reviews would employ the methods identified in the corresponding Section 2.2.1.3 of NRC (2003).

In CRWMS M&O (2000a,c, 1999), DOE documented the approach and technical basis for credibility of its total system performance assessment results. Concerns about the consistency among assumptions in different individual modules of the performance assessment code have been documented in Sections 5.1.3.1–5.1.3.14 of this report. DOE indicated its total system performance assessment 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. Verification will use an independent review process to check a tabular form that lists the different elements of the conceptual models and records how they were incorporated into the DOE total system performance assessment. The second phase was designed to ensure the GoldSim model (a registered trademark of Golder Associates Inc.) (GoldSim Technology Group, 2003) 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 GoldSim Technology Group for GoldSim and is specifically related to the Yucca Mountain model. This phase contains 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 data files and GoldSim arguments, and stand-alone codes 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 of a dynamically linked library to an output file and comparing these inputs with the correct values as output from the upstream dynamically linked library.

The NRC staff has concerns about the DOE validation and verification of the total system performance assessment code. The verification process should demonstrate (i) the models used have been adequately tested for calculational correctness with all relevant data and 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 benchmarking or comparative testing against results from other software for cases where accuracy of the code cannot be judged otherwise. DOE included the elements of verification in its total system performance assessment for site recommendation and supporting documents, but did not rigorously verify the modules and the full code or adequately report the results. A specific verification plan was not provided, and the verification was not uniform as presented in CRWMS M&O (2000a). Furthermore, the NRC review of CRWMS M&O (2000b) found errors in verification of the hand calculations and abstractions in the performance assessment that were being used or applied outside the intended ranges (Reamer, 2001b). Verification was performed only on a median input value run without rationale to justify this verification is sufficient for a probabilistic model. CRWMS M&O (2000b) included various levels of analyses to demonstrate 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 a hierarchical manner.

DOE indicated models used within the total system performance assessment for the license application will be validated in accordance with AP–3.10Q (CRWMS M&O, 2000b). This procedure requires comparing analysis results against data acquired from the laboratory, field experiments, natural and manmade analog studies, or other relevant observations to validate models used in the total system performance assessment. The procedure also requires 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 using and documenting an alternative approach. Alternative approaches may include one or more of the following activities: (i) peer review or review by international collaborations (e.g., Nuclear Energy Agency); (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 obtained during performance confirmation studies.

The NRC staff has concerns about the steps DOE performed to build confidence in its total system performance assessment models are as follows. Confidence building in models should include demonstrating (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 the total system performance assessment for site recommendation model (CRWMS M&O, 2000a). A model validation plan did not appear to exist, however, at the time of the review. Rigorous model validation at the system level did not appear to have been conducted or had not been adequately reported. For example, the discussion of validation of the mathematical model of the biosphere (GENII-S) (Leigh, et al., 1993) included only aspects of software verification. DOE collected field and laboratory data to support detailed hydrological calculations from which abstractions are made when representing the data in tabular form for use in performance assessments. That report (Leigh, et al., 1993) did 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 had not been made for all the constituent models, such as validating the colloidal transport model with data from the C-Wells Testing Complex. The DOE audits of the total system performance assessment program identified problems with the validation of models, and DOE issued a corrective action report (Bechtel SAIC Company, LLC, 2001a) to address these problems.

The NRC staff presented these concerns to DOE, and general agreements were reached at a DOE and NRC meeting (Reamer, 2001a). For TSPAI.4.05, DOE agreed to document the process used to develop confidence in the total system performance assessment models, such as described in NRC (1999) and for TSPAI.4.07, to document compliance with the improved process in the verification documentation required by AP–SI.1Q (DOE, 2001). DOE has also agreed (Reamer, 2001a), under TSPAI.4.06, to document the implementation of the process for model confidence building and demonstrate compliance with model confidence criteria in accordance with the applicable procedures.

Subsequent to these agreements, DOE outlined an approach for both verification and validation of the total system performance assessment for license application model and software implementation as part of an overall plan to ensure confidence in the estimation of regulatory performance variables (Bechtel SAIC Company, LLC, 2002a).

Model validation, which compares model results with observations but does not use data already used to calibrate that model, is somewhat difficult because of the long time scales involved. DOE proposed methods to address this problem and, according to discussion in Bechtel SAIC Company, LLC (2002a), expects to rely primarily on

- Comparison of results from an independent total system performance assessment model
- Intensive scientific and technical review by (it is presumed) individuals not directly involved in the design, implementation, and calibration of the total system performance assessment license application

Compared with information in previous DOE reports (CRWMS M&O, 2000a,c), the use of natural analogs for validation and confidence building appeared to be reduced. Verification proposals now included the use of runs representative of higher doses contributing most to overall risk, for example, conditions related to the 95th percentile dose. DOE released a report (Bechtel SAIC Company, LLC, 2002b) to direct attention to those processes and barriers that are most significant to risk and, hence, should be supported by the most verification and validation. For example, DOE presented the results of many simulations using supplemental models (Bechtel SAIC Company, LLC, 2001b) for nominal and disruptive event scenarios showing the effects of changes to individual assumptions and barrier performance on dose estimates.

In Ziegler (2002a), DOE provided information pertaining to Agreement TSPAI.4.05. By providing update AP–SIII.10Q (DOE, 2002) to the DOE model validation procedures, DOE satisfied the intent of Agreement TSPAI.4.05. However, the NRC staff (Schlueter, 2002b) has six observations for DOE to consider as it implements the new procedure:

1. Adequate confidence in the models used in the performance assessment should exist at the time the performance assessment documentation is issued or the results are relied on. Continuing confidence-building efforts to include performance confirmation activities (i.e., confirming model results or reaffirming appropriateness of the model) is appropriate, provided sufficient confidence-building measures are in place and result in an adequate level of confidence.

2. Currently, one (or more) of several approaches may be used to build confidence in a model. Some approaches, taken individually, are insufficient to yield adequate model confidence, therefore, using a combination of the approaches is acceptable.

3. The more objective approach (e.g., corroboration with data not used in model development) typically yields greater model confidence than the more subjective approaches such as peer review and technical review. Objective confidence-building measures should be used, where possible, in place of more subjective measures. If data are reasonably obtainable, corroboration with these data should be used either before or in conjunction with other confidence-building approaches.

4. If reviews are used to build confidence in a model, the review should encompass, to the extent practical, the full body of information necessary to evaluate the model. Information contained in references, relevant data, supporting documents, and alternative models can provide insight into the appropriateness and limitations of a model.

5. Due to the absence of predetermined acceptance criteria in confidence building, the conclusions drawn from model confidence-building efforts will tend to be subjective. Therefore, it is important to document judgments of the usefulness and limitations of those confidence-building measures used.

6. Corroboration of results with alternative mathematical models needs to consider the confidence in these alternative models and how that confidence is reached.

Because DOE has provided the revised AP– SIII.10Q procedure, DOE has satisfied the intent of Agreement TSPAI.4.05, and staff regards that agreement as "complete." Implementation, however, would be monitored as DOE responds to Agreements TSPAI.4.06 and 4.07.

By letter (Ziegler, 2002b), DOE submitted information pertaining to Agreement TSPAI.4.07. This letter stated DOE revised AP–SI.1Q and developed two new procedures [AP–SI.2Q (DOE, 2003a) and AP–SI.3Q (DOE, 2003b)] to provide more specific guidance on qualification, verification, and validation of software. DOE also made a regulatory commitment to retest legacy software by developing a new procedure applicable to legacy software that would apply the key steps of AP–SI.3Q. In response to the NRC comments (Schlueter, 2003b) about the DOE letter, DOE affirmed (DOE, 2004) that (i) the procedure for retesting legacy software has been prepared and (ii) software qualification record packages for 65 codes have been developed in accordance with the new procedures. In addition, DOE stated, "verification of compliance by a review of selected quality assurance records is an activity more appropriately conducted onsite." In response, NRC stated (Reamer, 2004b) staff review of the latest version of relevant procedures suggests DOE has a reasonably complete framework for ensuring adequate verification of software used to support a potential license application. Hence, Agreement TSPAI.4.07 can be considered "complete."

Treatment of scenario and parameter uncertainties described in CRWMS M&O (2000a) appears to be appropriate. The approach outlined in CRWMS M&O (2000c) appears reasonable 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 of the alternative conceptual models is more conservative and using that model in the analysis. The NRC staff has concerns, however, that, in CRWMS M&O (2000a), DOE weights the results of the alternative conceptual models, based on the probability of the model being correct, without an appropriate technical basis for assigning these weights. Additionally, it is not clear to the NRC staff if DOE will analyze the effects of alternative conceptual models for more than one process at a time. The processes may interact with each other and potentially have a greater effect on the results than when analyzed individually through alternative conceptual models. The aforementioned approach (essentially completing a one-off replacement of the conceptual model with an alternative model) leads to difficulties in determining which alternative conceptual models significantly affect risk and which ones do not. When many alternative conceptual models exist for features or processes, the number of combinations of alternative conceptual models at the system level becomes large. To address these concerns, DOE agreed (Reamer, 2001a), for TSPAI.4.01, 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.

Subsequently, DOE responded to Agreement TSPAI.4.01 with a report (Bechtel SAIC Company, LLC, 2002c) that provided guidelines for developing and documenting alternative conceptual models, model abstractions, and parameter uncertainties in the total system performance assessment for the potential license application. In response (Schlueter, 2002a), NRC concluded it was premature to consider the agreement complete because there was no evidence of successful application nor were these guidelines the equivalent of audited quality assurance procedures. The staff further identified six aspects of the DOE approach that required clarification: (i) use of the term reasonableness; (ii) application of the DOE criterion on consistency with available data and scientific understanding—if the absence of validation information is used to reject an alternative conceptual model—this DOE approach and subsequent decisions must be documented and justified; (iii) documentation of the effects of alternative conceptual models and their uncertainties on the performance assessment, including presentation of disaggregated results of alternative conceptual models; (iv) determination of how weighting alternative conceptual models will avoid underestimating risk; (v) use of sensitive or key parameters, from previous analyses, when evaluating potential future alternative conceptual models; and (vi) conveyance of the guidance to the model developers that would ensure consistency in the development of model validation criteria and the systematic treatment of uncertainty throughout the performance assessment model.

The methodology outlined by DOE in CRWMS M&O (1999) for sampling parameter uncertainty seems reasonable. This use of Latin Hypercube Sampling permits parameters to be efficiently sampled across the ranges of uncertainty. This sampling would appear reasonable 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.

Overall, the available information, along with key technical issues agreements between DOE and NRC, is sufficient to expect that the information necessary to assess credibility of the DOE total system performance assessment results will be available at the time of a potential license application.

5.1.4.1.5 Summary and Status of Key Technical Issue Subissues and Agreements

Table 5.1.4.1-1 provides the status of DOE and NRC agreements pertaining to the analysis of repository performance to address the postclosure individual protection standard.

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. The NRC review to date does not constitute a compliance determination. Note the status and the detailed agreements pertaining to all the key technical issue subissues are provided in Table 1.2-1 and Appendix A.

Table 5.1.4.1-1. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreement*
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

Table 5.1.4.1-1. Related Key Technical Issue Subissues and Agreements (continued)			
Key Technical Issue	**Subissue**	**Status**	**Related Agreement***
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 review methods.			

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———. "Total System Performance Assessment and Integration Key Technical Issue Agreement 4.05." Letter (July 1) to J.D. Ziegler, DOE. Washington, DC: NRC. 2002b. <www.nrc.gov/reading-rm/adams.html/ ML021830120>

———. "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. <www.nrc.gov/waste/hlw-disposal/public-involvement/mtg-archive.html#KTI>

Ziegler, J.D. "Transmittal of Information to Address Key Technical Issue (KTI) TSPAI 4.05." Letter (May 7) to J.R. Schlueter, NRC. Washington, DC: DOE. 2002a. <www.nrc.gov/reading-rm/adams.html/ML021330453>

———. "Transmittal of Information to Address Key Technical Issue (KTI) Agreement Total System Performance Assessment and Integration (TSPAI) 4.07." Letter (December 23) to J.R. Schlueter, NRC. Las Vegas, Nevada: DOE. 2002b. <www.nrc.gov/ reading-rm/adams.html/ ML030020018>

5.1.4.2 Demonstration of Compliance with the Human Intrusion Standard

5.1.4.2.1 Description of Issue

This section documents review of the DOE approach for assessing the effects of human intrusion on the repository system as required by 10 CFR 63.321. The stylized human intrusion scenario is described in 10 CFR 63.322, as (i) a single ground water 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 human intrusion are documented in the total system performance assessment and model reports for the site recommendation (CRWMS M&O, 2000a,b) and additional supporting analysis and model reports (Bechtel SAIC Company, LLC, 2001a,b,c; DOE, 2002). This section reviews DOE analysis to assess whether DOE methodology and data are sufficient for conducting a detailed review. This is not a compliance review.

5.1.4.2.2 Relationship to Key Technical Issue Subissues

This section incorporates subject matter previously captured in the following two 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 has been made to explicitly identify each subissue.

5.1.4.2.3 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 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 ground water pathway (CRWMS M&O, 2000c). Note 10 CFR 63.312(c) specifies an annual water demand of about 3.7×10^6 m^3 [exactly 3,000 acre-feet] and, therefore, fixes the dilution rate of radionuclides.

5.1.4.2.4 Technical Basis

NRC developed a review plan (NRC, 2003) consistent with the acceptance criteria and review methods found in previous issue resolution status reports. A status assessment of the DOE approaches for including Demonstration of Compliance with the Human Intrusion Standard in total system performance assessment abstractions is provided in the following subsections. This assessment is organized according to the three review methods identified in Section 2.2.1.4.2.2 of the review plan (NRC, 2003): (i) Evaluation of the Time of Occurrence of an Intrusion Event, (ii) Evaluation of an Intrusion Event That Demonstrates That 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 Representation of the Intrusion Event. This review is limited to evaluating the adequacy of DOE approach and data. Additionally, the beginning of each of the following subsections contains a summary of risk insights considerations used to focus the assessments on those aspects most important to repository safety.

5.1.4.2.4.1 Evaluation of the Time of Occurrence of an Intrusion Event

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 the waste package would degrade sufficiently that human intrusion could occur without recognition by the drillers. The second step, which will be presented in more detail in Section 5.1.4.2.4.2, requires an assessment be performed if a waste package is projected to be penetrated at or before 10,000 years after disposal.

Staff found the method for estimating the time of earliest intrusion presented by DOE (CRWMS M&O, 2000a; DOE, 2002) was generally satisfactory. The DOE approach presented in the total system performance assessment report for the site recommendation (CRWMS M&O, 2000a) and supporting analyses (Bechtel SAIC Company, LLC, 2001a,b,c; DOE, 2002) assumed the human intrusion occurred 100 years after closure of the repository. DOE stated 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. More recently, DOE indicated its intention to demonstrate the earliest time human intrusion could occur without recognition by a driller is 30,000 years, the time at which DOE believes the waste packages will begin to fail as a result of corrosion (Bechtel SAIC Company, LLC, 2002). DOE reported elsewhere results from analyses that assume human intrusion occurs 30,000 years following closure (Bechtel SAIC Company, LLC, 2001a,b,c; DOE, 2002). If DOE uses this approach, it must provide, as required by 10 CFR 63.321, the analyses and technical bases used to justify the time of occurrence.

Overall, the available information is sufficient to expect that the information necessary to assess Demonstration of Compliance with the Human Intrusion Standard with respect to evaluation of the earliest time of an intrusion event will be available at the time of a potential license application.

5.1.4.2.4.2 Calculation of the Annual Dose to the Reasonably Maximally Exposed Individual from an Intrusion Event

Modeling this prescribed human-intrusion scenario using the TPA Version 4.1 code and assuming the intrusion occurs 100 years after closure of the repository gave peak total expected annual doses to the reasonably maximally exposed individual near 10^{-6} Sv [0.1 mrem] in 10,000 years (Mohanty, et al., 2002). The calculated dose remains low primarily because of the limited spent nuclear fuel inventory available in this scenario (i.e., one waste package, as defined by 10 CFR 63.322).

The methods presented by DOE (CRWMS M&O 2000a; DOE, 2002) for evaluating the annual dose to the reasonably maximally exposed individual in any year during 10,000 years resulting from human intrusion appear reasonable. 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, it should be demonstrated 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 10,000 years. DOE used its total system performance assessment code for this demonstration in submitted reports (CRWMS M&O 2000a; DOE, 2002). More recently, DOE indicated its intention to demonstrate the earliest time human intrusion could occur without recognition by a driller is 30,000 years, the time at which DOE believes the waste packages will begin to fail as a result of corrosion (Bechtel SAIC Company, LLC, 2002). DOE reported results from analyses that assume human intrusion occurs 30,000 years following closure (Bechtel SAIC Company, LLC, 2001a,b,c; DOE, 2002).

Overall, the available information is sufficient to expect that the information necessary to assess that the annual dose to the reasonably maximally exposed individual in any year during 10,000 years because of a human intrusion event will be available at the time of a potential license application.

5.1.4.2.4.3 The Total System Performance Assessment Code Representation of the Intrusion Event

Modeling this prescribed human-intrusion scenario using the TPA Version 4.1 code and assuming the intrusion occurs 100 years after closure of the repository gave peak total expected annual doses to the reasonably maximally exposed individual near 10^{-6} Sv [0.1 mrem] in 10,000 years (Mohanty, et al., 2002).

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 (Reamer, 2001) and follow:

- Volume and chemistry of drilling fluids are ignored in analysis.
- Rate of infiltration is unaffected by the presence of the borehole.
- Cladding in the penetrated waste package is perforated because of the event, but not completely failed.
- Properties of the rubblized borehole (porosity, fluid saturation, and dispersivity) are represented by the matrix properties of an unsaturated zone fault.

Since the technical exchange, DOE has provided more detailed descriptions of its intended approach for human intrusion calculations (Bechtel SAIC Company, LLC, 2001a,b, 2002; DOE, 2002). DOE has indicated that the approach will conform to 10 CFR Part 63 and the methods and results will be documented in the Total System Performance Assessment—License Application Technical Report and Total System Performance Assessment—License Application Model Report.

DOE should ensure human intrusion calculations are stable with respect to the number of realizations and timestepping used. This comment was raised in Reamer (2001). 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. DOE agreed the supporting 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. These reports will be available at the time of license application. 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 discretizations.

Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.4.2.5), is sufficient to expect that the information necessary to assess compliance with the human intrusion standard with respect to data uncertainty being characterized and propagated through model abstraction will be available at the time of a potential license application.

5.1.4.2.5 Summary and Status of Key Technical Issue Subissues and Agreements

Table 5.1.4.2-1 provides the status of all key technical issue subissues referenced in Section 5.1.4.2.2 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 review methods discussed in Section 5.1.4.2.4. Note the status and the detailed agreements pertaining to all the key technical issue subissues are provided in Table 1.2-1 and Appendix A.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (e.g., specified testing or analyses), indicates that information necessary to begin a technical review will likely be available at the time of a potential license application.

Table 5.1.4.2-1. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreement*
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 three of the human intrusions.

5.1.4.2.6 References

Bechtel SAIC Company, LLC. "Total System Performance Assessment—License Application Methods and Approach." TDR-WIS-PA-000006. Rev. 00. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2002.

———. "Contract No. DE-AC08-01RW12101—Total System Performance Assessment Analyses for Disposal of Commercial and DOE Waste Inventories at Yucca Mountain—Input to Final Environmental Impact Statement and Site Suitability Evaluation REV 00 ICN 02." Letter from Bechtel SAIC Company, LLC to J.R. Summerson (DOE/YMSCO). December 11, 2001a.

———. "Contract No. DE-AC08-01RW12101—Total System Performance Assessment Sensitivity Analyses for Final Nuclear Regulatory Commission Regulations, Rev 00 ICN 01." Letter from Bechtel SAIC Company, LLC to S.J. Brocoum (DOE/YMSCO). December 11, 2001b.

———. "FY01 Supplemental Science and Performance Analyses—Volume 1: Scientific Bases and Analyses." TDR-MGR-MD-000007. Rev. 00 ICN 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2001c.

CRWMS M&O. "Total System Performance Assessment for the Site Recommendation." TDR-WIS-PA-000001. Rev. 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000a.

———. "Total System Performance Assessment (TSPA) Model for Site Recommendation." MDL-WIS-PA-000002. Rev. 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. Rev. 04 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000c.

DOE. DOE/EIS–0250F, "Final 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." Washington, DC: Office of Civilian Radioactive Waste Management. 2002.

Mohanty, S., R. Codell, J.M. Menchaca, R. Janetzke, M. Smith, P. LaPlante, M. Rahimi, and A. Lozano. "System-Level Performance Assessment of the Proposed Repository at Yucca Mountain Using the TPA Version 4.1 Code." CNWRA 2002-05. Rev. 1. San Antonio, Texas: CNWRA. 2002.

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

———. "Issue Resolution Status Report, Key Technical Issue: Total System Performance Assessment and Integration." Rev. 3. Washington, DC: NRC. 2000.

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. <www.nrc.gov/waste/hlw-disposal/public-involvement/mtg-archive.html#KTI>

5.1.4.3 Analysis of Repository Performance That Demonstrates Compliance with Separate Ground Water Protection Standards

5.1.4.3.1 Description of Issue

This section about the analysis of repository performance that demonstrates compliance with separate ground water protection standards addresses the DOE approach for conducting a total system performance assessment of ground water contamination arising from releases of radionuclides from the Yucca Mountain disposal system. The separate ground water protection standards detailed in 10 CFR 63.331 state DOE must demonstrate there is a reasonable expectation that, for 10,000 years of undisturbed performance after disposal, releases of radionuclides from waste in the Yucca Mountain disposal system into the accessible environment will not cause the level of radioactivity in the representative volume of ground water to exceed the limits in Table 5.1.4.3-1.

Requirements for the determination of the representative volume are specified at 10 CFR 63.332. The representative volume is the volume of ground water that would be withdrawn annually from an aquifer to supply a water demand of 3,700,000 m^3 [3,000 acre-ft] per year. DOE must determine the position of the representative volume by assuming the volume includes the highest concentration level in the plume of contamination in the accessible environment. DOE must calculate the dimensions of the representative volume as either a well-capture zone or as a slice of the plume.

The DOE approach for demonstrating compliance with separate ground water protection standards is documented in Bechtel SAIC Company, LLC (2003, Section 8.1.5). This section reviews a previous ground water protection compliance assessment conducted by DOE in the site recommendation (CRWMS M&O, 2000), as well as the approach DOE will use in a potential license application as documented in Bechtel SAIC Company, LLC (2003).

5.1.4.3.2 Relationship to Key Technical Issue Subissues

There are no subissues specific to the demonstration of compliance with separate ground water protection standards. The proposed 10 CFR Part 63 regulations, on which the key technical issue meetings were based, did not include separate ground water protection standards distinct from an all-pathways dose standard that included the ground water pathway. These standards were issued by EPA in its promulgation of 40 CFR Part 197 on June 13, 2001. Aspects of the

Table 5.1.4.3-1. Limits on Radionuclides in the Representative Volume

Radionuclide of Type of Radiation Emitted	Limit	Is Natural Background Included?
Combined Ra-226 and Ra-228	0.185 Bq/L [5 pCi/L or 18.9 pCi/gal]	Yes
Gross alpha (including Ra-226 but excluding radon and uranium)	0.555 Bq/L [15 pCi/L or 56.8 pCi/gal]	Yes
Combined beta- and photon-emitting radionuclides	0.04 mSv [4 mrem] per year to the whole body or any organ, based on drinking 2 L [0.53 gal] per day from the representative volume	No

analysis of repository performance that demonstrate compliance with separate ground water protection standards, however, have long been evaluated as necessary to the computation of concentrations of radionuclides in ground water required as part of the demonstration of compliance with the individual protection standard. Computation of concentrations of radionuclides in ground water is addressed in Section 5.1.3.12 of this report and incorporates subject matter previously captured in the following six key technical issue subissues:

- Unsaturated and Saturated Flow under Isothermal Conditions: Subissue 5—Saturated Zone Flow and Dilution Processes (NRC, 2000a)
- Radionuclide Transport: Subissue 3—Radionuclide Transport through Alluvium (NRC, 2000b)
- 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)

5.1.4.3.3 *Importance to Postclosure Performance*

Because the individual, all-pathway dose limit of 0.15 mSv [15 mrem] required as part of the individual protection standard includes a ground water pathway, radionuclide concentrations in ground water must be computed in performance assessments for demonstrating compliance with all relevant postclosure performance objectives. In CRWMS M&O (2000, Section 4.1.5), DOE includes demonstration of compliance with the separate ground water protection standards as an auxiliary output from the nominal case. A similar approach is planned for the total system performance assessment for the potential license application, as documented in Bechtel SAIC Company, LLC (2003). The demonstration of compliance with the individual protection standard at 10 CFR 63.311 contains most of the analysis required to demonstrate compliance with the separate ground water protection standards at 10 CFR 63.331 and is evaluated in this report by discussions of Flow Paths in the Saturated Zone (Section 5.1.3.8), Radionuclide Transport in the Saturated Zone (Section 5.1.3.9), and concentration of radionuclides in ground water (Section 5.1.3.12). This section will focus on those aspects of the compliance demonstration unique to the separate ground water protection standards.

5.1.4.3.4 Technical Basis

5.1.4.3.4.1 Demonstration That the Ground Water Radioactivity and Drinking Water Doses Do Not Exceed the Separate Ground Water Protection Standard

Because demonstrating compliance with the individual protection standard also requires consideration of ground water pathways, there is a major question of consistency within the potential license application. In general, the demonstration of compliance with the separate ground water protection standards should be consistent with other analyses involving the use of ground water with due consideration to regulatory differences between the two standards 10 CFR 63.311 and 63.331. The technical basis for evaluating transport of radionuclides in ground water is discussed in this report in the sections on Flow Paths in the Saturated Zone (Section 5.1.3.8), Radionuclide Transport in the Saturated Zone (Section 5.1.3.9), and Concentration of Radionuclides in Ground Water (Section 5.1.3.12). Several requirements are specific to demonstration of compliance with the separate ground water protection standards.

1. DOE must account for natural background levels of Ra-226, Ra-228, and gross alpha contamination (excluding radon and uranium).

2. DOE must compute a whole body dose and an organ dose for beta- and photon-emitting radionuclides arising from the consumption of 2 L [0.53 gal] of water per day.

3. Unlikely features, events, and processes or unlikely sequences of features, events, and processes are to be excluded from demonstration of compliance with the ground water protection standards.

The DOE site recommendation (CRWMS M&O, 2000) provides the only example to date of a detailed assessment of compliance with the then proposed ground water protection standard at 40 CFR 197.35 (EPA, 1999). Because the approach to demonstration of compliance with the separate ground water protection standards relies on auxiliary analyses, using intermediate results obtained from the analysis of the nominal scenario, most aspects of demonstration of compliance with the separate ground water compliance standards are consistent with the nominal scenario class. Transport of radium (Ra-226 and Ra-228), however, is not modeled explicitly in the nominal scenario class. Instead, DOE assumes these radionuclides are in secular equilibrium with their parent nuclides (Th-230 and Th-232). The relative conservatism of this approach is not clear. Although this approach is conservative in that the time required to establish secular equilibrium can be a substantial portion of the compliance period, it is not conservative if radium is more mobile than thorium or if radium is preferentially released. Ingrown, mobile radium could, therefore, migrate farther than the potentially less mobile thorium parent nuclide. Radium is likely to be significantly retarded, however, during transport from the repository to the accessible environment. Retardation of radium would mitigate concern about a potential lack of conservatism associated with assuming radium transport is analogous to thorium transport.

Measurements of the existing background activity of Ra-226, Ra-228, and other radionuclides that contribute to gross alpha activity (excluding radon and uranium) have been obtained by DOE and included in its assessment of compliance with activity concentration limits (CRWMS M&O, 2000, 1999). CRWMS M&O (2000) bases the determination of background on a measurement from a single well installed by the Nevada Department of Transportation near the intersection of U.S. Route 95 and Nevada State Route 373. Although reference is made to other wells, no documentation in CRWMS M&O is provided to demonstrate this well is representative of the region. In Bechtel SAIC Company, LLC (2003), DOE states the variability in the natural background levels of Ra-226, Ra-228, and gross alpha is likely to be greater than the magnitude of the contribution from the repository. The planned approach would, therefore, include only the mean value for natural background and would add the stochastically variable repository contribution to the mean value of background to allow interpretation of repository performance. It appears DOE has sufficient information to determine appropriate background levels for Ra-226, Ra-228, and gross alpha in ground water.

As part of its analysis, DOE determines the key beta- and photon-emitting radionuclides and corresponding critical organs are C-14 for fat, Tc-99 for the gastrointestinal tract, and I-129 for the thyroid. Based on a drinking water consumption rate of 2 L [0.53 gal] per day, DOE determines the 0.04-mSv [4-mrem] dose limit is reached at activity concentrations of 2,000, 900, and 1 pCi/L for C-14, Tc-99, and I-129. The justification for including only these beta- and photon-emitting radionuclides for demonstrating compliance with the separate ground water protection standards is not provided in CRWMS M&O (2000). These nuclides, however, are likely to contribute the majority of the beta-photon dose to a receptor in the accessible environment within the 10,000-year compliance period based on their relatively high mobility in the environment.

Overall, the available information is sufficient to expect that the information necessary to assess the DOE demonstration that ground water radioactivity and doses will not exceed the separate ground water protection standard will be available at the time of a potential license application.

5.1.4.3.4.2 Methods and Assumptions Used to Determine the Location of the Representative Volume of Ground Water

The focus for the location of the representative volume is ensuring the highest contamination level in the plume is captured within the representative volume. Because demonstrating compliance with the individual protection standard also requires consideration of ground water pathways, a major question is one of consistency within the potential license application. In addition to general considerations of consistency within the potential license application, the prescriptive nature of these requirements entails several specific requirements related to the location of the representative volume of ground water.

1. The representative volume determined by DOE must include the highest concentration level in the plume of contamination in the accessible environment.

2. The aquifer within which the representative volume is located must contain less than 10,000 mg [0.22 lb] of total dissolved solids per liter.

The approach in CRWMS M&O (2000) to determine water concentrations for comparison with compliance limits was to assume any radionuclide reaching the accessible environment boundary would be captured and mixed within an annual well withdrawal of 1,591,000 m^3 [1,285 acre-ft] per year. A similar approach is planned for the potential license application (Bechtel SAIC Company, LLC, 2003), with the difference that an annual well water withdrawal of 3,700,000 m^3 [3,000 acre-ft] will be used to be consistent with the final 10 CFR 63.332. This approach requires the assumption that no changes in aquifer chemistry could result in the sudden release of sorbed radioactivity within the plume of contamination.

CRWMS M&O (2000) contains an evaluation of the total dissolved solids concentration of the ground water pumped at the receptor location. Water at the accessible environment boundary was reported to contain 385 mg/L [3.21×10^{-3} lb/gal], well below the 10,000-mg/L [8.32×10^{-2}-lb/gal] limit on usable water. DOE is, therefore, not excluding any part of the plume based on nonpotability.

CRWMS M&O (2000) provides the only example to date of a detailed assessment of compliance with the then proposed ground water protection standard at 40 CFR 197.35 (EPA, 1999). That document used auxiliary outputs from the nominal scenario to demonstrate compliance with ground water protection standards. This approach ensures a high degree of consistency between the different postclosure performance objectives requiring estimation of ground water concentrations.

Overall, the available information is sufficient to expect that the information necessary to assess the DOE calculation of the position of the representative volume of ground water will be available at the time of a potential license application.

5.1.4.3.4.3 Methods and Assumptions Used to Determine the Dimension of the Representative Volume of Ground Water

The focus for the dimensions of the representative volume is ensuring the highest contamination level in the plume is captured within the representative volume. The representative volume contains 3,700,000 m^3 [3,000 acre-ft]. Because demonstrating compliance with the individual protection standard also requires consideration of ground water pathways, there is a major question of consistency within the potential license application. In addition to general considerations of consistency within the potential license application, the prescriptive nature of these requirements entails several specific requirements related to the dimensions of the representative volume of ground water.

1. The representative volume determined by DOE must include the highest concentration level in the plume of contamination in the accessible environment.

2. DOE must provide the dimensions of the representative volume as either a well capture zone or a slice of the plume of contamination. Dimensions of the representative volume must be based on average characteristics of the aquifers along the radionuclide migration path using cautious, but reasonable values as determined by site characterization.

3. The representative volume must contain 3,700,000 m^3 [3,000 acre-ft].

In CRWMS M&O (2000), DOE does not provide the dimensions of the representative volume as either a well capture zone or a slice of the plume of contamination. The information reviewed in this report regarding flow paths in the saturated zone and radionuclide transport in the saturated zone contains information relevant to determining the dimensions of either a well capture zone or a plume of contamination. Consistent with the approach documented in Section 5.1.3.12 of this report, however, DOE assumes in CRWMS M&O (2000) that any radionuclide reaching the accessible environment boundary would be captured and mixed within an annual well withdrawal of 1,585,000 m^3 [1,285 acre-ft] per year. That analysis states exact dimensions of the representative volume may change in each Monte Carlo realization with this method; therefore, dimensions are not provided. Although CRWMS M&O does not provide any technical justification to demonstrate this approach is consistent with wellwater concentrations computed using either a well capture zone or a slice of the plume, this approach appears consistent with the assumption of a large well capture zone of sufficient size to capture all the activity released into the accessible environment. Given this volume is held constant at 3,700,000 m^3 [3,000 acre-ft] per year, this approach would not result in inappropriate dilution of the material associated with unrealistically large well capture zones. This approach requires the assumption that no changes in aquifer chemistry could result in the sudden release of sorbed radioactivity within the plume of contamination (cf. Key Technical Issue Agreement TSPAI.3.31 discussed in Section 5.1.3.9, Radionuclide Transport in the Saturated Zone). There is no evidence to suggest significant changes to the water chemistry of the source aquifer during the 10,000-year period. Such an approach is intended to eliminate the need to defend any particular set of parameters used to compute a well capture zone.

In the site recommendation (CRWMS M&O, 2000), DOE assumes an annual well withdrawal of 1,585,000 m^3 [1285 acre-ft] per year, consistent with the then-proposed EPA standard. The final EPA rule at 40 CFR Part 197 requires a representative volume of 3,700,000 m^3 [3,000 acre-ft] per year; this approach is documented in Bechtel SAIC Company, LLC (2003).

CRWMS M&O (2000) provides the only example to date of a detailed assessment of compliance with the then proposed ground water protection standard at 40 CFR 197.35 (EPA, 1999). That document used auxiliary outputs from the nominal scenario to demonstrate compliance with ground water protection standards. This approach ensures a high degree of consistency between the different postclosure performance objectives requiring estimation of ground water concentrations.

Overall, the available information is sufficient to expect that the information necessary to assess the DOE demonstration that methods and assumptions used in calculating the physical dimensions of the representative volume of ground water will be available at the time of a potential license application.

5.1.4.3.5 Summary and Status of Key Technical Issue Subissues and Agreements

The resolution status of this integrated subissue is based on the resolution status of each of the contributing key technical subissues. The relevant subissues are tracked elsewhere in this document, primarily in the sections on Flow Paths in the Saturated Zone (Section 5.1.3.8), Radionuclide Transport in the Saturated Zone (Section 5.1.3.9), and Concentration of *Radionuclides in Ground Water (Section 5.1.3.12).*

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (e.g., specified testing or analyses), indicates that information necessary to begin a technical review will likely be available at the time of a potential license application. As noted in this section of the report, DOE should provide additional documentation of some aspects of the approach.

5.1.4.3.6 References

Bechtel SAIC Company, LLC. "Total System Performance Assessment—License Application Methods and Approach." TDR–WIS–PA–000006. Rev. 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2003.

CRWMS M&O. "Total System Performance Assessment for the Site Recommendation." TDR–WIS–PA–000001. Rev. 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000.

———. "Radioactivity in FY 1998 Groundwater Samples from Wells and Springs Near Yucca Mountain." BA0000000–01717–5705–00029. Rev. 00. Las Vegas, Nevada: CRWMS M&O. 1999.

EPA. "Environmental Radiation Protection Standards for Yucca Mountain, Nevada." Chapter 1—Environmental Protection Agency. 40 CFR Part 197. *Federal Register.* Vol. 64, No. 166. pp. 46976–47016. August 27, 1999.

———. "Issue Resolution Status Report, Key Technical Issue: Unsaturated and Saturated Flow Under Isothermal Conditions." Rev. 2. Washington, DC: NRC. 2000a.

———. "Issue Resolution Status Report, Key Technical Issue: Radionuclide Transport." Rev. 2. Washington, DC: NRC. 2000b.

———. "Issue Resolution Status Report, Key Technical Issue: Total System Performance Assessment and Integration." Rev. 3. Washington, DC: NRC. 2000c.

6 PERFORMANCE CONFIRMATION

6.1 Research and Development Program to Resolve Safety Questions

6.1.1 Description of Issue

Requirements for the content of the license application at 10 CFR 63.21(c)(16) specify that the U.S. Department of Energy (DOE) identifies those structures, systems, and components of the geologic repository, both surface and subsurface, that require research and development to confirm adequacy of design. These requirements also specify 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 is expected to 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 when 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 potential license application.

The U.S. Nuclear Regulatory Commission (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 (2003). 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 could 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.

6.1.2 Relationship to Key Technical Issue Subissues

Specific topics for the research and development programs to resolve safety questions will be identified when DOE has completed its safety analyses to support the license application for construction authorization. Safety questions may be related to existing integrated subissues that may not be resolved adequately at the time of license application. Also, it is possible that other safety questions will emerge before submission of a potential license application.

6.1.3 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 adequacy of any proposed research and development program. The integrated safety significance of safety questions will be considered when the NRC determines whether to approve a construction authorization.

6.1.4 Technical Basis

Because safety questions and their associated research and development programs have not yet been presented in a potential license application, there is no technical basis to evaluate. The approach for the review of any such concerns and programs is provided in NUREG–1804 (NRC, 2003). The insights provided in the most recent version of the NRC Risk Insights Baseline Report (Appendix D) will be used to assist staff review of safety questions and their associated research and development programs.

6.1.5 Summary

No safety questions have yet been identified. Consequently, the associated research and development programs have not been developed.

If a license application is submitted, the NRC staff will evaluate the research and development programs for any safety questions using the approach in NUREG–1804 (NRC, 2003).

6.1.6 Reference

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

6.2 Performance Confirmation Program

6.2.1 Description of Issue

Performance confirmation is the program of tests, experiments, and analyses to evaluate adequacy of the information used to determine that 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 potential repository and subsequent emplacement of the wastes. Key geologic, hydrologic, geomechanical, geochemical, 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.

6.2.2 Importance to Safety and Postclosure Performance

The DOE Performance Confirmation Program plan is intended to address the range of postclosure performance topics and their associated uncertainties. 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 to discover potential negative effects on the safety of the potential repository. Planning for—and later, conducting—an effective Performance Confirmation Program is, therefore, an important part of the DOE compliance with the performance objectives.

6.2.3 Technical Basis

Because the Performance Confirmation Program plan has not yet been presented in sufficient detail by DOE, currently there is no technical basis to evaluate the plan. The approach for reviewing the Performance Confirmation Program is provided in NRC (2003). The insights provided in the most recent version of the NRC Risk Insights Baseline Report (Appendix D) will be used to assist staff review of the Performance Confirmation Program.

6.2.4 Summary

The DOE activities conducted to date, as part of site characterization, have begun to establish baseline information against which future repository performance can be evaluated (Barr, 2003). DOE anticipates the transition from developing a baseline to monitoring and modeling the performance effects of changes from baseline conditions will occur before emplacement of waste in the potential repository. According to Civilian Radioactive Waste Management System Management and Operating Contractor (CRWMS M&O) (2000), DOE plans include the following activities.

<u>Identify Performance Confirmation Factors and Parameters</u>. Identify the factors (processes) and related parameters important to postclosure safety that should be monitored as part of performance confirmation.

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.

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.

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.

Plan and Set Up the Performance Confirmation Test and Monitoring Program. Conduct detailed planning, construct the testing and monitoring facilities, and set up instrumentation necessary for the Performance Confirmation Program, including establishment of the ambient baseline, if necessary.

Monitor, Test, and Collect Data. Perform the testing and monitoring activities necessary to collect data in accordance with applicable regulations and quality assurance requirements.

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.

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.

Currently, CRWMS M&O (2000) is undergoing a major revision and is scheduled to be released in the later half of fiscal year 2004. DOE provided an overview of the methodology adopted for screening potential performance confirmation activities (Barr, 2003). The overview summarized 71 key activities that DOE plans to address in the revised Performance Confirmation Program plan. The staff will review, consistent with the guidance in NRC (2003) and insights provided in the most recent version of the NRC Risk Insights Baseline Report (Appendix D), the revised DOE Performance Confirmation Program plan.

6.2.5 References

Barr, D. "Performance Confirmation." Presentation to the Nuclear Waste Technical Review Board, Fall 2003 Board Meeting, September 17, 2003. Amargosa Valley, Nevada. 2003. <www.nwtrb.gov/meetings/030917.doc>

CRWMS M&O. "Performance Confirmation Plan." TDR–PCS–SE–000001. Rev. 01 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000.

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

7 ADMINISTRATIVE AND PROGRAMMATIC REQUIREMENTS

7.1 Quality Assurance Program

The status of the U.S. Department of Energy (DOE) quality assurance program assessment and the program implementation is discussed in the following sections.

7.1.1 Background

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the contents of the DOE quality assurance program and each subsequent change to that program. The program appears sufficient for use in developing a potential license application, for the nature of the work DOE has performed to date.

DOE issued Corrective Action Report BSC–01–C–001 on May 3, 2001, which documents systematic examples of inadequate model validation in 18 of 24 model validations examined during a DOE audit. Corrective Action Report BSC–01–C–002, issued June 12, 2001, documents failures of implementation of the quality assurance program related to software. On May 17, 2001 (Reamer, 2001), a letter issued to DOE stated the NRC staff identified technical errors and inconsistencies between the Total System Performance Assessment for Site Recommendation documents, the underlying analysis and model reports, the associated GoldSim (registered trademark by Golder Associates Inc.) computer code results (GoldSim Technology Group, LLC, 2004), and the associated hand calculations. These problems the DOE and NRC staffs identified are repetitive and indicate previous corrective actions were not effective.

To address the problems identified in Corrective Action Reports BSC–01–C–001 and BSC–01–C–002 and other recurring problems, DOE issued its Management Improvement Initiatives (DOE, 2002) on July 19, 2002. The Management Improvement Initiatives charts the DOE path forward for overall improvements. One objective of the Management Improvement Initiatives is to ensure timely and effective corrective actions are implemented so problems are promptly and effectively resolved.

Subsequently, DOE issued Corrective Action Report BSC(B)–03–107 on March 28, 2003, because of recurring conditions and ineffective corrective actions regarding data issues.

By a letter to NRC dated April 5, 2004 (Chu, 2004), DOE determined all the Management Improvement Initiatives objectives had been accomplished, and DOE transitioned the Management Improvement Initiatives to routine management practices.

7.1.2 Assessment of DOE Approach

The DOE quality assurance program focuses on analytical work associated with the site recommendation and the potential license application. The program has changed with time to address performance issues and management objectives within the project.

The NRC staff has observed certain performance-based audits conducted by DOE. 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 problems continue with procedure compliance, and, some technical reports contain insufficient detail to document the bases for certain assumptions, inputs, and equations.

The NRC staff was concerned that recurring problems in the areas of models, software, and data will have an impact on the NRC staff ability to effectively complete their evaluation of the potential license application within the time required by law. As a result, the NRC staff conducted three audits between November 2003 and January 2004 to independently evaluate the technical information in selected technical reports and supporting information considered significant to repository performance. The technical information included field and experimental data, models, analyses, and justifications for any assumptions and conclusions presented by DOE. The NRC staff used its risk insights baseline to select the analysis and model reports believed to be of high or medium significance to repository performance. The NRC staff also evaluated the processes used in developing analysis and model reports and the effectiveness of corrective actions in eliminating recurring problems in the areas of models, software, and data.

The NRC staff identified concerns with all three technical reports audited, indicating other technical reports supporting the potential license application may be similarly affected. The NRC staff identified some concerns with clarity regarding the DOE technical bases presented in the technical reports evaluated and also with the presentation of sufficient technical information to support those explanations. These concerns are summarized next.

1. In some cases, DOE did not explain its technical basis such that the NRC staff could understand how the DOE conclusions were reached. DOE may have provided sufficient technical information but, because the DOE explanation was not satisfactory, the NRC staff could not determine the sufficiency or adequacy of the technical information.

2. DOE adequately explained its technical basis but did not provide the technical information necessary to support that explanation. Technical information includes experimental data, analog information, analyses, and expert judgment.

The NRC staff concluded, if a license application is submitted, deficiencies in the technical reports may lead to large numbers of requests for additional information and may impact the ability of the NRC staff to complete the license application review within the mandated period.

7.1.3 Implementation of Corrective Action

In September 2003, DOE implemented its revised corrective action program. The revision includes using a single system for all levels of conditions (i.e., conditions adverse to quality and to lower-level deficiencies) and enhanced trend analysis. Early results from the trend analyses indicate human performance is a major contributor to problems. DOE initiated a human performance improvement initiative to address this trend, however, realized benefits of the initiative are likely to be long range.

In addition to revising the corrective action and trending programs resulting from the Management Improvement Initiatives, DOE established a comprehensive system of performance indicators that includes indicators for corrective action and quality program effectiveness. These performance indicators yield valuable information for the DOE management decisions. DOE monitors performance measures and reports quarterly results to NRC.

At the time of this Integrated Issue Resolution Status Report, DOE had completed and verified actions for Corrective Action Reports BSC–01–C–002 and BSC(B)–03–107, while actions for Corrective Action Report BSC–01–C–001 were in progress.

In response to the NRC staff conclusions identified as a result of their evaluation of technical information in the DOE technical reports described previously, DOE convened reviews of 100 percent of the technical reports supporting the potential license application to identify and correct deficiencies. The reviews, to be completed by August 2004, will employ subject matter experts independent from the technical report development.

7.1.4 Summary

The DOE quality assurance program content is sufficient for use in developing a potential license application, for the type of activities performed by DOE to date. However, the program has not been reviewed with respect to the full range of activities to be conducted at the potential repository, as will be required following the receipt of a potential DOE license application.

With respect to program implementation, DOE has identified recurring deficiencies and experienced a history of ineffective corrective actions in models, software, and data. The Management Improvement Initiatives and associated improvements to the correction action program was instituted to improve program implementation and result in higher quality products. In addition, the 100-percent review of technical reports DOE is conducting was implemented to correct deficiencies and enhance the quality of the potential license application.

The NRC staff may continue to observe the DOE audits and discuss quality assurance program problems and corrective actions with DOE. If determined necessary, the NRC staff may conduct additional audits or other activities to determine if remedial actions are effective in correcting technical product deficiencies. Also, the NRC onsite representatives will continue to routinely interact with DOE and its management and operating contractor about progress of corrective measures.

7.1.5 References

Chu, M. "Management Improvement Initiatives." Letter (April 5) to M. Virgilio, NRC. Washington, DC: DOE, Office of Civilian Radioactive Waste Management. 2004. <www.nrc.gov/reading-rm/ adams.html>

DOE. "DOE (OCRWM) Management Improvement Initiatives." Rev. 00. Las Vegas, Nevada: DOE. 2002.

GoldSim Technology Group "Graphical Simulation Environment User's Guide." Issaquah, Washington: GoldSim Technology Group, LLC. 2004.

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. <www.nrc.gov/reading-rm/ adams.html>

7.2 Records, Reports, Tests, and Inspections

The requirements of Subpart D of 10 CFR Part 63 in this area have not been the subject of DOE and NRC prelicensing discussions and no issues have been identified.

7.3 Training and Certification of Personnel

7.3.1 DOE Organization Structure As It Pertains to Construction and Operation of Geologic Repository Operations Area

The requirements of subpart H of 10 CFR Part 63 in this area have not been the subject of DOE and NRC prelicensing discussions and no issues have been identified.

7.3.2 Key Positions Assigned Responsibility for Safety and Operations of Geologic Repository Operations Area

The requirements of subpart H of 10 CFR Part 63 in this area have not been the subject of DOE and NRC prelicensing discussions and no issues have been identified.

7.3.3 Personnel Qualifications and Training Requirements

The requirements of subpart H of 10 CFR Part 63 in this area have not been the subject of DOE and NRC prelicensing discussions and no issues have been identified.

7.4 Expert Elicitation

7.4.1 Description of Issue

Nearly every aspect of site characterization, repository design, preclosure safety analysis, and postclosure performance assessment includes uncertainties. The primary method to evaluate and, to the extent practical, reduce these uncertainties is through collection of sufficient data and information during site characterization and design. Uncertainties will remain, however, in site characterization and safety and performance assessments because of factors such as temporal and spatial variations in 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. Consequently, the NRC staff anticipates it will be necessary to complement and supplement data obtained during site characterization and design 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 accepts the use of expert elicitation in evaluating and interpreting the factual bases of license applications. Thus, the NRC staff will give appropriate consideration to the judgments of DOE experts on technical aspects related to characterization and design of a potential geologic repository at Yucca Mountain. The expectation is that DOE use of expert elicitations will complement and supplement more objective sources of scientific and technical information, such as field investigations, analyses, and experimentation (NRC, 1996). Formal elicitation procedures, used prudently and appropriately, will 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 properly, formal elicitation reveals a 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 also may help groups of experts to resolve differences in their estimates by providing a common scale of measurement and a common vocabulary for expressing their judgments.

7.4.2 Relationship to Key Technical Issues

The staff evaluation of the DOE use and application of expert elicitation in developing parameters and parameter uncertainty important to preclosure safety and postclosure performance assessment addresses four key technical issues are: (i) Igneous Activity, (ii) Structural Deformation and Seismicity, (iii) Unsaturated and Saturated Flow Under Isothermal Conditions, and (iv) Radionuclide Transport. Information contained in this section incorporates subject matter previously captured in the following key technical issue subissues:

- Igneous Activity: Subissue 1—Probability (NRC, 1999a)
- Structural Deformation and Seismicity: Subissue 1—Faulting (NRC, 1999b)
- Structural Deformation and Seismicity: Subissue 2—Seismicity (NRC, 1999b)
- Radionuclide Transport: Subissue 1—Radionuclide Transport Through Porous Rock (NRC, 2000)

- Radionuclide Transport: Subissue 2—Radionuclide Transport Through Fractured Rock (NRC, 2000)
- Unsaturated and Saturated Flow Under Isothermal Conditions: Subissue 5—Saturated Zone Flow Conditions and Dilution Processes (NRC, 1999c)

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 would provide to NRC. The resolution status of expert elicitation 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. In parallel with NUREG–1804 (NRC, 2003) this section addresses two review methods: whether DOE used NUREG–1563 (NRC, 1996) or equivalent procedures and whether any updates to the DOE expert elicitation process were adequately documented and were based on appropriate methods.

7.4.3 Importance to Preclosure Safety and Postclosure Performance

The DOE use of expert elicitation and expert judgment is important to both preclosure safety and postclosure performance assessment calculations. DOE relied on expert elicitation to derive (i) the probability of igneous disruption of the potential repository; (ii) levels of vibratory ground motions from earthquakes used as inputs to preclosure seismic design and safety analysis as well as postclosure performance assessment of drift, waste package, and drip shield stability; (iii) ground water-specific discharge; and (iv) sorption coefficient distributions. Details of the risk significance of these parameters and parameter uncertainty derived from the DOE expert elicitation process are discussed in greater detail in the following sections of this report:

- 4.1.1 — Site Description As It Pertains to Preclosure Safety Analysis
- 5.1.2.2 — Identification of Events with Probabilities Greater than 10^{-8} per Year
- 5.1.3.2 — Mechanical Disruption of Engineered Barriers
- 5.1.3.7 — Radionuclide Transport in the Unsaturated Zone
- 5.1.3.8 — Flow Paths in the Saturated Zone
- 5.1.3.9 — Radionuclide Transport in the Saturated Zone
- 5.1.3.10 — Volcanic Disruption of Waste Packages

7.4.4 Staff Evaluation of DOE Use of Expert Elicitation

7.4.4.1 Probabilistic Volcanic Hazards Analysis

Large-magnitude silicic volcanic eruptions have not occurred in southern Nevada in the last 10 million years. There is evidence, however, of lesser-magnitude basaltic igneous 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 potential repository site—occurring approximately 80,000 years ago. Because of the potentially undesirable consequences of an igneous event, volcanism has been intensely investigated and debated for the last two decades. Uncertainties associated with igneous activity include

- Number, location, and age of past activity
- Physical characteristics of past eruptions
- Structural control of past or future volcanic activity
- Adequacy of probabilistic models for future volcanic activity
- Sufficiency of existing data for reliable probabilistic estimates of 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 range of probability values. In an attempt to address the areas of controversy as well as establish a basis for probabilistic calculations that could assess the potential effects of volcanism on repository performance, DOE assembled 10 experts and conducted an expert elicitation in 1995. The elicitation process consisted of four workshops and two field trips to the Yucca Mountain site. The resulting elicitation, documented in Civilian Radioactive Waste Management System Management and Operating Contractor (CRWMS M&O) (1996), evaluated a range of probability models, estimated uncertainties in model results caused by variations in model parameters, and determined a probability distribution for use in the DOE performance assessment models for Yucca Mountain. The NRC and the Center for Nuclear Waste Regulatory Analyses (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 elicitations and generally conducted the elicitations in accordance with the guidance set forth in NUREG–1563 (NRC, 1996).

Nevertheless, as explained in Section 5.1.2.2 of this document, the staff performed a review of CRWMS M&O (1996) and had technical concerns regarding these results and their application in the DOE analyses for performance assessment calculations.

As a result of these staff concerns, NRC reached two agreements with DOE (Schlueter, 2000a). In the first, DOE agreed to include, in a potential license application, for information purposes, the results of a single-point sensitivity analysis for extrusive and intrusive igneous activity at a probability of 10^{-7}/yr. Use of this single-point value will provide staff with the information necessary to review the effects of the DOE probability values, and of alternative conceptual models, on the risk estimate. This analysis has been previously presented in Bechtel SAIC Company LLC (2001a, Figure 4.3-1; 2001b). In addition, a new aeromagnetic survey was conducted for the Yucca Mountain region (Blakely, et al., 2000). In some locations, aeromagnetic surveys can locate igneous features that have been buried by sediments. At the August 2000 Igneous Activity Technical Exchange (Schlueter, 2000a), DOE also agreed to examine the results of this new survey for the presence of previously unrecognized buried igneous features and to evaluate the effects of these possible features on the CRWMS M&O (1996) probability estimates.

7.4.4.2 Probabilistic Seismic Hazards Analysis

DOE developed comprehensive probabilistic seismic and faulting hazard assessments to characterize the potential seismic and faulting hazards at Yucca Mountain

(CRWMS M&O, 1998). 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 evaluated the seismic source characterization. The other panel developed 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. Details of the technical aspects of the probabilistic seismic hazard assessment are provided in Section 4.1.1.3.5, Site Geology and Seismology, of this report.

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.

7.4.4.2.1 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, 1998). 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 in the seismic source zone characterization. Technical details of aspects of the seismic and fault displacement hazard results are provided in Section 5.1.2.2.4.2, Faulting, and Section 5.1.2.2.4.3, Seismicity, of this report.

Staff reviewed the information developed by DOE through the documentation process on fault displacement and seismic source zone characterization (CRWMS M&O, 1998) and found it sufficient to use in a potential license application for Yucca Mountain. DOE adequately justified the need for the elicitation and conducted the elicitation in accordance with the guidance set forth in NUREG–1563 (NRC, 1996). Although geological, geophysical, and seismotectonic studies continue in the Yucca Mountain region after the seismic source experts completed their assessments in 1998, no new information or analyses have surfaced that would require the experts to reconsider their results. Rather, geological, geophysical, and seismic data gathered to date remain consistent with the information and results presented in the seismic source portion of the DOE probabilistic seismic hazards assessment (CRWMS M&O, 1998).

7.4.4.2.2 Ground-Motion Attenuation

DOE assembled seven experts for the ground-motion elicitation, which was conducted in parallel with the seismic source zone elicitation. The ground-motion experts were asked 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 ground-motion relationships.

Staff reviewed the information about ground-motion attenuation developed by DOE through the documentation process (CRWMS M&O, 1998) and found it insufficient to use in a potential license application for Yucca Mountain (subject to the agreement described in Section 7.4.6, Conclusions). 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. Specifically, DOE did not provide sufficient documentation demonstrating the ground-motion experts clearly understood the implications of their ground-motion parameter inputs (part of postelicitation feedback), which are necessary for the ground-motion model development process. This postelicitation feedback is necessary to verify the technical integrity of the elicitation process as well as the traceability of the assessment. Consequently, the absence of postelicitation feedback documentation diminishes the acceptability and credibility of the elicitation results because the process 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 completeness of the elicitation feedback process. In particular, examination of several of the ground-motion models illustrated a large range of unexplained differences exists between the experts inputs regarding predicted ground-motions and the epistemic and aleatory uncertainties. In some instances, staff noted wide differences between experts and a 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 the experts will support the probabilistic seismic hazard assessment results. In the ground-motion elicitation, the experts provided intermediate results 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 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.

Staff independently examined the basis for the elicited ground-motion attenuation models and results and identified several questions about the DOE postelicitation feedback and documentation process (CRWMS M&O, 1998). At the October 2000 Technical Exchange on Structural Deformation and Seismicity (Schlueter, 2000b), 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 staff questions after this presentation, DOE agreed (Schlueter, 2000b) to provide additional documentation describing the process used to elicit the ground-motion attenuation models (Structural Deformation and Seismicity Agreement 2.01). In a letter dated December 21, 2000, DOE provided information it believed was responsive to the agreement made with the staff in October 2000 (Schlueter, 2000b). After a review of this new submittal, staff concluded most information provided was already available, and it did not materially contribute to the closure of this issue.

On August 27 and 28, 2002, a meeting concerning the Structural Deformation and Seismicity Agreement 2.01 was held between representatives of DOE and NRC at the NRC On-site Representatives Office in Las Vegas, Nevada. This meeting was to clarify the issue and to verify whether documentation of the ground-motion assessments and related expert elicitation for the Yucca Mountain Probabilistic Seismic Hazard Assessment were adequate and consistent with the guidance in NUREG–1563 (NRC, 1996). Additional documentation was provided to NRC by DOE as a result of discussions during that meeting. However, DOE has not yet provided all requested documents nor given NRC a complete discussion how the documentation satisfies the NRC concerns in the agreement. DOE plans to complete the agreement response in Technical Basis Document 14: Low Probability Seismic Events, which is scheduled to be available to NRC in June 2004 (Ziegler, 2004). Overall, the available information, along with the key technical issue agreement between DOE and NRC (Structural Deformation and Seismicity Agreement 2.01), is sufficient to conclude the information necessary to evaluate the ground-motion expert elicitation will be available by the time of a potential license application.

DOE recently indicated, however, that it may revise the ground-motion expert elicitation results, especially as they pertain to postclosure performance assessments. The revisions are in response to technical concerns (discussed in the following paragraphs and in Section 5.1.3.2) about the lack of realism in the earthquake ground motions from the DOE seismic hazard study at low annual exceedence probabilities (between approximately 10^{-6} and 10^{-8}). If DOE completes this reassessment prior to submission of a potential license application, Structural Deformation and Seismicity Agreement 2.01 may become irrelevant. If DOE uses the expert elicitation process to complete its reassessment, the staff will review the updated elicitation to confirm documentation is adequate to provide a transparent view of the updating process and the resulting judgments and that the elicitation uses appropriate methods.

7.4.4.3 Ground Water-Specific Discharge

In NUREG–1762 (NRC, 2002), the staff evaluation of how DOE estimated ground water-specific discharge values, including uncertainty, was discussed in the Expert Elicitation chapter, Administrative and Programmatic requirements. The technical issue relating to the ground water flux uncertainty range addressed during the Saturated Zone Flow and Transport Expert Elicitation Project is now discussed in Section 5.1.3.8, Flow Paths in the Saturated Zone, of this report. In summary, the current DOE approach to treating uncertainty in ground water-specific discharge relies mainly on site data and uses the expert elicitation estimates as supporting evidence for constraints placed on the range of uncertainty.

7.4.4.4 Sorption Coefficient Parameter Distributions

Sorption coefficient (K_d) parameter distributions are important to understand radionuclide transport phenomena in both the unsaturated and saturated zones (Sections 5.1.3.7 and 5.1.3.9 of this report).

Previous DOE estimates of parameter uncertainty were based on a series of informal expert judgments. A recent update (Bechtel SAIC Company, LLC, 2003, Attachments I and II) provided a more systematic technical basis for the K_d parameter distributions. In NUREG–1762 (NRC, 2002), the staff evaluation of how DOE estimated K_d parameter distributions, including uncertainty, was discussed in the Expert Elicitation chapter, Administrative and Programmatic requirements. The technical issues relating to the K_d parameter distributions are discussed in Sections 5.1.3.7, Radionuclide Transport in the Unsaturated Zone, and 5.1.3.9, Radionuclide Transport in the Saturated Zone in this report.

To improve the transparency and traceability of the DOE decisionmaking in this area, DOE agreed previously (Reamer, 2000) to provide the documentation necessary to evaluate adequacy of the technical basis used to support the expert elicitation or expert judgment and the DOE approach. Although upper and lower limits of the K_d parameter distributions are based on experimental data supported by process modeling (Bechtel SAIC Company, LLC, 2003, Attachments I and II), shapes of the distributions are assigned through expert judgment. DOE performed a series of bounding analyses on mildly sorbing (neptunium) and strongly sorbing (plutonium) radioelements to constrain the effects of parameter uncertainty on transport. These analyses indicate that uncertainty in retardation may reduce transport time significantly, and the documentation of any expert elicitation or expert judgment 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 as described in NUREG–1563 (NRC, 1996).

7.4.5 Status of Past DOE Elicitations

7.4.5.1 Probabilistic Volcanic Hazards Analysis

DOE and NRC disagree about the scope and magnitude of effects from the new aeromagnetic information on the DOE probability estimate derived by expert elicitation (Schlueter, 2002; Ziegler, 2003, 2002). The technical basis for this disagreement is discussed in Section 5.1.2.2 of this report. In summary, DOE considers all information developed since the 1995 elicitation, including the new aeromagnetic survey interpretations, as having insignificant effects on the DOE probability estimate derived by expert elicitation (Ziegler, 2003, 2002). Thus, DOE maintains the 1995 elicitation (CRWMS M&O, 1996) is suitable for use in a potential license application (Ziegler, 2003, 2002). NRC, however, considers the range of new information likely has significant effects on models and data used for the DOE probability elicitation. Thus, the 1995 DOE elicitation should be updated or the new information (Schlueter, 2002) included in analyses using other suitable techniques in NUREG–1563 (NRC, 1996). Information provided by DOE (Ziegler, 2003) to address the staff concerns regarding incorporation of new information into the 1995 DOE elicitation is being assessed by NRC.

7.4.5.2 Probabilistic Seismic Hazards Analysis

7.4.5.2.1 Seismic Source and Fault Displacement Characterization

No further action in this area is required at this time.

7.4.5.2.2 Ground-Motion Attenuation

To close this issue at the staff level, DOE should provide the documentation originally requested by NRC during the October 2000 Structural Deformation and Seismicity Technical Exchange (Schlueter, 2000b). Staff seek the 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 models in the broader probabilistic seismic hazard assessment. The DOE plans indicate the issue will be addressed in the forthcoming Technical Basis Document 14: Low Probability Seismic Events, which is scheduled to be available to NRC in June 2004 (Ziegler, 2004).

Alternatively, DOE may be revising the ground-motion expert elicitation to constrain the unrealistic ground motions predicted by the probabilistic seismic hazard assessment at low annual exceedence probabilities. This revision would make Structural Deformation and Seismicity Agreement 2.01 moot. Instead, NRC and CNWRA staffs would need to review the revised DOE approach and results during the potential license application review.

7.4.5.3 Ground Water-Specific Discharge

Staff evaluation of the DOE ground water-specific discharge value estimates, including uncertainty during the Saturated Zone Flow and Transport Expert Elicitation Project is now discussed in Section 5.1.3.8, Flow Paths in the Saturated Zone. No specific questions relate to the expert elicitation process. No open issues or concerns relate to specific discharge estimates obtained from expert elicitation.

7.4.5.4 Sorption Coefficient Parameter Distributions

DOE agreed previously (Reamer, 2000) to provide the documentation necessary to evaluate adequacy of the technical basis used to support expert elicitation or expert judgment and the DOE approach. Documentation provided by DOE should be adequate to allow an external reviewer to trace the elicitation or judgment process used to establish the shape of the K_d parameter distributions. In particular, DOE should provide information that is sufficiently complete to allow the reviewer to evaluate how the judgments are implemented in total system performance assessments in NUREG–1563 (NRC, 1996). Reasoning may be based on risk insights or on demonstration that the shape of the K_d parameter distributions is biased toward conservative (i.e., low K_d) values.

7.4.6 Summary

The staff has continued to monitor the DOE implementation of guidance in NUREG–1563 (NRC, 1996). Thus far, the NRC observation of the DOE-sponsored elicitations show no substantial deviations between the DOE implementation and the NRC guidance. Although some elicitations have weaknesses (Austin, 1997, 1996; Bell, 1998, 1997), these weaknesses do not appear fundamentally to change the conclusion or outcome of total system performance assessments. Because there are weaknesses in some elicitations, staff obtained detailed agreements from DOE to provide information that can resolve specific NRC concerns, as noted in Section 7.4.6.

Staff will continue to monitor any reexamination by DOE of elicitation results and any need to update these results when new site characterization, design, or performance assessment information becomes available. In this regard, DOE agreed to provide its administrative procedure describing treatment of new data after completion of an elicitation (Bell, 1997).

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7.5 Plans for Startup Activities and Testing

The DOE plans for startup activities and testing have not been the subject of DOE and NRC prelicensing discussions and no issues have been identified.

7.6 Plans for Conduct of Normal Activities, Including Maintenance, Surveillance, and Periodic Testing

The DOE plans for conduct of normal activities, including maintenance, surveillance, and periodic testing have not been the subject of DOE and NRC prelicensing discussions and no issues have been identified.

7.7 Emergency Planning

The requirements of Subpart I of 10 CFR Part 63 concerning emergency planning have not been the subject of DOE and NRC prelicensing discussions and no issues have been identified.

7.8 Controls to Restrict Access and Regulate Land Uses

The DOE controls to restrict access and regulate land use have not been the subject of DOE and NRC prelicensing discussions and no issues have been identified.

7.9 Uses of Geologic Repository Operations Area for Purposes Other Than Disposal of Radioactive Wastes

The DOE plans for uses of the geologic repository operations area for purposes other than the disposal of radioactive wastes have not been the subject of DOE and NRC prelicensing discussions and no issues have been identified.

7.10 License Specifications

The license specifications have not been the subject of DOE and NRC prelicensing discussions and no issues have been identified. License specifications will be identified during the NRC detailed safety review of the DOE license application.

8 SUMMARY AND CONCLUSIONS

This Integrated Issue Resolution Status Report provides the updated status of technical issues concerning the potential geologic repository at Yucca Mountain. These issues have been developed through interactions between the U.S. Department of Energy (DOE) and the U.S. Nuclear Regulatory Commission (NRC), consistent with the Nuclear Waste Policy Act (1982). These interactions, including document reviews and public technical exchanges, have focused on technical issues that, if addressed, will increase the likelihood that any DOE license application will contain the information necessary for an efficient and effective regulatory review.

Starting in August 2000, the DOE and NRC staffs conducted technical exchanges with the specific objective of prelicensing issue resolution of what were identified as the key technical issues. The 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 determined to be insufficient, NRC reached agreements with DOE to provide further information or analyses. These agreements specify the additional information DOE will collect, a schedule for obtaining such information, and a mechanism for providing the information to the NRC staff. The key technical issues are defined as resolved at the staff level when the NRC staff considers the information gathered by DOE sufficient for the staff to conduct a detailed technical review after submittal of a potential license application. Resolution, however, does not imply any conclusions regarding the end result of such a review, and any issue can be reopened if new information becomes available.

DOE completed submittal of its agreement responses to NRC in September 2004. The NRC staff is continuing its review of the DOE responses. With the few exceptions noted, this update of the Integrated Issue Resolution Status Report is based on information available through March 2004. The NRC staff will continue to review information provided by DOE, and will continue to provide feedback to DOE, until submittal of a potential license application.

This update of the Integrated Issue Resolution Status Report follows the structure of the Yucca Mountain Review Plan (NUREG–1804, NRC, 2003), and covers issues related to preclosure safety, postclosure performance, and other general, administrative, and programmatic aspects of the proposed repository, as drawn from the regulatory requirements of 10 CFR Part 63. The amount of information in each section of the report reflects the significance of the topic to repository safety and performance, and the extent of prelicensing interactions on that topic between NRC and DOE.

In the preclosure area, the Yucca Mountain Review Plan identifies 10 topics to be addressed in any future license application for the potential high-level waste repository at Yucca Mountain. Three of these topics (Meeting the 10 CFR Part 20 As Low As Is Reasonably Achievable Requirements for Normal Operations and Category 1 Event Sequences; Plans for Retrieval and Alternate Storage of Radioactive Wastes; and Plans for Permanent Closure and Decontamination, or Decontamination and Dismantlement of Surface Facilities) have not been the subject of DOE and NRC prelicensing discussions and no issues have been identified. Some information is available for each of the other seven topics (Sections 4.1.1–4.1.7 in Chapter 4), and the status of identified issues in these areas is discussed. It is important to note that the preliminary DOE facility design is being modified to include changes in layout, design, and functionality, and the amount of information available at the time of this review was limited. The DOE facility design will be evaluated during the NRC safety evaluation of any potential DOE license application.

In the area of postclosure performance assessment, four broad topics are identified in the Yucca Mountain Review Plan (Sections 5.1.1–5.1.4 in Chapter 5). The postclosure area has been the subject of the most extensive interactions between the NRC and DOE staffs, and topics in this area have been developed in the most detail. Nine key technical issues have been identified for postclosure, which have been further divided into 37 subissues.

As part of its high-level waste risk insights initiative, the NRC staff evaluated the significance for waste isolation of each of the postclosure key technical issue subissues. This risk analysis is presented in the Risk Insights Baseline Report (Appendix D). The risk insights were considered in developing the discussions of postclosure performance in the present report, and will help the NRC staff to focus its review of a potential license application.

The majority of the postclosure subissues are classified as closed-pending. Two hundred and ninety-three key technical issue agreements were reached with DOE that identified the information necessary for these subissues to gain the closed-pending classification. The full text of these agreements and the current status are provided in Appendix A. As of August 2004, the NRC staff had no further questions on 111 of the agreements and these are considered to be closed within the prelicensing context.

The postclosure performance assessment includes 14 model abstractions for the projected behavior of the natural and engineered barrier systems. These are discussed in detail (Section 5.1.3) following the five review methods outlined in NUREG–1804 (NRC, 2003). By these review methods for each model abstraction, in most cases it is likely that, along with the key technical issue agreements, information necessary to assess the topic will be available at the time of a potential license application. For those cases where this is not apparent, the specific areas where additional information may be necessary are elucidated.

In the general, administrative, and programmatic areas, prelicensing interactions between the NRC and DOE have been limited to a few specific areas, and this is reflected in the level of detail in the present report. The general information topics covered in Chapter 3 have had very limited interaction, and no issues have been identified during prelicensing. Some limited information is available on the topic of performance confirmation (discussed in Chapter 6), and information has been developed on two administrative and programmatic topics, quality assurance and expert elicitation (Chapter 7). The other areas noted in Chapters 6 and 7 have had little or no prelicensing interaction, and no issues have been identified.

For performance confirmation, the DOE activities conducted to date, as part of site characterization, have established some baseline information for the Yucca Mountain site. While DOE has given a preliminary overview of its anticipated program, a complete performance confirmation plan was not available for review in the present report. The DOE quality assurance program content appears sufficient for use in developing a potential license application, for the type of activities performed by DOE to date, but the program has not been reviewed with respect to the full range of activities to be conducted at the potential repository. Expert elicitation is used in a number of technical areas. While generally in accord with accepted procedures, the NRC staff has noted in several instances a lack of complete documentation of the elicitation process.

The NRC staff recognizes that a number of the DOE plans and programs to meet the administrative and programmatic requirements of 10 CFR Part 63 are still in development and will not be addressed in detail until a potential application for receipt and possession of nuclear materials is submitted. The knowledge available at the time of construction authorization is likely to be less than at the subsequent licensing stages. However, at each stage, DOE should provide sufficient information to support that stage.

In summary, the information contained in this update of the Integrated Issue Resolution Status Report has been developed during prelicensing interactions between the NRC and DOE staffs. The prelicensing activities are intended to increase the likelihood that any potential license application will be of high quality, so that the NRC staff will be able to complete its review in an effective and efficient manner. The NRC evaluation will begin with submittal of a potential license application. The results of the technical review by the NRC staff will be documented in the safety evaluation report.

8.1 References

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10. SUPPLEMENTARY NOTES

Revision of NUREG-1762; supercedes previous version

11. ABSTRACT *(200 words or less)*

This Integrated Issue Resolution Status Report provides background information about 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 has, for many years, engaged in prelicensing interactions with DOE and various stakeholders. In recent years, DOE and NRC have reached a number of agreements related to key technical issues important to repository performance after permanent closure and items important to safety during the period before permanent closure. During the prelicensing period, the NRC staff also have undertaken a risk insights initiative to enhance the use of available risk information and develop, as a common basis for understanding, the significance of features, events, and processes that may affect the performance of potential engineered and natural barriers at Yucca Mountain.

This report provides an overview of available information and status (as of March 2004, with exceptions as noted) of the Key Technical Issue agreements reached between DOE and NRC. The report also documents the risk insights (Appendix D) and information considered by the NRC staff in formulating their views, including in-depth reviews of available DOE and contractor documents; the independent confirmatory work of NRC and its contractor, the Center for Nuclear Waste Regulatory Analyses; published literature; and other publicly available information.

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