NUREG-1762

Integrated Issue Resolution Status Report

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

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Integrated Issue Resolution Status Report

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ABSTRACT

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

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

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

CONTENTS

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

CONTENTS (continued)

FIGURES

FIGURES (continued)

Figure | Page

TABLES

TABLES (continued)

EXECUTIVE SUMMARY

Introduction

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

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

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

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

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

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

Background

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

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

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

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

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

Report Structure

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

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

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

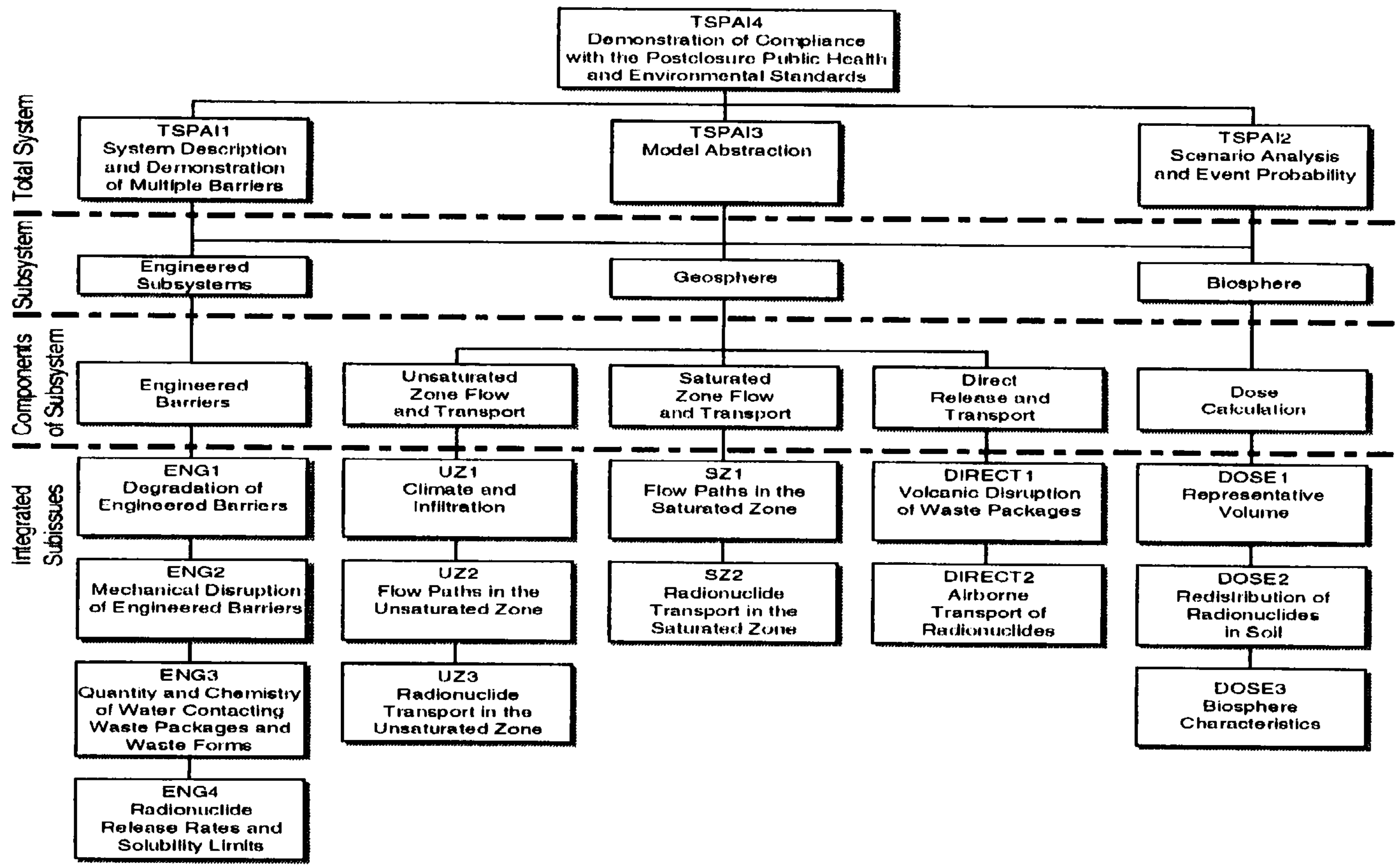

Figure 1. Components of Postclosure Performance Assessment Review

Preclosure Summary

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

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

Postclosure Summary

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

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

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

Summary

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

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

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

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

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

PREFACE

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

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

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

<u>References</u>

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

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

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

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

1 INTRODUCTION

1.1 Background and Report Structure

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

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

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

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

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

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

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

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

Table 1.1-1. Key Technical Issues and Associated Subissues

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

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

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

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

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

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

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

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

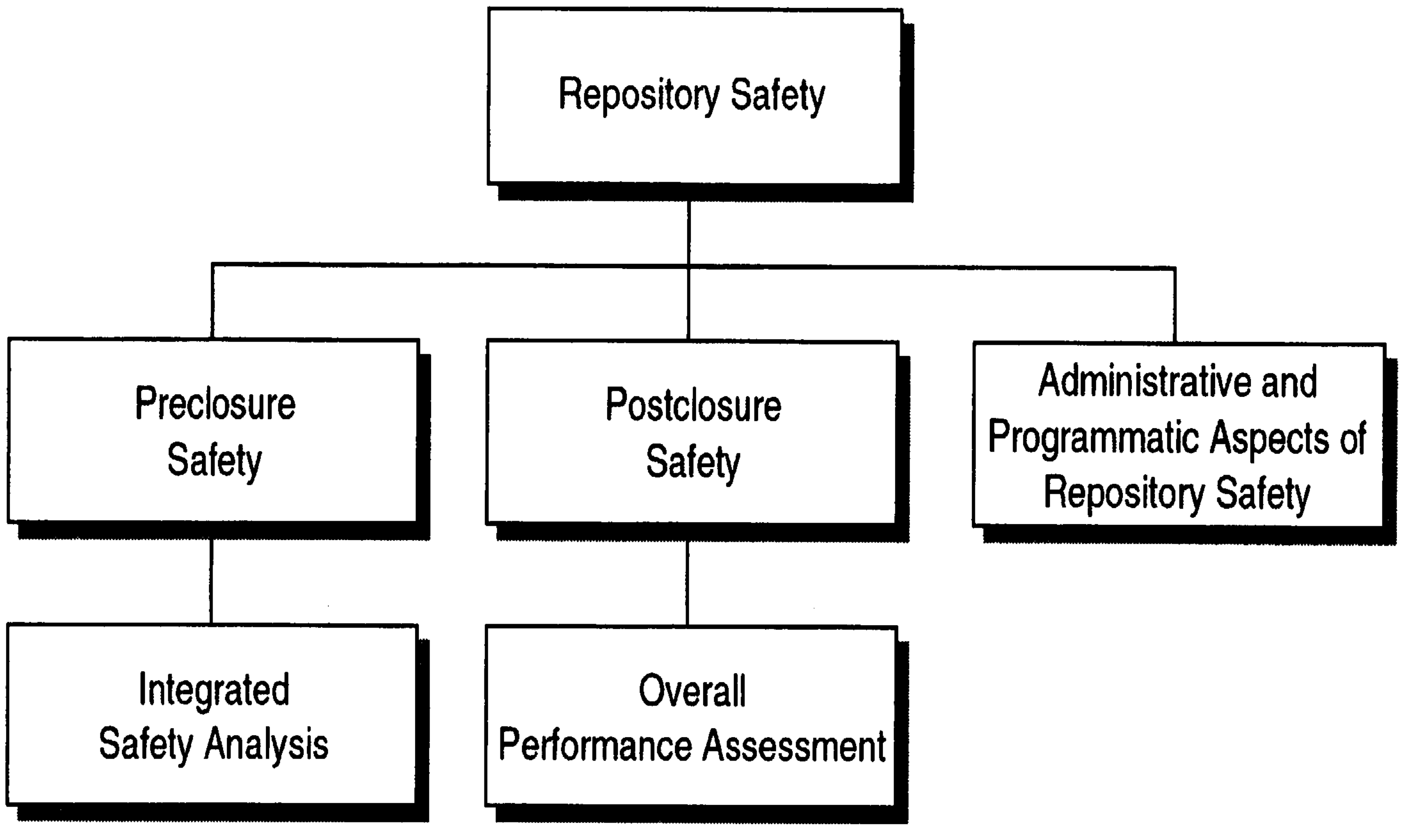

Figure 1.1-1. Review Components of Repository Safety

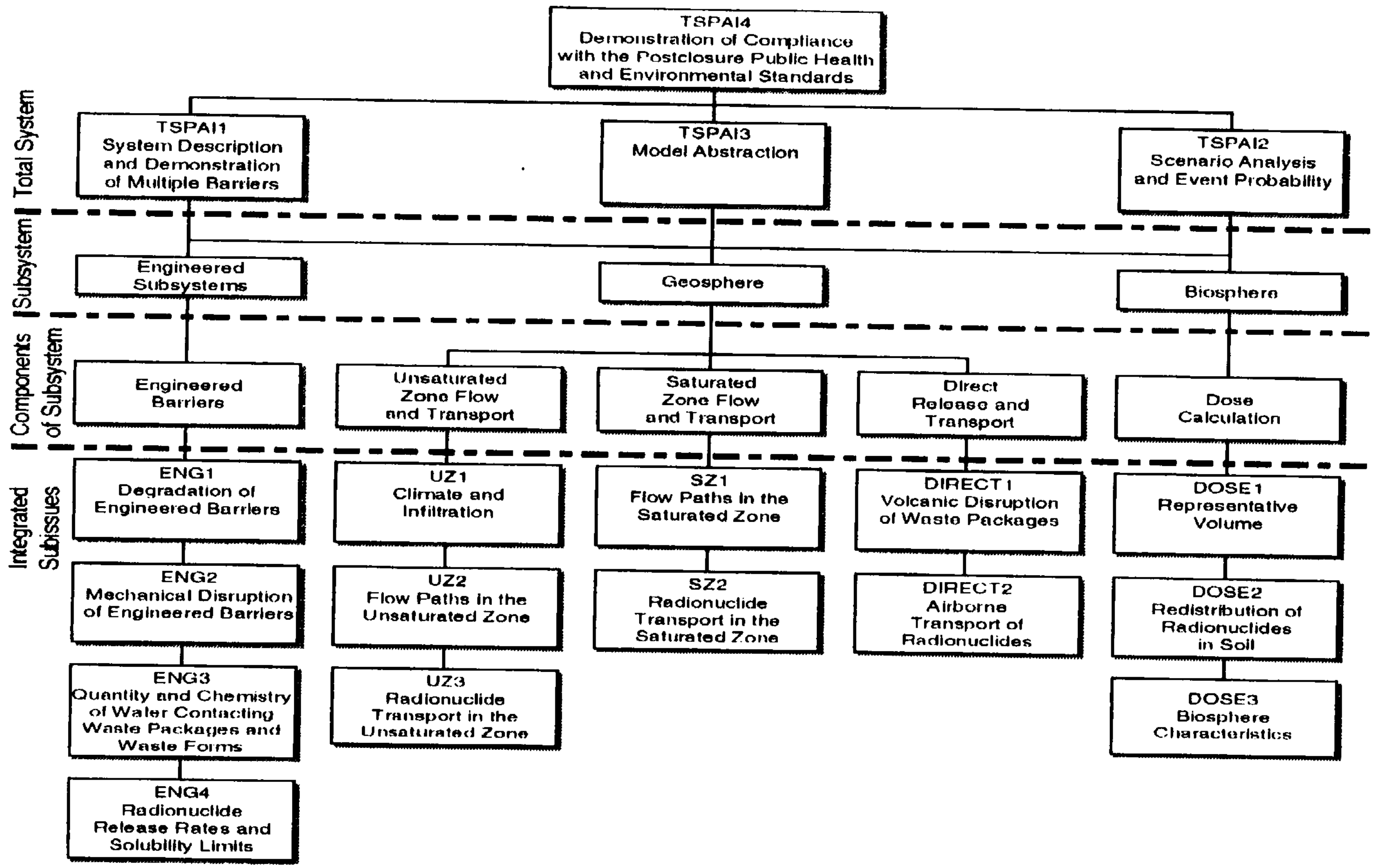

Figure 1.1-2. Components of Postclosure Performance Assessment Review

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

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

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

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

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

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

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

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

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

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

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

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

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

1.2 Prelicensing Issue Resolution Process

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

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

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

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

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

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

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

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

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

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

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

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

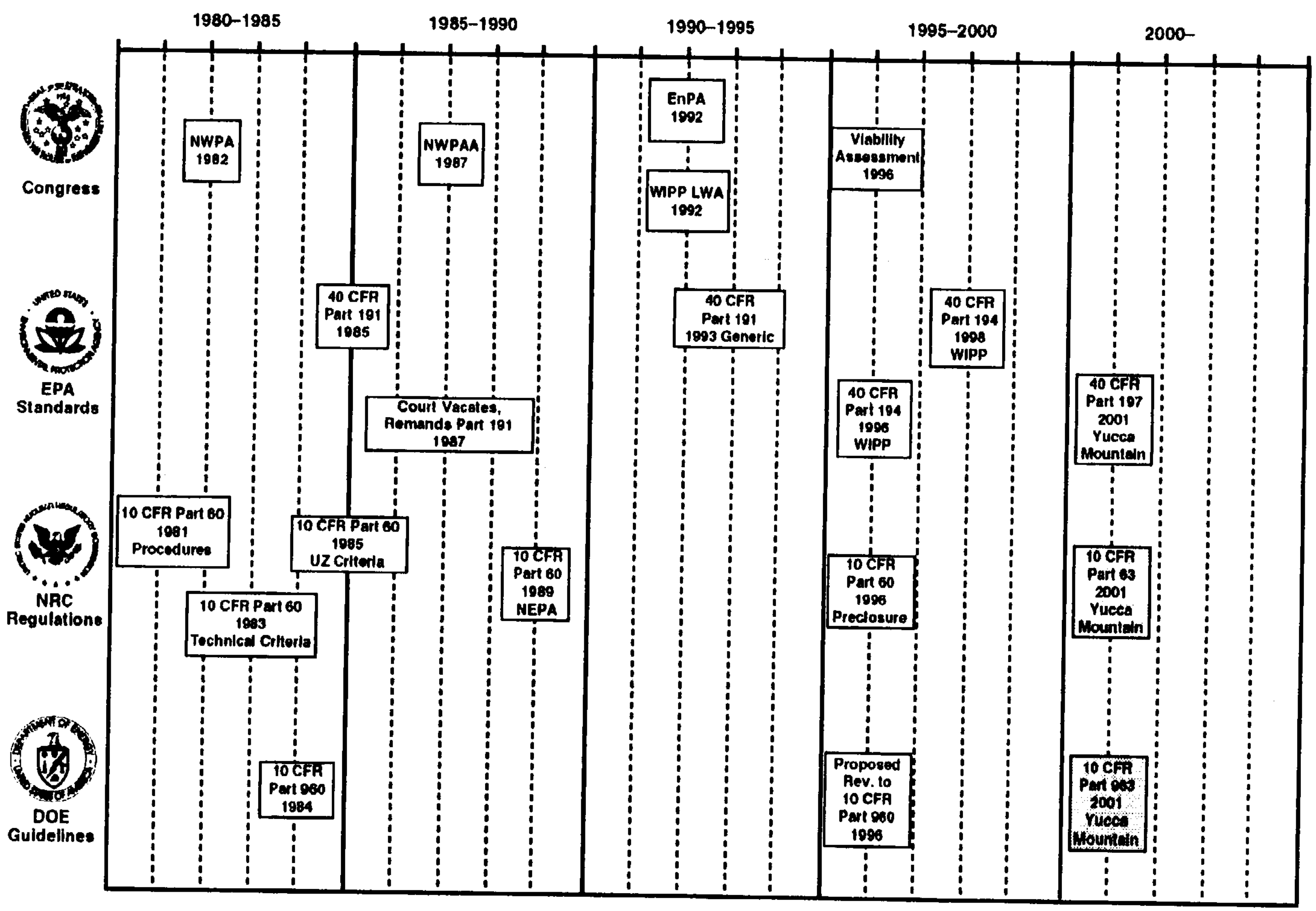

Figure 1.1-3. Timeline of Legislative and Regulatory Events, 1980–2000

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

EPA Standards

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

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

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

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

NRC Regulations

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

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

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

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

Introduction

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

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

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

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

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

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

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

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

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

Identification of the NRC Policy Issues

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

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

1.4 Risk-Informing NRC Reviews

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

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

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

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

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

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

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

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

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

1.5 Preclosure and Postclosure Review Processes

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

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

- Have consistent and appropriate assumptions and initial and boundary conditions been propagated throughout the DOE models and calculations?
- Are the conditions and assumptions used to generate any look-up tables or regression equations consistent with other conditions and assumptions in the preclosure and postclosure safety analyses?
- Have important design features that will set the initial and boundary conditions for models and calculations been included?
- Has DOE considered the space-time dimensionality appropriately?
- Have important physical phenomena and couplings been included in the preclosure and postclosure safety analyses?
- Has sufficient justification been provided for any excluded coupling?

(2) Data Are Sufficient for Model Justification

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

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

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

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

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

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

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

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

- Are any bounding assumptions technically defensible?

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

- Have any correlations between parameter values been appropriately considered?

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

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

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

- Has DOE considered plausible alternative models?

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

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

(5) Model Abstraction Output Is Supported by Objective Comparisons

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

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

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

1.6 References

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

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2 REPOSITORY SAFETY BEFORE PERMANENT CLOSURE

2.1 Preclosure Safety Analysis

2.1.1 Site Description As It Pertains to Preclosure Safety Analysis

2.1.1.1 Description of Issue

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

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

2.1.1.2 Importance to Safety

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

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

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

2.1.1.3 Technical Basis

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

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

2.1.1.3.1 Site Geography

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

Site Location

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

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

Significant Natural and Manmade Features

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

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

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

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

Repository Safety Before Permanent Closure

Site Maps

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

2.1.1.3.2 Regional Demography

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

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

2.1.1.3.3 Local Meteorology and Regional Climatology

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

Climate and Meteorological Conditions

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

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

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

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

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 102–107 mm/yr [4 in/yr] in the Amargosa farms area. Average precipitation values are based on 30-year records.

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

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

Severe Weather

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

2.1.1.3.4 Regional and Local Surface and Groundwater Hydrology

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

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

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

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

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

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

Surface Waters

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

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

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

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

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

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

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

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

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

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

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

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

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

Groundwater

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

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

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

Repository Safety Before Permanent Closure

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

<u>Summary</u>

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

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

2.1.1.3.5 Site Geology and Seismology

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

Site Geology

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

Regional Geologic Setting

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

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

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

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

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

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

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

Regional Tectonic Setting

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

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

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

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

Repository Safety Before Permanent Closure

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

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

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

Quaternary Stratigraphy and Surficial Processes

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

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

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

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

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).

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

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

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

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

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

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

<u>Site Structural Geology</u>

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

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

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

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

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

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

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

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

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

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

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

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

Site Geoengineering Properties

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

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

Integrated Site Model

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

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

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

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

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

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

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

<u>Natural Resources</u>

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

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

Rock Properties

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

Stability and Suitability of Subsurface Materials

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

Soil Properties

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

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

Stability and Suitability of Surface Materials

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

Seismic and Faulting Hazards

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

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

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

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

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

Seismic Design

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

Facility Stability

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

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

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

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

2.1.1.3.6 Igneous Activity

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

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

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

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

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

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

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

2.1.1.3.7 Site Geomorphology

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

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

2.1.1.3.8 Site Geochemistry

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

<u>Geochemistry of Subsurface Waters</u>

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

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

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

Geochemistry of Rock Strata

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

Geochemical Alterations

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

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

2.1.1.4 Status and Path Forward

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

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

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

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

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

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

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

2.1.2.1 Description of Issue

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

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

2.1.2.2 Importance to Safety

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

2.1.2.3 Technical Basis

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

- Description of high-level waste characteristics
- Descriptions and design details of engineered barrier system components (e.g., waste package, drip shield, and backfill, if any)
- Description of geologic repository operations area processes activities and procedures including human interactions and interfaces and interactions between structures, systems, and components.

2.1.2.4 Status and Path Forward

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

2.1.2.5 References

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

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

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

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

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

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2.1.3 Identification of Hazards and Initiating Events

2.1.3.1 Description of Issue

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

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

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

2.1.3.2 Importance to Safety

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

2.1.3.3 Technical Basis

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

Repository Safety Before Permanent Closure

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Table 2.1.3-2. List of Natural Hazards with DOE Assessment (after CRWMS M&O, 1999a; DOE, 2001a) (continued)			
No.	**Hazard**	**Hazard Definition**	**DOE Assessment**
7	Eperogenic Displacement	Geomorphic processes of uplift and subsidence that produced broader features of continents and oceans	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is not sufficient to pose credible hazard during 100-year preclosure period
8	Erosion	Slow wearing of soil and rock by weathering, mass wasting, and action of streams	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process not sufficient to pose credible hazard during 100-year preclosure period
9	Extreme Weather Fluctuations	Various types of weather fluctuations that pose unusual design challenges	Not applicable to the hazards list • No potential exists at Yucca Mountain
10	Extreme Wind	Fastest mile of wind with 100-year return period	Applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficient during 100-year preclosure period • Potential consequence is indeterminant; assumed equivalent to true • Annual event frequency $\geq 10^{-6}$ • Not included in another analysis
11	Flood (Storm, River Diversion)	Area covered with water from storm or river diversion caused by inadequate drainage	Applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficiently high during 100-year preclosure period • Consequences of process are sufficiently high • Annual event frequency $\geq 10^{-6}$ • Not included in another analysis
12	Fungus, Bacteria, and Algae	General class of microorganisms that may be present in subsurface environment	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficiently high during 100-year preclosure period • Consequence of process not significant to affect 100-year preclosure period
13	Glacial Erosion	Lowering of Earth's surface due to grinding and scouring by glacier ice armed with rock fragments	Not applicable to the hazards list • No potential exists at Yucca Mountain for a glacier
14	Glaciation	Formation, movement, and recession of glaciers or ice sheets	Not applicable to the hazards list • No potential exists at Yucca Mountain for a glacier and associated climate change

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

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

Table 2.1.3-2. List of Natural Hazards with DOE Assessment (after CRWMS M&O, 1999a; DOE, 2001a) (continued)			
No.	**Hazard**	**Hazard Definition**	**DOE Assessment**
22	Low River Level	Low level of river water used for cooling	Not applicable to the hazards list • No potential exists at Yucca Mountain because there is no river nearby
23	Meteorite Impact	Impact of meteoroid reaching Earth's surface without completely vaporizing	Not applicable to the hazards list. • Potential exists at Yucca Mountain • Rate of process is sufficiently high during 100-year preclosure period • Consequence is indeterminant; assumed equivalent to true • Annual event frequency $\leq 10^{-6}$
24	Orogenic Diastrophism	Movement of Earth's crust produced by tectonic processes where structures within fold-belt mountain areas formed, including thrusting, folding, and faulting	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is too low to affect 100-year preclosure period
25	Rainstorm	Storm that produces 100-year or greater maximum rainfall rate occurring for one day	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficiently high during 100-year preclosure period • Consequence is indeterminant; assumed significant • Annual event frequency $\geq 10^{-6}$ • Bounded by debris avalanche, flooding, and landslide events for which this is initiator
26	Range Fire	Combustion of natural vegetation external to repository that propagates to combustible materials within operations area	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficiently high during 100-year operational period • Consequences are significant • Annual event frequency $\geq 10^{-6}$ • Will be addressed in fire hazard analyses

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

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

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

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

Table 2.1.3-2. List of Natural Hazards with DOE Assessment (after CRWMS M&O, 1999a; DOE, 2001a) (continued)			
No.	**Hazard**	**Hazard Definition**	**DOE Assessment**
36	Subsidence	Sudden sinking or gradual downward settling of Earth's surface with little or no horizontal motion	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficiently high to affect 100-year preclosure period • Consequence is indeterminant; assumed significant • Annual event frequency $\geq 10^{-6}$ • Screened out because subsurface fault displacement will be only natural phenomenon that would result in collapse of underground excavations leading to subsidence; emplacement levels would be at least 200 m [656 ft] below the directly overlying ground surface; emplacement drifts will be supported by rock bolts, steel mesh, and steel sets; no surface-handling facilities will be directly over emplacement drifts
37	Tornado	Small cyclone generally less than 500 m [1,650 ft] in diameter with extremely strong winds	Applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficient to affect 100-year preclosure period • Consequence is indeterminant; hence assumed significant • Annual event frequency $\geq 10^{-6}$ • Not bounded by another analysis
38	Tsunami	Gravitational sea wave produced by large-scale, short-duration disturbance on ocean floor	Not applicable to the hazards list • No potential exists at Yucca Mountain because there is no coastal region
39	Undetected Geologic Features	Geologic features of concern to the 100-year preclosure period include natural events such as faults and volcanoes	Not applicable to the hazards list • No potential exists at Yucca Mountain; site characterization provided sufficient assurance that these types of activities would have been detected
40	Undetected Geologic Processes	Geologic processes of concern to the 100-year preclosure period include events such as erosion, tectonic, and seismic processes	Not applicable to the hazards list • No potential exists at Yucca Mountain; site characterization provided sufficient assurance that these types of activities would have been detected

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

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

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

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

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

Table 2.1.3-3. List of Human-Induced Events with DOE Assessment (after CRWMS M&O, 1999a; DOE, 2001a) (continued)			
No.	**Hazard**	**Hazard Definition**	**DOE Assessment**
5	Loss of Off-site/On-site Power	Loss of electric power either generated or controlled by persons outside repository system or loss of power within repository	Applicable to the hazards list • Potential exists at Yucca Mountain • Rate of the process is indeterminant at this time, hence assumed significant • Consequence is indeterminant; hence assumed significant • Annual event frequency is indeterminant at this time; hence assumed significant • Not bounded by another analysis
6	Military Activity-Induced Accidents	Accidents resulting from military activities Nevada Test Site or Nellis Air Force Range	Applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is indeterminant at this time; hence assumed significant • Consequence of the process is indeterminant at this time; hence assumed significant • Annual event frequency is indeterminant at this time; hence assumed significant • Not bounded by another analysis
7	Pipeline Accidents	Industrial pipeline transporting hazardous materials	Not applicable to the hazards list • No potential exists at Yucca Mountain; no industrial activities requiring pipelines containing hazardous materials exist or are planned to be located near the site
8	Undetected Past Intrusions	Past intrusions involve mining activities where deep shafts, drill holes, or tunnels may have been excavated	Not applicable to the hazards list • No potential exists at Yucca Mountain; site characterization provided sufficient assurance that these types of activities would have been detected

2.1.3.3.1 Hazards and Initiating Events Consideration

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

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

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

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

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

- Event is not bounded by analysis of another event.

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

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

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

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

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

2.1.3.3.1.1 Aircraft Crash Hazard

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

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

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

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

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

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

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

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

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

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

2.1.3.3.1.2 Tornado Missiles Hazard

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

2.1.3.3.1.3 Volcanic Ash Fall Hazard

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

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

[2]Ibid.

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

2.1.3.3.1.4 Operational Hazards

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

2.1.3.3.2 Site Data

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

2.1.3.3.2.1 Aircraft Crash Hazard

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

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

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

[4]Ibid.

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

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

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

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

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

[6]Ibid.

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

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

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

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

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

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

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

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

[9]Ibid.

[10]Ibid.

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

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

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

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

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

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

[12] Ibid.

[13] Ibid.

[14] Ibid.

2.1.3.3.2.2 Tornado Missiles Hazard

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

2.1.3.3.2.3 Volcanic Ash Fall Hazard

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

2.1.3.3.3 Probability of Occurrence Determination

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

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

[16]Ibid.

[17]Ibid.

2.1.3.3.3.1 Aircraft Crash Hazard

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

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

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

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

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

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

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

Approximately 54,000 annual flights of DOE aircraft use Desert Rock, Yucca, and Pahute Mesa airfields (CRWMS M&O, 1999d). Information is not available, however, about the number of annual flights to each of these airfields. Staff assumed, in one scenario, that all 54,000 flights use Desert Rock airfield. Staff also made another estimate assuming one-third of the 54,000 flights use each airport, which, by nature of the runway surface, is not a valid assumption. The effective area, A_{eff} of the surface facilities at the proposed repository has been calculated as the sum of the effective areas of each of the five structures where radioactive materials potentially can be located (CRWMS M&O, 1999d) and is equal to 0.641 km² [0.251 mi²] (Ghosh and Sagar, 2001). The effective width of the airway, *W*, is $16 + 2 \times 17.6$, or 51.2 km [32 mi], because the airway V105–V135 is 16 km [10 mi] wide and at a distance of 17.6 statute miles [11 statute miles] from the proposed site (CRWMS M&O, 1999d). Therefore, the annual probability of crash, *P*, from DOE aircraft and aircraft chartered by DOE, based on NRC (1981a), is

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

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

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

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

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

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

[20]Ibid.

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

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

2.1.3.3.3.2 Tornado Missiles Hazard

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

2.1.3.3.3.3 Volcanic Ash Fall Hazard

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

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

[22]Ibid.

[23]Ibid.

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

2.1.3.3.4 Exclusion or Inclusion of Hazards and Initiating Events

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

2.1.3.3.4.1 Aircraft Crash Hazard

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

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

[25]Ibid.

2.1.3.3.4.2 Tornado Missiles Hazard

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

2.1.3.3.4.3 Volcanic Ash Fall

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

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

2.1.3.3.4.4 List of Hazards and Initiating Events

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

2.1.3.4 Status and Path Forward

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

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

[27]Ibid.

[28]Ibid.

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

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

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

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

2.1.3.5 References

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

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2.1.4 Identification of Event Sequences

2.1.4.1 Description of Issue

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

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

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

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

2.1.4.2 Importance to Safety

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

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

2.1.4.3 Technical Basis

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

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

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

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

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

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

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

2.1.4.3.1 Justification for Methodology and Assumptions

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

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

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

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

[3]Ibid.

2.1.4.3.2 Identification of Category 1 and 2 Event Sequences

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

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

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

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

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

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

2.1.4.4 Status and Path Forward

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

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

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

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

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

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

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

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

2.1.4.5 References

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

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

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

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

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

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

[8]Ibid.

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

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

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

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

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

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

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

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

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

2.1.5 Consequence Analyses

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

2.1.5.1.1 Description of Issue

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

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

2.1.5.1.2 Importance to Safety

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

2.1.5.1.3 Technical Basis

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

2.1.5.1.3.1 Hazard Consideration

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

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

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

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

2.1.5.1.3.2 Methods and Assumptions

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

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

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

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

2.1.5.1.3.3 Compliance with Regulatory Requirements

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

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

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

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

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

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

[2]Ibid.

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

2.1.5.1.4 Status and Path Forward

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

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

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

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

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

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

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

2.1.5.1.5 References

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

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

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

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

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

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

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

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

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

2.1.5.2.1 Description of Issue

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

2.1.5.2.2 Importance to Safety

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

2.1.5.2.3 Technical Basis

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

2.1.5.2.3.1 Hazard Consideration

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

2.1.5.2.3.2 Methods and Assumptions

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

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

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

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

2.1.5.2.3.3 Compliance with Regulatory Requirements

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

2.1.5.2.4 Status and Path Forward

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

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

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

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

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

2.1.5.2.5 References

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

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

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

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

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

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

2.1.6.1 Description of Issue

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

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

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

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

2.1.6.2 Importance to Safety

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

2.1.6.3 Technical Basis

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

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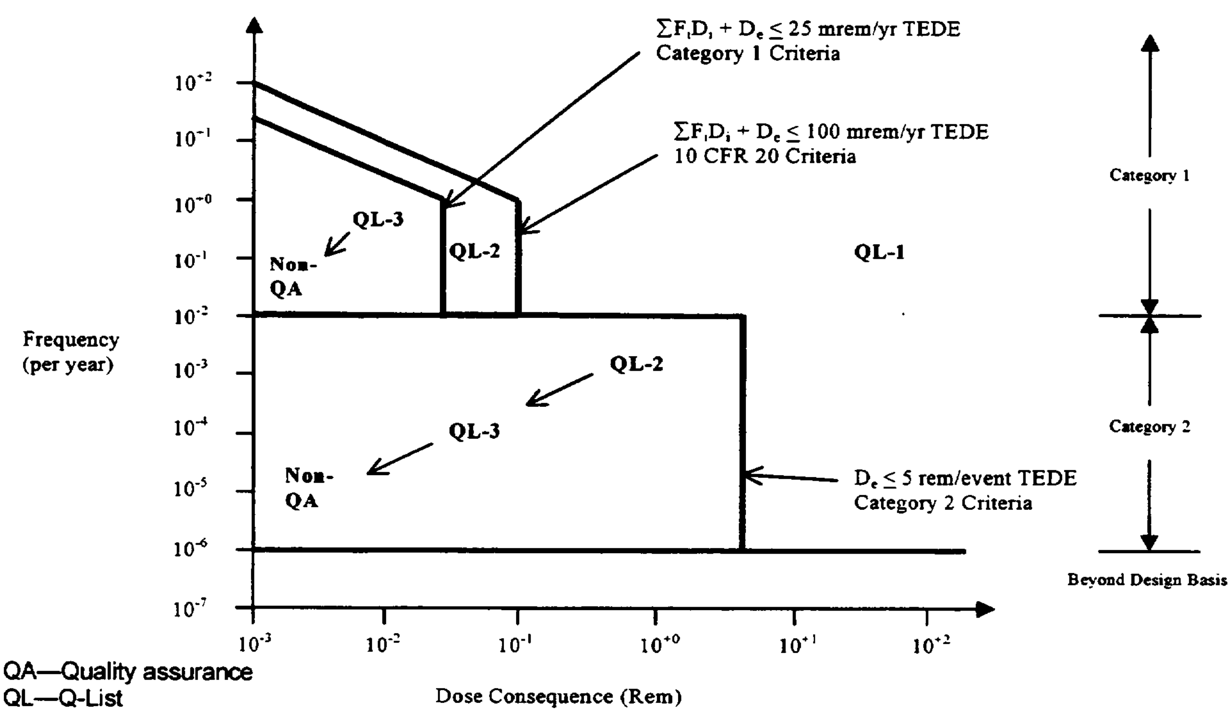
[1]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

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

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



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

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

Repository Safety Before Permanent Closure

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

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

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

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

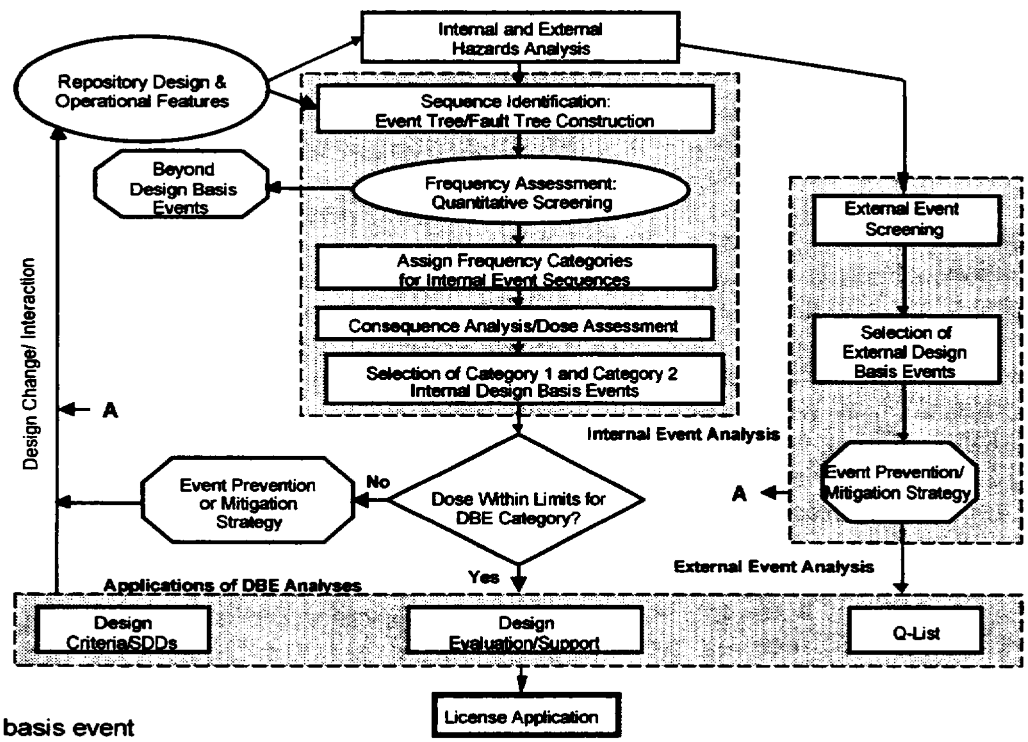
The following discussion identifies concerns associated with the DOE list of structures, systems, and components important to safety. Each of the following concerns was discussed in the DOE and NRC Technical Exchange and Management Meeting on Preclosure Safety and, agreements were reached for the resolution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

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

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

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

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



DBE—Design basis event
SDD—System description document

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

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

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

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

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

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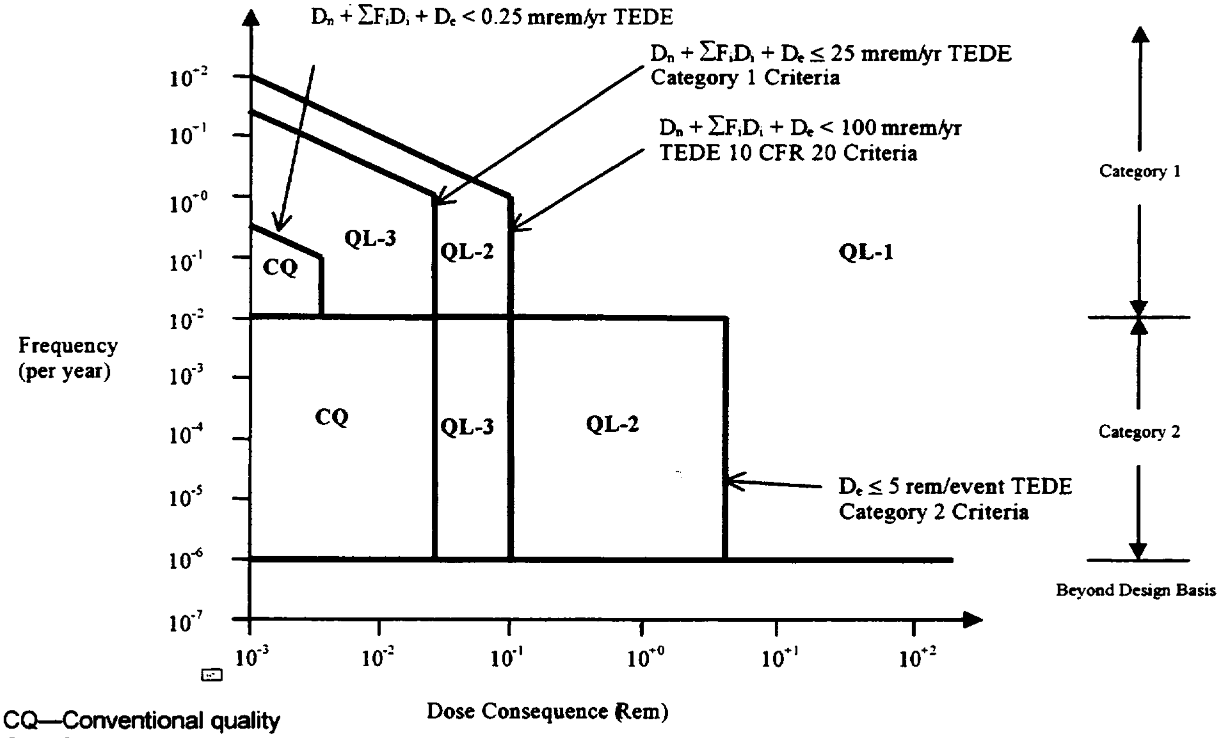
[6]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Tehnical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[7]Ibid.

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



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

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

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

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

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

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

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

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

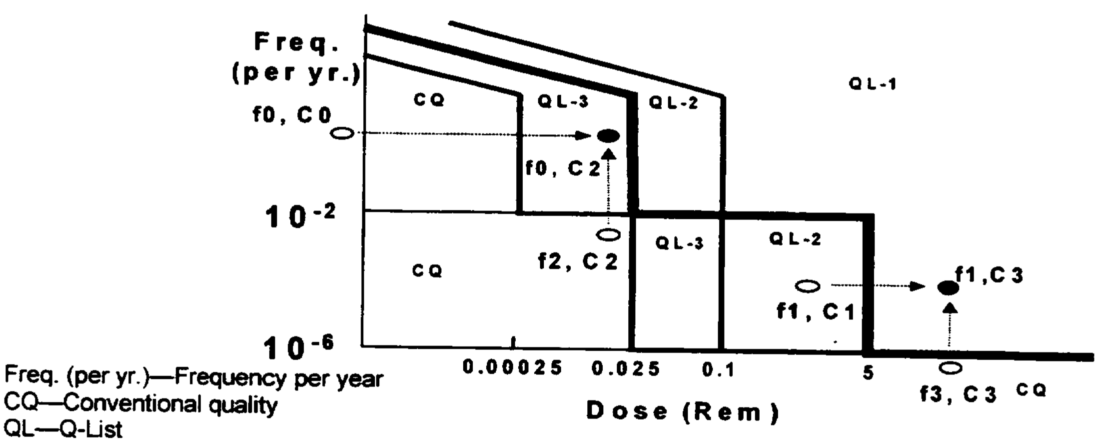

Figure 2.1.6-4. Overview of the DOE Proposed Classification Process[12]

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

[12]Ibid.

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

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

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

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

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

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

2.1.6.3.2 Administrative or Procedural Safety Controls Are Adequate

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

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

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

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

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

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

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

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

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

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

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

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

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

[18]Ibid.

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

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

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

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

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

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

[21]Ibid.

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

[23]Ibid.

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

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

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

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

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

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

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

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

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

2.1.6.4 Status and Path Forward

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

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

Repository Safety Before Permanent Closure

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

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

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

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

2.1.6.5 References

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

2.1.7.1 Description of Issue

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

Repository Safety Before Permanent Closure

Surface Facility

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

Subsurface Facility

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

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

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

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

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

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

Engineered Barrier Subsystem

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

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

Repository Safety Before Permanent Closure

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

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

2.1.7.2 Importance to Safety

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

Surface Facility

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

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

Subsurface Facility

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

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

Engineered Barrier Subsystem

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

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

2.1.7.3 Technical Basis

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

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

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

2.1.7.3.2 Geologic Repository Operations Area Design Methodologies

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

2.1.7.3.3 Geologic Repository Operations Area Design and Design Analyses

2.1.7.3.3.1 Surface Facilities

Assumptions, Codes, and Standards for Surface Facilities Design

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

Materials for Surface Facilities Design

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

Load Combinations for Surface Facilities Design

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

Design Analyses and Documentation

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

2.1.7.3.3.2 Subsurface Facility

Assumptions, Codes, and Standards for Subsurface Facility Design

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

Subsurface Operating Systems

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

Materials and Material Properties for Subsurface Facility Design

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

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

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

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

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

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

Load Combinations for Subsurface Facility Design

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

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

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

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

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

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

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

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

[4]Ibid.

[5]Ibid.

Models and Rock Properties for Subsurface Facility Design

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

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

Model Boundary Conditions

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

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

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

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

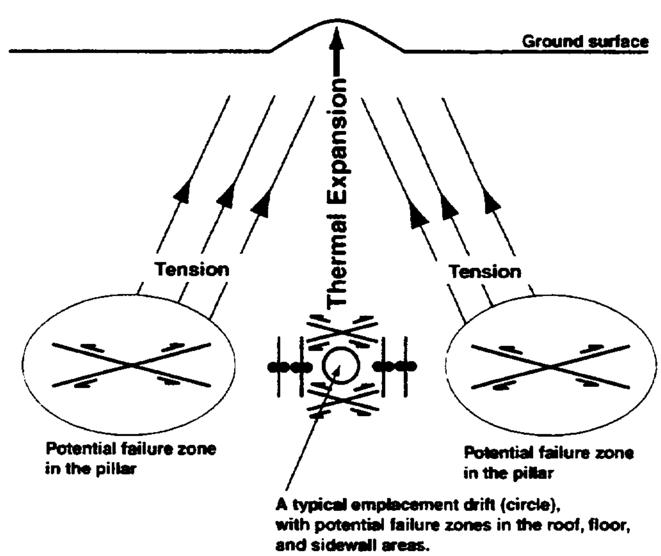

Figure 2.1.7-1. Schematic Illustration of the Anticipated Mechanisms of Thermal-Mechanical Response, Showing the Effects of the Emplacement Geometry on the Distributions of Zones of Potential Rock Failure in a Horizontal Array of Drifts. (Actual Development of the Failure Zones Would Be Determined by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

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

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

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

<u>Model Dimensionality</u>

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

The NRC staff concern about the appropriateness of two-dimensional thermal-mechanical modeling of the emplacement drifts arises because the *in-situ* horizontal principal stresses (Stock, et al., 1985) and several of the fracture sets (CRWMS M&O, 2000n) are oblique to the

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

proposed drift alignment (252° azimuth, that is S72 °W). The ambient minimum principal stress is horizontal and oriented N60 °W–N65 °W (Stock, et al., 1985), which is 40–45 degrees from the drift-normal plane (the assumed orientation of the minimum principal stress for the two-dimensional modeling). Also, the dip direction of the subhorizontal fractures, which are likely to dominate the rock-failure mechanism as illustrated in Figure 2.1.7-1 (Ofoegbu, 2001), lies in the 40–60-degree range (i.e., 10–30 ° from the drift orientation). Therefore, the two-dimensional models are not favorably oriented to detect slip on the subhorizontal fractures. Three-dimensional modeling may be necessary to determine the effects of these structural features that are oblique to the drift alignment.

Other areas for which three-dimensional modeling may also be necessary include (i) stability of the turnout area (between the emplacement drifts and the access mains), which may be subjected to a combination of vertical tension and high-horizontal compression similar to the phenomenon illustrated in Figure 2.1.7-1 (Ofoegbu, 2001); (ii) effects of greater heat conduction rates through the drift floor because steel members in the floor (invert and pallet) that are in direct or indirect contact with the waste package provide a faster heat-flow path into the rock; (iii) stability of the structural components of the invert (transverse and longitudinal beams) and the interaction of the transverse beams with the drift wall under heated conditions; and (iv) effects of ground-surface topography drift-parallel thermal gradients on thermal stress and, consequently, drift stability. DOE has agreed[8] to address this concern.

Model Representation of Fracture Network

Discontinuum models used in the thermal-mechanical analyses for site recommendation (CRWMS M&O, 2000k) were based on a regular fracture pattern composed from the mean fracture-set attitudes (dip and dip direction) and spacing, but the uncertainties in the fracture-set properties and their effects on the calculated results were not discussed. The DOE fracture data (CRWMS M&O, 2000n,o) indicate a considerable variation of the fracture-attitude parameters and spacing around the mean values for fracture sets, which means that the *in-situ* fracture pattern is irregular and variable. The simplified pattern used in the DOE analyses may be adequate for conducting numerical experiments, but the differences between the model and *in-situ* fracture patterns should be understood and factored into the interpretation of the analyses results and the facility design. DOE has agreed[9] to address this concern.

Model Representation of Seismic Loading

Seismic loading was represented in the models as a sinusoidal velocity history with a frequency of 10 Hz, an amplitude equal to the estimated peak ground velocity for the site, and a duration of 3 seconds (CRWMS M&O, 2000k). This approach for representing seismic loading was based on three assumptions (CRWMS M&O, 2000k, Sections 5.3.1–5.3.3). DOE assumed that (i) the use of a sinusoidal wave of constant amplitude is conservative because it results in

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

[9]Ibid.

applying more cycles of the peak ground velocity at a point than would occur in an actual seismic event; (ii) a frequency of 10 Hz results in a seismic wavelength of a few hundred meters {considering the estimated shear wave velocity of approximately 3,000 m/s [9,843 ft/s]}, and this wave length is appropriate because seismic waves generally have large wave lengths; and (iii) the 30 cycles of motion that result from applying a 10-Hz sinusoidal motion for 3 seconds is conservative because the host rock does not show significant nonlinear behavior during seismic loading.

The justifications given for the three assumptions do not include an explanation of how it was determined that the applied velocity history constitutes an adequate representation of the ground-motion time history for Yucca Mountain. The site-specific ground-motion time history would differ from the model velocity history in terms of frequency content, amplitude variation, and duration of loading, so a comparison of the two might examine the total energy delivered to the rock in either case and the amount of that energy available to cause rock failure (e.g., by fracture slip). Such a comparison may be accomplished through a combination of theoretical analysis, scaled-model testing, and numerical experimentation. Numerical modeling results indicate that the dynamic response of the rock mass surrounding the emplacement drifts could be underestimated if a sinusoidal motion with a frequency of 10 Hz and a duration of 3 seconds is used in the analysis instead of the site-specific ground motions (Hsiung, et al., 2001). This overestimation could potentially result in a design of a ground-support system that is insufficient. DOE has agreed[10] to address this concern.

<u>Rock-Mass Mechanical Properties: Effects of Lithophysae</u>

The values of rock-mass mechanical properties for lithophysal and nonlithophysal rock units were determined using empirical correlations between such properties and the rock-mass quality indices, such as the Q index of Barton, et al. (1974) or the RMR (Rock Mass Rating) index of Bieniawski (1979). These quality indices were developed to account for the effects of fractures on the mechanical characteristics of a rock mass. The use of the Q and RMR indexes to account for the effects of lithophysae (CRWMS M&O, 2000k) is unprecedented and not supported by any data on or model investigation of the effects of lithophysae on the mechanical characteristics of rock.

The values of the Q and RMR indexes are determined through an accumulation of a set of categorical variables that are assigned values to represent aspects of the mechanical attributes of fractures. For example,

$$Q = (RQD/J_n) \times (J_r/J_a) \times (J_w/SRF) \qquad (2.1.7\text{-}1)$$

where RQD is the rock quality designation, J_n is the joint-set number, J_r and J_a represent joint roughness and alteration, and J_w and SRF are factors used to represent water pressure and rock stress (Barton, et al., 1974). The ratio (J_w/SRF) is set to one if Q is used to determine

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

parameter values for stress analyses (instead of being used directly to design ground support) because the effects of water pressure and rock stress can be accounted for directly in such analyses. Each of the parameters used to calculate Q is assigned a value from tables compiled by the original developers of the technique (Barton, et al., 1974). Generally, the ratio (RQD/J_n) represents the unfractured-rock block size, (J_r/J_a) represents the strength of the joint (or fracture) surfaces, and (J_w/SRF) represents the stress state. It is conceivable that the lithophysal content of a rock may be correlated somewhat with the RQD value, but none of the other parameters can be readily correlated to the mechanical attributes of lithophysae.

Therefore, using the Q index to characterize the effects of lithophysae on the mechanical characteristics of a rock mass is tantamount to assuming the RQD alone is sufficient as a mechanical-behavior index. This assumption was rejected several decades ago (e.g., consider the histories of the Q and RMR indexes). Therefore, there is currently inadequate technical basis to support the use of either Q or RMR to characterize the mechanical behavior of the lithophysal tuff. Although these indices may be appropriate for accounting for the effects of fractures, some modification of their values would be necessary if DOE uses the indexes to account for the effects of lithophysae. The technical basis for such modification is all the more important because about 75 percent of the proposed emplacement area may lie within the lithophysal rock units. To address these insufficiencies by license application, DOE agreed at the DOE and NRC Technical Exchange on Repository Design and Thermal-Mechanical Effects[11] to provide additional documentation.

Rock-Mass Mechanical Properties: Effects of Fractures

The DOE approach to mechanical characterization of Yucca Mountain is to determine the values of mechanical properties using empirical correlations between the properties and the rock-mass quality indexes, such as Q and RMR. Two sets of Q and RMR values were determined along the Exploratory Studies Facility main drift and North and South Ramps based on a scan-line survey and a full-periphery map of the tunnel (CRWMS M&O, 1997a, Figures 39 and 40). The rock mass was classified into five quality categories: RMQ1, RMQ2, RMQ3, RMQ4, and RMQ5 (with RMQ1 associated with the smallest Q value and RMQ5 the greatest), based on the frequency distribution of Q and RMR values determined from the Exploratory Studies Facility and augmented with data from borehole logs (CRWMS M&O, 1997b). The range of Q and RMR values associated with each quality category is different for each of the stratigraphic units that comprise the repository host rock [i.e., the middle nonlithophysal, lower lithophysal, and lower nonlithophysal units of the Topopah Spring Welded Tuff (CRWMS M&O, 2000m, Figure 5)]. It is expected that approximately 75 percent of the repository block would lie within the lower lithophysal unit, but the part of the Exploratory Studies Facility that intersects the repository host rock lies mainly within the middle nonlithophysal unit. A second exploratory drift, the cross-block drift, was excavated to obtain more data for the lower lithophysal unit. Although the fracture data from the cross-block drift have been reported (CRWMS M&O, 2000n), the resulting Q and RMR data have not been

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

compiled in any DOE report known to the NRC staff. The available Q and RMR data have been combined with intact rock data from laboratory testing (CRWMS M&O, 1997b) to determine the values of rock-mass mechanical properties using empirical relationships from the literature (CRWMS M&O, 1997a).

This DOE approach to mechanical characterization is generally consistent with the current methods of accounting for the effects of fractures on the mechanical characteristics of rock masses (e.g., Barton, et al., 1974; Bieniawski, 1979; Hoek and Brown, 1997). There are, however, two concerns about the DOE implementation of the approach: (i) DOE uses empirical relationships (between rock-mass quality indices and mechanical properties) from the literature without sufficient site-specific data to verify the applicability of the relationships to the site and, hence, to determine the uncertainties associated with using such relationships; and (ii) DOE has not presented sufficient information to permit an independent assessment of the appropriateness of the intact rock data used in conjunction with the rock-mass quality indices to evaluate the rock-mass mechanical properties. To address these concerns by license application, DOE agreed at the DOE and NRC Technical Exchange on Repository Design and Thermal-Mechanical Effects[12] to provide additional documentation. These concerns are best illustrated through a discussion of the specific rock-mass mechanical properties, as in the following.

<u>Rock-Mass Young's Modulus, E_m:</u> DOE determined values of E_m using two empirical relationships from the literature (Serafim and Pereira, 1983; Palmstrom, 1996) and examined the sensitivity of the calculated E_m to the scan-line or full-periphery data and to different methods of interpreting the Q and RMR values for the empirical relationships (CRWMS M&O, 1997a). The results show E_m values for Topopah Spring Welded Tuff rocks in the range 8.98–14.62 GPa [1,302.5–2,120.5 ksi] for the RMQ1 and 24.46–45.08 GPa [3,547.7–6,538.4 ksi] for RMQ5. DOE concluded (CRWMS M&O, 1997a, p.74), based on the variability of these results, that "*In-situ* field testing from several spatially correlated intervals within each thermomechanical unit in the Exploratory Studies Facility Main Loop is recommended to validate the range of empirically based rock mass modulus estimates." In March 1997, DOE expressed a similar conclusion (CRWMS M&O, 1997b, Table 2-16) that the information available on rock-mass stiffness would not satisfy the DOE standard for either the viability assessment or license application. The site-specific E_m data collected by DOE to date (based on information known by NRC staff) consist of six data points from Exploratory Studies Facility convergence analyses and one data point each from plate-loading and Goodman-Jack tests. As argued earlier (NRC, 2000a), these data are too sparse [in its coverage within the E_m-versus-Q (or RMR) space] to provide a reliable estimate of the uncertainties associated with using the empirical relationships from the literature. E_m is important because the induced thermal stress is directly proportional to the rock-mass stiffness. Consequently, the induced thermal stress can be known no better than the uncertainty in the rock-mass stiffness. Therefore, the predicted performance of underground openings under thermal-loading conditions is at best as uncertain as the knowledge of the rock-mass stiffness.

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

Rock-Mass Strength: DOE determined the values of rock-mass strength parameters for implementing the Mohr-Coulomb strength criterion (friction angle, Φ_m, and cohesion intercept, c_m) using an empirical approach developed by Hoek and Brown (1997). The Hoek-Brown approach consists of using the Hoek-Brown failure criterion (Hoek and Brown, 1980, 1997) to calculate sets of σ_1-versus-σ_3 values (where σ_1 and σ_3 are the maximum and minimum principal compressive stresses) to define the failure envelope for a rock mass and fitting a straight line to the results to determine Φ_m and c_m. Hoek and Brown (1997) indicated that the values of Φ_m and c_m determined using this approach are sensitive to the range of σ_3 values and the values of the intact-rock parameters—unconfined compressive strength, σ_{ci}, and Hoek-Brown parameter, m_i—used to generate the failure envelope. The intact rock parameters σ_{ci} and m_i should be evaluated using statistical analyses of laboratory triaxial-test results obtained with values of σ_3 in the range $0 < \sigma_3 < 0.5\sigma_{ci}$ (Hoek and Brown, 1997).

The DOE implementation of the Hoek-Brown approach using Topopah Spring Welded Tuff data from the Exploratory Studies Facility gave Φ_m = 56–57° and c_m = 1.9–2.6 MPa [0.276–0.377 ksi] for the RMQ1 rock-mass category and Φ_m = 58° and c_m = 3.9–6.6 MPa [0.566–0.957 ksi] for RMQ5, based on straight-line fits to the strength envelope for σ_3 values in the range $0 \leq \sigma_3 \leq 3$ MPa [0–0.44 ksi] (CRWMS M&O, 1997a). A revision of the calculation using strength envelopes in the range $0 \leq \sigma_3 \leq 42$ MPa [0–6.1 ksi] (CRWMS M&O, 2000k) gave Φ_m = 37° and c_m = 8 MPa [1.2 ksi] for RMQ1, and Φ_m = 42–43° and c_m = 12–13 MPa [1.7–1.9 ksi] for RMQ5. The two sets of strength parameters [i.e., the original set from CRWMS M&O (1997a) and the revised set from CRWMS M&O (2000k)] are given in CRWMS M&O (2000k, Tables 4-5a and 4-5b), but the original set was used for continuum analyses of the stability of the emplacement drifts. The five sets of continuum thermal-mechanical analyses presented in CRWMS M&O (2000k, Figures 6-22, 6-23, and 6-27) were based on Φ_m = 56° and c_m = 2 MPa [0.3 ksi] for RMQ1 and Φ_m = 58° and c_m = 4.1 MPa [0.6 ksi] for RMQ5. One analysis was presented based on Φ_m = 37° and c_m = 2 MPa [0.3 ksi] for RMQ1 (CRWMS M&O, 2000k, Figure 6-29), and the failure zone predicted from this analysis (for an unsupported opening) extended into the rock mass from the drift wall approximately 2.5 times as much as the failure zone predicted using Φ_m = 56° and c_m = 2 MPa [0.3 ksi].

The friction angle values suggested in the original strength-parameter set are significantly larger than the values commonly encountered in the literature. For example, an implementation of the Hoek-Brown approach in Hoek and Brown (1997) using Q = 0.53 for RMQ1 and Q = 12 for RMQ5 [based on CRWMS M&O (2000k)] would give Φ_m = 23–40°for RMQ1 and Φ_m = 27–47° for RMQ5, for m_i values in the 5–35 range. The Hoek-Brown implementation of the approach (Hoek and Brown, 1997, Figure 8) suggests a maximum Φ_m value of approximately 52 degrees for a rock mass with Q = 166, which is at least one order of magnitude greater than the Q values of approximately 0.5–15 for the repository host rock mass. The Hoek-Brown implementation would, therefore, imply much smaller values of Φ_m for the repository rock mass than the values of Φ_m suggested in CRWMS M&O (1997a).

DOE addressed this concern by providing the revised strength-parameter set (CRWMS M&O, 2000k) based on an application of the Hoek-Brown approach (Hoek and Brown, 1997) using a broader range of confining pressure than the range used to obtain the original strength-parameter set (CRWMS M&O, 1997a). The use of a broader range of confining pressure, however, addresses only one of the staff concerns regarding the

CRWMS M&O (1997a) strength-parameter set. There are still unresolved concerns that can potentially affect the values of the rock-mass strength parameters. First, the value of m_i used for the calculations was specified as 20 [based on CRWMS M&O (1997b)], but the laboratory data used to evaluate m_i or the range of the m_i values were not provided. Second, the value of σ_{ci} was based on conventional unconfined compression test data without any adjustments to account for the effects of sustained loading (infinitely slow loading rates) at the site. The relationship between the unconfined compressive strength of intact rock under fast loading (conventional loading rates used for laboratory testing) and sustained loading (slow loading rates that occur *in situ*) is well documented in the literature (e.g., Lajtai and Schmidtke, 1986; Martin and Chandler, 1994). The effect of the relationship is that only approximately 50 percent of the laboratory intact-rock strength is applicable to site conditions, considering the loading-rate effects only. DOE uses 100 percent of the laboratory σ_{ci} value and has not presented the technical basis for doing so.

<u>Rock-Mass Thermal Expansivity, α_m:</u> The thermal-mechanical analyses for site recommendation (CRWMS M&O, 2000k) were conducted using average intact-rock thermal-expansivity for the repository-level stratigraphic units, based on laboratory data from CRWMS M&O (1997b). DOE argued that the use of intact-rock thermal expansivity, instead of rock-mass expansivity, would be adequate for assessing the stability of underground openings because the intact-rock expansivity would result in greater-than-anticipated stresses. The NRC staff agree that the intact-rock thermal expansivity would give upper-bound estimates of the anticipated thermal expansion of the rock mass at a given location, but using an average thermal expansivity for the different stratigraphic units may result in a misleading assessment of the stability of the emplacement drifts. Because the stratigraphic interfaces are approximately horizontal, the differences in thermal expansivity between the stratigraphic units will likely increase the thermally induced shear stress on the subhorizontal fractures. Because slip on the subhorizontal fractures is potentially the dominant rock-failure mechanism in the emplacement area, the features of the environment that may affect the magnitudes of shear stress on the subhorizontal fractures deserve specific attention. DOE stated (CRWMS M&O, 2000k) that the differences between the intact-rock expansivity for the different stratigraphic units (CRWMS M&O, 1997b, Table 5-15) are not significant. The differences may be significant, however, because of their potential effect on slip on the subhorizontal fractures; therefore, DOE should develop sufficient technical information to evaluate the significance.

<u>Rock-Mass Thermal Properties</u>

DOE uses intact-rock thermal properties (thermal conductivity, specific heat capacity, and density) to characterize the rock-mass thermal behavior (CRWMS M&O, 2000k). As discussed (NRC, 2000a), the NRC staff agree that the thermal response of a rock mass (evolution of temperature distributions around a buried heat source in the rock mass) can be assessed satisfactorily using the intact-rock thermal properties.

<u>Fracture-Surface Mechanical Properties</u>

The fracture-surface mechanical properties, which are used for discontinuum modeling, are the stiffness parameters (shear and normal stiffness), the strength parameters (friction angle and cohesion), and the postfailure dilation parameter. DOE reported fracture-surface mechanical properties from two sources. First, CRWMS M&O (1997b) gives data from laboratory testing of core specimens. The data consist of normal stiffness of approximately 74 MPa/mm [271 ksi/in] from 11 Topopah Spring Welded Tuff core specimens tested with a normal stress of 2.5 MPa, [0.36 ksi] and a friction angle of approximately 41° from 12 Topopah Spring Welded Tuff core specimens (5 lower nonlithophysal, 5 lower lithophysal, and 2 middle nonlithophysal). Second, friction angles in the range 60–64 degrees were determined for Topopah Spring Welded Tuff fracture surfaces based on an interpretation of Exploratory Studies Facility fracture data. The interpretation, however, included an incorrect assumption that the residual friction angle of fractures is equal to the rock-mass friction angle (CRWMS M&O, 1997a, Section 7.3), which provides a possible explanation for the unusually high values of fracture friction angle from the Exploratory Studies Facility data.

The laboratory fracture data (CRWMS M&O, 1997b, Tables 5-39 and 5-40) are potentially useful, but DOE needs to determine if the data are representative of the site and provide the associated technical bases. Furthermore, no information has been provided about the fracture shear stiffness, dilation, or variation of shear or normal stiffness with normal stress.

Spatial and Temporal Variations of Mechanical Properties

Rock-mass mechanical properties vary both vertically and laterally at Yucca Mountain because of the site stratigraphy and variations in the mechanical properties of intact rock and fractures, other fracture properties (such as frequency, spacing, and continuity), and lithophysae content. The mechanical properties may also vary with time because of potential changes resulting from coupled thermal-hydrological-chemical-mechanical processes.

Spatial Variation of Mechanical Properties: DOE (CRWMS M&O, 2000k) stated that using the mechanical properties for the RMQ1 and RMQ5 rock-mass categories in thermal-mechanical analyses adequately represented the spatial variation of mechanical properties at Yucca Mountain because these two rock-mass categories envelop the worst and best expected rock conditions at the site. To support this argument, DOE needs to demonstrate the validity of two premises: (i) that the range of rock-mass quality determined from the Exploratory Studies Facility and, possibly, the cross-block drift, envelops the qualities within the repository block; and (ii) that the quality classification based on the Q and RMR indices, which were developed to account for the effects of fractures, is applicable to the lower lithophysal rock unit, in which lithophysae are expected to contribute significantly to the mechanical behavior.

Time-Dependent Degradation of Mechanical Properties: Time-dependent degradation of the repository host rock was not discussed in the DOE thermal-mechanical analyses for site recommendation (CRWMS M&O, 2000k), but is potentially important because an operational life up to 175 years with possible extension to 300 years may be expected for the ground-support system (CRWMS M&O, 2000d). A DOE expert panel on drift stability (Brekke, et al., 1999) indicated that degradation of the rock mass can be expected because of coupled thermal-hydrological-mechanical processes operating over a long period of time. Thermal, water-pressure, and rock-stress gradients that occur in the rock mass after the emplacement of nuclear waste would drive processes such as thermally induced fracture propagation, rock loosening, and cyclical evaporation and condensation of water. Such processes can be expected to cause degradation of the rock mass.

Rock-mass degradation related to the geochemical response of the system to elevated temperature can also be expected. Heat generated from nuclear waste is expected to cause a geochemical response because mineral stabilities and equilibria depend on temperature; geochemical reaction rates in the presence of water would accelerate at elevated temperature; and the thermal gradients would cause redistribution of moisture, solutes, and carbon dioxide, which are essential to the chemical reactions (Murphy, 1993). Reaction-path modeling of the natural gas-water-rock geochemical system at Yucca Mountain (Murphy, 1993) indicates that the anticipated geochemical reactions include dissolution of feldspars; precipitation of secondary minerals, such as clinoptilolite, smectite, and calcite; and increase in pH and aqueous sodium bicarbonate concentrations. Although the repository-induced mineralogical changes are likely to affect only a small rock volume, the changes are expected to be localized at fluid-rock interfaces such as fracture walls and lithophysal cavities. Consequently, the alteration minerals would be expected to develop as lithophysal-cavity deposits or fracture coatings.

Mineral-alteration products currently occur at Yucca Mountain mostly as fracture coating and as lithophysal-cavity deposits (Carlos, et al., 1995). The mineralogy, thickness, and amount and uniformity of coverage of fracture coatings are highly variable and uncertain (Thoma, et al., 1992). The coatings consist mainly of zeolites, manganese oxide minerals, silica phases, carbonates (mostly calcite), and clay minerals (mostly smectite but occasionally illite). Smectite is fairly ubiquitous in fractures throughout the volcanic sequence (Carlos, et al., 1995). The genesis of the fracture coatings at Yucca Mountain is not well understood, but the coatings are generally secondary minerals formed as alteration products of primary minerals such as glass, feldspar, and silica phases (Murphy, 1993; Carlos, et al., 1995; Levy, et al., 1996).

If the fracture coatings that develop after waste emplacement consist dominantly of quartz and other silica phases (e.g., Lin and Daily, 1984; Daily, et al., 1987; Matyskiela, 1997), the shear strength of fractures and, therefore, the rock-mass strength can be expected to increase. If fracture coatings consist mainly of secondary minerals, such as smectite and calcite that are mechanically weaker than the primary minerals (Kenney, 1967; Mitchell, 1976), a weakening of the fractures and, therefore, the rock mass can be expected. The secondary minerals would develop either as fracture-wall precipitates from aqueous solutions or in-place alteration products of fracture-wall rock. The result would be a change in the mechanical characteristics of fractures within the affected zone from their current classification as generally "rough, irregular, and tightly healed" to a mechanically weaker category of generally "wide and filled with clay minerals (or other alteration products) thick enough to prevent wall-rock contact" (Barton, et al., 1974).

The magnitudes, rates, and spatial distributions of the anticipated degradation of the repository host rock will be difficult, if at all possible, to evaluate. However, degradation of the host rock can reasonably be expected (Brekke, et al., 1999), and it can produce a significant impact on the stability of the emplacement drifts (NRC, 2000a). Therefore, degradation of the host rock should be accounted for in assessing the performance of the subsurface structures, systems, and components and adequate technical basis provided to support the approach used to account for it.

Uncertainties in Mechanical Properties

Mechanical-property uncertainties were not discussed in the DOE analyses of ground-support performance for site recommendation (CRWMS M&O, 2000k). CRWMS M&O (1997b), Table 2-9, for example, indicates a mean value of 104 MPa [15.1 ksi] with a standard deviation of 61 MPa [8.8 ksi] for the unconfined compressive strength of the lower lithophysal intact rock, but this uncertainty in the intact-rock strength is not reflected in the ground-support design analyses (CRWMS M&O, 2000k). As discussed earlier, there are considerable uncertainties in all the mechanical properties needed for design analyses. The influence of such uncertainties on the assessment of the performance of the subsurface structures, systems, and components should be clearly identified, and the identification should be supported with adequate technical basis.

As previously discussed, DOE agreed to address these NRC concerns regarding specific rock-mass mechanical properties during the DOE and NRC Technical Exchange on Repository Design and Thermal-Mechanical Effects.[13]

Subsurface Ground-Support Systems Design

There is currently no outstanding NRC staff concern about design methodology. NRC has accepted the DOE proposed design methodology in DOE (1997). There are, however, several concerns with the DOE implementation of the design methodology as discussed previously in this Subsection.

Subsurface Ventilation System Design

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

Subsurface Power and Power Distribution Systems Design

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

Maintenance Plan for Subsurface Facility Design

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

2.1.7.3.3.3 Waste Package and Engineered Barrier Subsystem Design

Engineered Barrier Subsystem and Controls Are Adequately Designed

The acceptance criterion for waste package and engineered barrier subsystem structures, systems, and components and their controls addresses the need to prevent waste form

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

degradation and provide containment, criticality control, shielding, and thermal control of the high-level waste during the preclosure period. In addition to the waste package, other engineered barrier subsystem structures, systems, and components that may be used to achieve these requirements include, but are not limited to, drip shields, waste package pallet supports and invert, backfill, and sorption barriers. To demonstrate that this acceptance criterion has been satisfied, DOE must provide a description and assessment of the components for the various types of waste packages including containers and internal structures. This information must also be provided for other relevant important to safety engineered barrier subsystem components (e.g., drip shield, waste package supports and invert, and such).

Specific information expected from DOE includes the following: (i) identification of the materials, methods, and processes used in the fabrication of containers, internal waste package components, and engineered barrier subsystem components (must be consistent with accepted design criteria, codes, standards, and specifications); (ii) specifications for container and internal waste package materials that are in agreement with those established in the final design (including consideration of the specifications for the closure welding, preparation for welding, materials to be used in the welds, and inspection of the welds that comply with applicable American Society of Mechanical Engineers codes); (iii) basis for nondestructive examination methods used to detect and evaluate defects that may lead to premature failure of the fabricated containers and other structural components of the waste packages; (iv) criticality design criteria consistent with those used in model calculations that support the design; (v) analyses demonstrating that the shielding provided by the containers is sufficient (including estimates of dose rates, a description of the source of data for the evaluation and the methods for estimating dose rate, and identification of the computational codes used); (vi) analyses demonstrating that the components of the waste package and internals are designed to sustain loads from normal operation and Categories 1 and 2 event sequences; (vii) analyses demonstrating that thermal control is such that the fuel cladding temperature will be sufficiently low to prevent cladding failure; (viii) evidence the materials used in construction of the internal components of the waste package are compatible with the waste form; (ix) analyses demonstrating the design of any drip shield, including materials of construction, configuration, and method of emplacement, is sufficient to prevent water from contacting the waste packages and does not impair safe handling of the waste package during subsurface maintenance operations; (x) analyses demonstrating that the design of any backfill, including materials and physical characteristics, configuration, and methods of emplacement and compaction, is adequate to reduce the relative humidity near the waste packages; and (xi) analyses demonstrating that the design of any sorption barrier is adequate to control the migration of radionuclides and materials. The postclosure performance of the engineered barrier subsystem is addressed in Sections 3.3.1 and 3.3.2.

Overall, the current information, along with the information to be provided according to the agreements reached between DOE and NRC in the Container Life and Source Term,[14]

[14]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Container Life and Source Term (September 12–13, 2000)." Letter (October 4) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

Repository Design and Thermal-Mechanical Effects,[15] Preclosure Safety,[16] and Range of Operating Temperatures[17] Technical Exchanges, is sufficient to conclude that the necessary information needed to assess the design of the waste package and engineered barrier subsystem structures, systems, and components and safety controls will be available at the time of a potential license application. The designs of the waste package, drip shields, and the waste package pallet have yet to be finalized. In addition, the fabrication, remediation, and waste package and drip shield emplacement methods are currently being developed.

Waste Package Design Description

The current waste package design consists of two concentric cylinders (i.e., disposal containers, fabricated from plate material). The inner disposal container will be fabricated using Type 316 nuclear grade stainless steel that is a minimum of 50 mm [1.97 in]-thick (CRWMS M&O, 2001a). The inner disposal container will fit inside the outer disposal container that is constructed from 20-mm [0.79-in]-thick Alloy 22. A radial gap of 0 to 4 mm [0 to 0.16 in] will be used between the inner and outer disposal containers to allow for differential thermal expansion to occur without introducing thermally induced stresses. The axial gap between the inner and outer disposal containers, which may be more important as far as differential thermal expansion stresses are concerned, is 10 mm [0.39 in] (CRWMS M&O, 2000e). Type 316 nuclear grade stainless steel was selected for the inner disposal container to provide mechanical integrity to the waste package during both the preclosure and postclosure periods of the proposed repository. The selection of Alloy 22 as the outer disposal container material was based on the resistance of this nickel-chromium-molybdenum-tungsten alloy to both localized corrosion and stress corrosion cracking in chloride-containing environments. Placement of the corrosion-resistant Alloy 22 container on the outside of the Type 316 nuclear grade stainless steel is designed to provide long-term protection of the inner container material (CRWMS M&O, 2000f).

There are several waste package configurations for the site recommendation waste package design needed to encapsulate the various commercial spent nuclear fuel waste forms (CRWMS M&O, 2000f). These configurations include designs for pressurized water reactor fuel containing either 12 or 21 pressurized water reactor assemblies with absorber plates and 21 pressurized water reactor assemblies with control rods. Two waste package configurations are required for boiling water reactor fuel that contains either 44 boiling water fuel assemblies with absorber plates or 24 boiling water reactor fuel assemblies with thick absorber plates.

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

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

[17]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Range of Operating Temperatures." Letter (October 2) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

Moreover, there are additional waste package configurations for the disposal of defense high-level waste and DOE-owned spent nuclear fuel.

The waste package will be constructed by rolling the plate materials into cylinders. A longitudinal weld will be used to complete the cylinder. Welding will also be used to connect two cylinders together to provide sufficient length for the spent nuclear fuel and high-level waste. The bottom lids of the disposal containers are also welded in place. Although the Type 316 nuclear grade stainless steel inner disposal container provides mechanical integrity to the waste package, the Alloy 22 outer disposal container will be required to sustain loads during lifting and transport. Lifting trunnions will be attached to the outer surface of the Alloy 22 disposal container to facilitate the necessary lifting and transport operations. The design of the inner disposal container will be specific to the waste package contents. Unique internal support structures are required for pressurized water reactor fuel, boiling water reactor fuel, and high-level waste glass (CRWMS M&O, 2000f). After the internal support structure is constructed inside the inner disposal container, the inner Type 316 nuclear grade stainless steel container will be inserted into the Alloy 22 outer disposal container. After the loading of the disposal containers, the containers will be sealed with lids that are welded in place. One lid is used for the Type 316 nuclear grade stainless steel, and a dual-closure lid design is used for the Alloy 22 outer disposal container (CRWMS M&O, 2000e).

In summary, the waste package design description appears to incorporate design features for containment. The design of the waste package is still under development, so DOE will provide additional design information in future documents. These documents will be reviewed as they become available.

<u>Waste Package Internal Components Design Description</u>

Internal components of the waste packages include basket guides, corner guides, fuel tubes, and defense high-level waste canister guides (CRWMS M&O, 2000f). The internal components are designed to facilitate heat transfer from the interior of the waste package to the exterior surface of the outer disposal container, by way of thermal conduction, to keep fuel cladding temperatures within specified limits, control criticality, and provide structural support to the waste package. In addition, the materials used in the waste packages must be compatible with the waste form, spent nuclear fuel cladding, and the waste package disposal container materials. The materials should not be reactive or pyrophoric.

The design of the waste packages for commercial spent nuclear fuel also contains stainless steel boron alloy plates (absorber plates) to provide criticality control. When criticality control is provided by the spent nuclear fuel control rods, the absorber plates are replaced with carbon steel plates to provide structural support and maintain the desired geometric configuration. The internal structure must maintain the desired geometric configuration when subjected to mechanical loads to provide criticality protection during handling, emplacement, and retrieval (CRWMS M&O, 2000f). In addition, the material used to provide criticality control must be compatible with the other materials and components inside the waste package and must not degrade the waste form. DOE identified Neutronit A978, which is similar in composition to Type 316L stainless steel with 1.6 percent boron added, as the material that will be used for the absorber plates.

The DOE description of the internal components of the waste package includes the necessary components for configuring the waste, providing criticality control, and transferring heat necessary to keep the internal temperature of the waste packages below design limits (see the appropriate topical discussions provided in this section for additional details pertaining to criticality design criteria and fuel cladding temperature control). The design of the waste package is still being developed, so DOE will provide additional design information in future documents. These documents will be reviewed as they become available.

Drip Shield Design Description

The description of the drip shield, its fabrication sequence, and the emplacement methods are not complete. The design of the drip shield is still under development (CRWMS M&O, 2001a). The current drip shield design calls for a Titanium Grade 24 support structure covered with 15-mm [0.59-in]-thick Titanium Grade 7 plate. Individual segments of the drip shield are connected together using a vertically sliding interlock configuration. The drip shield will be installed at the end of the preclosure period. The intended function of the drip shield is to divert any dripping water from contacting the waste packages and protect the waste package against rockfall and drift collapse in the postclosure period (CRWMS M&O, 2001b). Emplacement of the drip shields at earlier times would prevent the inspection of the waste packages when using remotely controlled inspection gantries (CRWMS M&O, 2000p).

DOE has provided a conceptual design description for the drip shield, including the materials of construction, configuration, and method of emplacement. Details of the fabrication methods have yet to be provided, however. An assessment of the ability of the proposed drip shield to withstand mechanically disruptive events for the postclosure period is provided in Section 3.3.2.4.4.1. Even though all potential postclosure design basis events are not applicable to the preclosure period, the comments pertaining to the general analysis methodology used by DOE to demonstrate the structural integrity of the drip shield are relevant to the preclosure safety case. DOE will provide additional design information in future documents. These documents will be reviewed as they become available.

Waste Package Pallet

The waste package pallet is designed using Alloy 22 plate material (CRWMS M&O, 2000g). Each waste package pallet has two V-shaped supports that are connected together using stainless steel rails. Two sizes of emplacement pallets will be required to accommodate the different waste package lengths.

DOE performed structural evaluations of the emplacement pallet corresponding to static loading by the waste package and lifting during handling operations (CRWMS M&O, 2000q,r). The results of analyses used to support these structural evaluations are reported using stress intensity values. Because no clear definition of stress intensity was provided, however, it has been assumed that the reported values of stress intensity are consistent with the definition provided in American Society of Mechanical Engineers (2001, Subparagraph NB–3213.1). In addition, it is not clear if the normal stress components generated at the contact interface between the waste package and pallet were taken into consideration when calculating the stress intensity results presented in the reports. Seismic loads were not addressed in the lifting of a loaded pallet structural evaluation. DOE must either assess the effects of seismic loads on a loaded pallet for all relevant handling operations or justify their exclusion. Similarly, DOE must assess the potential consequences of dropping a loaded emplacement pallet or provide the basis for excluding this particular event from consideration.

Disposal Container Fabrication and Closure

The disposal container will be fabricated according to American Society of Mechanical Engineers (1995a, Section III, Division 1, Subsection NB, Class 1 Rules for Construction of Nuclear Power Plant Components) to the maximum extent practicable (CRWMS M&O, 2001a). Deviations from the code will be documented and submitted for approval, but the disposal containers will not be nuclear or "N"-stamped pressure vessels (CRWMS M&O, 2001a). American Society of Mechanical Engineers (1995a) provides a standard for the fabrication of the disposal containers and requirements for inspection.

American Society of Mechanical Engineers (2001) provides rules for construction with the objective of protecting life and property, and a margin for deterioration in service, to assure a safe period of usefulness for boilers, pressure vessels, and nuclear components. The official American Society of Mechanical Engineers Boiler and Pressure Vessel Code symbol stamp may only be used to identify components constructed in accordance with the applicable rules of the code, which include requirements for materials, design, fabrication, examination, and inspection. Items not constructed in accordance with rules of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code may not be stamped, and such items may not meet the objectives of the code. DOE stated that the materials used in the fabrication of the disposal containers and the drip shield will meet the requirements in American Society of Mechanical Engineers (1995a, Section III, Division 1, Article NB–2000).

Filler materials used in welding processes must conform to the requirements specified in American Society of Mechanical Engineers (1995b, Section II, Part C). For the Type 316 nuclear grade stainless steel inner container, the filler material will be selected to control the delta ferrite content of the as-deposited weld metal. A ferrite number between 5 and 15, determined by Magna-gage measurements, is required in the inner disposal container fabrication welds (CRWMS M&O, 2001a). The weld filler material for the Alloy 22 outer container will be ENiCrMo-10 or a filler material used for welding alloys with the UNS (Unified Numbering System) number N06022 designation (CRWMS M&O, 2001a).

The preparation of the disposal containers and the procedures for welding will be in accordance with American Society of Mechanical Engineers requirements (1995c, Section IX). Welding will not be performed if the temperature of the base metal is lower than 0 °C [32 °F]. The maximum interpass temperature for austenitic stainless steels (including Type 316 nuclear grade stainless steel) and nickel alloys (including Alloy 22) is 175 °C [347 °F]. Each weld layer is required to be free of slag, inclusions, cracks, unacceptable porosity, and lack of fusion. Welding processes for the fabrication of the disposal containers may include shielded metal arc, gas tungsten arc, submerged arc, and gas metal arc, provided the processes are qualified (CRWMS M&O, 2001a).

Defects in the disposal container can be repaired by welding provided that the requirements in American Society of Mechanical Engineers (1995a, Section III) are met. All material defects and repairs must be appropriately documented (CRWMS M&O, 2001a). Weld repairs will be performed in accordance with American Society of Mechanical Engineers requirements (1995a, Section III, Division I, Article NB–4000). Only three repair cycles will be permitted without special approval (CRWMS M&O, 2001a). DOE did not provide any rationale or basis for this specification.

Fabrication of both the inner and outer disposal containers involves cutting, rolling, and welding operations. Fabrication of the cylinders that form the sides of the disposal containers is similar for both the inner and outer containers. After the plates are inspected, they are cut to form the cylinders and lids. The plates are then rolled into cylinders. The dimensions of the cylinders are adjusted to assure the final design dimensions can be achieved and to minimize distortion from welding. The longitudinal seam is then welded, and the completed weld is inspected. After the ends of the cylinders have been satisfactorily prepared, the two cylinders are welded together. A dimensional inspection is then performed, and if needed, the cylinder is machined to tolerance.

The remaining fabrication steps for the disposal containers are specific to the inner and outer containers. For the Type 316 nuclear grade stainless steel inner container, the bottom lid and the internal parts, such as baskets, corner guides, and separator plates, are installed. For the Alloy 22 outer container, an assembly support ring used to support the Type 316 nuclear grade stainless steel inner containers is welded into place, and the welds are machined to allow the inner cylinder to be properly installed into the outer container. The bottom lid is then fit and welded in place. The trunnion collar sleeve is then installed on the outside of the Alloy 22 outer container and welded in place. Solution annealing is performed at approximately 1,125 °C [2,057 °F] to eliminate residual stresses created during the fabrication processes. The solution annealing should also dissolve any secondary phase precipitates such as topologically close packed phases formed as a consequence of the welding processes. The Alloy 22 outer container is annealed in a furnace on a furnace car. The furnace car is used to transport the disposal container out of the oven where it is sprayed with water on both the inside and outside surfaces. The water quench is designed to reduce the temperature of the Alloy 22 outer container from 1,150 °C [2,102 °F] to below 800 °C [1,472 °F] in approximately 4 minutes. The cooling rate is then decreased to allow for the formation of compressive stresses.

For the inner Type 316 nuclear grade stainless steel container, the closure lid and shear rings are installed, and a seal weld is used to hold the shear rings in place using the gas metal arc weld method, which allows faster deposition rates (Stephenson, 1990). The evaluation of an Alloy 22 closure lid welding method has recently been reported (CRWMS M&O, 2001b). Welding methods considered were narrow groove gas tungsten arc welding, optimized gas tungsten arc welding, and plasma arc welding. The selection criteria considered, in decreasing weight of importance, were process recovery, residual stresses, equipment reliability, production rate, fit-up tolerances, remote operation capability, radiation hardening, and industrial experience. Plasma arc welding was rated the best for residual stresses and production rates. Optimized gas tungsten arc welding was rated the best for radiation hardening considerations. For all other selection criteria, the narrow groove gas tungsten arc welding method was determined to be the best method for the Alloy 22 closure lids.

To reduce residual stresses in the Alloy 22 final closure welds, laser peening is used on the inner Alloy 22 closure lid weld. Details of the process have not been reported. For the outer closure lid, local induction annealing of the extended outer shell is proposed as a method to eliminate residual tensile stresses in the Alloy 22 outer closure weld. Although the process is under development, the proposed induction annealing process would be used to heat the end of the Alloy 22 disposal container with the completed closure weld to a temperature of 1,150 °C [2,012 °F]. Forced air or water will be used to rapidly reduce the temperature of the closure weld region (CRWMS M&O, 2001b). Because the process is still under development, specifications for cooling times and temperature distributions have not been established.

The combination of cold work used in forming and machining operations and elevated temperature exposures as a result of welding and annealing processes may result in the precipitation of topologically close packed phases. During the solidification of the weld metal, molybdenum and tungsten segregate to the interdendritic regions leaving the dendrite core rich in nickel (Cieslak, et al., 1986a,b). The depletion of nickel and enrichment of molybdenum and tungsten in the interdendritic regions promote the precipitation of topologically close packed phases. The composition of all the topologically close packed phases, including σ, μ, and P phases, can contain more than 30-percent molybdenum (Raghavan, et al., 1984). The high concentration of molybdenum in these phases results in a depletion of molybdenum adjacent to the precipitates that reduces the resistance of the alloy to localized corrosion. Because the formation of the precipitates preferentially occurs in the weld regions and in the intergranular regions of the heat-affected zone adjacent to the welds, localized corrosion in the form of interdendritic and intergranular corrosion may be a consequence of the precipitation of topologically close packed phases (Heubner, et al., 1989). The ductility of σ, μ, and P phases is typically low compared with the austenitic matrix of the nickel-base alloy (Matthews, 1976; Tawancy, 1996). As a result, the precipitation of topologically close packed phases may reduce the ductility and impact strength of the alloy, particularly in welds or in the heat-affected zones of the welds.

The thermal stability of nickel-chromium-molybdenum alloys was evaluated using several criteria: (i) microstructural examination for the presence of secondary phase precipitates at the grain boundaries or in the interdendritic regions of welds; (ii) intergranular corrosion susceptibility; and (iii) mechanical properties such as ductility, yield strength, or impact toughness. Heubner, et al. (1989) provided a phase stability diagram for Alloy 22, based on microstructural examinations conducted after isothermal exposures at temperatures ranging from 550 to 900 °C [1,022 to 1,652 °F]. Heubner, et al. (1989) reported the precipitation of topologically close packed phases in times as short as 15 minutes at temperatures in the range 800–900 °C [1,022–1,652 °F]. A significant increase in the intergranular corrosion rate was observed after 1 hour at 800 °C [1,472 °F] based on the results of standardized tests (American Society for Testing and Materials International, 1999). Bulk precipitation of topologically close packed phases was reported to occur after 10 hours at 800 °C [1,472 °F] and after 3 hours at 900 °C [1,652 °F]. In contrast, the results reported by Rebak, et al. (2000) indicate complete grain boundary precipitation after 10 hours at 800 °C [1,472 °F] and bulk precipitation within the grains after 100 hours at 800 °C [1,472 °F].

The effect of topologically close packed phase precipitation on the mechanical properties of Alloy 22 has been reported at temperatures in the range 593–760 °C [1,099–1,400 °F]

(CRWMS M&O, 2000s; Rebak, et al., 2000). Table 2.1.7-1 combines the mechanical properties and corrosion rates reported by Rebak, et al. (2000) with the microstructural observations of the material after isothermal exposures. It is apparent that the corrosion rate increases in response to partial grain boundary precipitation. In contrast, the Charpy impact energy for Alloy 22, after thermal aging that results in partial coverage of the grain boundaries with topologically close packed phase precipitates, is quite high and similar to the impact energy for material in the solution-annealed condition. The reduction in area measured on tensile test specimens decreased slightly from 75 to 80 percent in the solution annealed condition to 70 to 75 percent. Complete grain boundary precipitation was required for significant decreases in ductility or impact toughness. The activation energy necessary to decrease the impact energy to 203 J [150 ft·lb] was determined to be 247 kJ/mol [59 kcal/mol].

At 760 °C [1,400 °F], the highest temperature for which Charpy data were reported by Rebak, et al. (2000), an exposure of 10 hours is required to decrease the Charpy impact energy to 203 J [150 ft·lb]. Assuming the extrapolation of activation energy is valid at temperatures greater than 760 °C [1,400 °F], an isothermal exposure after 1 hour at 870 °C [1,598 °F] would decrease the Charpy impact energy from 360 to 203 J [266 to 150 ft·lb].

Table 2.1.7-1. Relationship Between Alloy 22 Condition, Ductility, Impact Resistance, and Corrosion Rate Using American Society of Mechanical Engineers Standard Corrosion Test Methods

Alloy 22 Condition	Tensile Specimen Reduction in Area	Charpy Specimens Impact Energy, J [ft·lb]	Corrosion Rate in ASTM* G28A Test, mm/yr [in/yr]
No precipitates	75 to 80 percent	360 [266]	1 [0.04]
Precipitates partially cover grain boundary	70 to 75 percent	360 [266]	2 to 4 [0.08 to 0.16]
Complete coverage of grain boundaries	55 to 65 percent	140 to 240 [103 to 177]	4 to 20 [0.16 to 0.79]
Complete coverage of grain boundaries plus precipitation within grains	20 to 50 percent	< 100 [< 74]	> 20 [> 0.79]

*American Society for Testing and Materials. "Standard Test Methods of Detecting Susceptibility to Intergranular Corrosion in Wrought, Nickel-Rich, Chromium-Bearing Alloys." ASTM G 28-97. 2001 Annual Book of ASTM Standards. Volume 3.02. West Conshohocken, Pennsylvania: American Society for Testing and Materials. 2001.

Systematic studies on the effect of compositional variations of Alloy 22 on thermal stability have shown that molybdenum, tungsten, and iron decrease the phase stability of the alloy and increase the precipitation kinetics of topologically close packed phases (Heubner, et al., 1989). The compositional specifications for Alloy 22 include 12.5 to 14.5-percent molybdenum, 2.5 to 3.5-percent tungsten and 2 to 6-percent iron. These specifications are external specifications, and the internal specifications used at production mills are more stringent for alloying concentration variations. The ENiCrMo-10 welding filler metal compositional specifications include 2.5 to 4.5-percent tungsten, which is a broader specification range compared with Alloy 22. Variations in the composition of the Alloy 22 plate and the filler metal used in the welding process may alter the kinetics of topologically close packed phase precipitation.

Additional evaluation is needed to determine the effects of microstructural and compositional variations of the plate and filler materials on the thermal stability and mechanical properties of the Alloy 22 waste package outer container. This evaluation may result in unanticipated variations in waste package corrosion resistance and mechanical properties. To address these concerns, DOE agreed[18] to provide justification that the American Society for Mechanical Engineers Boiler and Pressure Vessel Code case for the use of Alloy 22 results in acceptable waste package mechanical properties considering allowed microstructural and compositional variations of Alloy 22 base metal and the allowed compositional variations in the weld filler metals used in the fabrication of the waste packages. In addition, DOE agreed[19] to provide justification that the mechanical properties of the disposal container fabrication and waste package closure welds are adequately represented considering the (i) range of welding methods used to construct the disposal containers, (ii) postweld annealing and stress mitigation processes, and (iii) postweld repairs. DOE indicated that future work will include development and testing of welding, heat treating, and inspection equipment and processes.

In summary, microstructural and compositional variations of the plate material and filler metals may alter the kinetics of topologically close packed phase precipitation because of welding and thermal exposures. As a result, the waste package mechanical properties may be affected by the fabrication processes used to construct and close the disposal containers. Additional information is needed to assess the effects of fabrication processes and compositional and microstructural variations on the mechanical properties of the waste package. With the DOE agreement to provide the additional information, sufficient information should be available at the time of a potential license application for NRC to make a regulatory decision.

Nondestructive Evaluation of the Disposal Container

Before fabrication, DOE plans to examine the plate material to be used in the fabrication of the disposal containers, according to American Society of Mechanical Engineers requirements (1995d, Section V). This examination will include an ultrasonic inspection of the plates to be

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

[19]Ibid.

used for fabrication of the inner and outer cylinders of the disposal container (CRWMS M&O, 2001a).

As described in previous sections, fabrication methods used for the outer and inner cylinders involve longitudinal and circumferential seam welds. DOE plans to perform nondestructive examination of both types of welds. Fabrication welds for the Alloy 22 outer cylinder will be examined using liquid-penetrant, radiographic, and ultrasonic testing techniques. In the case of the Type 316 nuclear grade stainless steel inner cylinder, however, the nondestructive examinations will be limited to liquid-penetrant testing (CRWMS M&O, 2001a).

The fabrication of the top outer lid of the disposal container is detailed in the waste package design sketch (CRWMS M&O, 2000f, design sketch SK-0175). There will be two circumferential partial penetration welds and two circumferential fillet welds involved in the fabrication of this lid. DOE does not intend to perform nondestructive examination of any of these lid fabrication welds (CRWMS M&O, 2001a).

Fabrication of the Alloy 22 outer container will include a support ring designed to hold the weight of the inner container after assembly of the two containers in a nested arrangement (CRWMS M&O, 2000f). The welds of the ring will be machined to allow the bottom lid of the outer disposal container to be installed flush to the bottom of the ring and the inner disposal container to sit on the top of the ring. The machined surfaces will be inspected using liquid-penetrant testing (CRWMS M&O, 2001a).

After the inner and outer cylinders of the disposal container are fabricated, the bottom lid for each cylinder will be welded in place. The welds will be subjected to nondestructive examinations using liquid-penetrant, radiographic, and ultrasonic testing techniques (CRWMS M&O, 2001a). The DOE does not plan to perform nondestructive examinations of any other welds in the disposal container.

DOE originally intended to perform liquid-penetrant, radiographic, and ultrasonic testing of all disposal container inner cylinder fabrication welds (CRWMS M&O, 2000t). As delineated in a revision of this report (CRWMS M&O, 2001a), DOE now plans to limit the nondestructive evaluation to liquid penetrant testing for these welds. Since liquid penetrant testing can only uncover surface flaws, this new approach will fail to detect subsurface flaws. The integrity of these welds is particularly important because the inner container is relied on to maintain the structural strength and integrity of the waste package after emplacement. American Society of Mechanical Engineers (1995a, Subarticle NB–5210), which deals with vessel welded joints, requires volumetric and surface nondestructive evaluation of the welds. DOE should justify why it intends to rely solely on liquid-penetrant testing for inspection of inner cylinder fabrication welds.

In the case of the Alloy 22 outer closure lid of the waste package, DOE plans to do volumetric nondestructive evaluation of the closure weld (CRWMS M&O, 2001b) but does not plan to carry out any nondestructive evaluation of the other welds used in fabrication of the lid. Further, DOE will also carry out liquid-penetrant, radiographic, and ultrasonic testing of the Alloy 22 bottom lid weld for the waste package (CRWMS M&O, 2001a). Because the failure of any of these component welds can lead to a failure of the waste package, it is not clear why a graded approach is being adopted for nondestructive evaluation of the various welds. DOE should clearly state the reasons for conducting varying degrees of nondestructive evaluation on the welds involved in the fabrication of the waste package.

DOE agreed[20] to provide justification that the nondestructive evaluation methods used to inspect the Alloy 22 and Type 316 nuclear grade stainless steel plate material and welds are sufficient and capable of detecting defects that may adversely affect waste package preclosure structural performance. An assessment of the nondestructive examination methods used in the fabrication of the disposal containers has not been provided. Although the applicability of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code for the design and construction of the disposal containers has not been established, the fabrication and nondestructive evaluation sequence that DOE proposed is not consistent with recent versions of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. American Society of Mechanical Engineers (1995a, Subarticle NB–5130) requires the examination of the weld edge before welding when the material is greater than 51 mm [2 in] thick. In addition, American Society of Mechanical Engineers (1995a, Subarticle NB–5210 and Paragraph NB–5221) requires the volumetric inspection of circumferential and longitudinal welds. Because the minimum thickness of the Type 316 nuclear grade stainless steel inner disposal container is 50 mm [1.97 in], some of the disposal container designs may require additional inspection before welding according to the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. The proposed use of liquid-penetrant testing as the only method to inspect the inner disposal container fabrication welds does not meet the requirements of volumetric inspection.

Nondestructive Evaluation of the Closure Welds

The waste package design involves three closure lids (CRWMS M&O, 2000f, design sketch SK–0175). Because of the high radiation fields that will be present after the containers are loaded, remote welding processes are required to close the disposal containers. Before installation of the closure lid, the prepared surfaces will be visually inspected using a remote camera, followed by a tactile coordinate measurement using a coordinate measuring machine. The coordinate measuring machine will locate the center of the disposal container, relative to the closure gantry manipulator coordinate system, and determine disposal container cylindricity. It will provide a redundant check of the visual inspection for the weld preparations. The lids will be tack welded first and then circumferentially welded using remote gas metal arc or gas tungsten arc welding methods. Three remote cameras (lead, trail, and inspection) on the robotic arm welder will provide real time weld inspection with digital image processing and machine vision techniques. In case of any alarm, the welding process will be stopped and the operator notified of the problem. It may be possible to immediately perform the repair at the weld station, and then resume the welding process. If the repair requires extensive machining, the disposal container will be moved to a repair station (CRWMS M&O, 2001b).

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

The inner disposal container lid, made of Type 316 nuclear grade stainless steel, will be 95 mm [3.74 in]-thick (CRWMS M&O, 2000e). A shear ring will be used with the inner lid. It will be assembled from three or four segments and welded in place. Gas metal arc welding will be used to perform this operation. The gas metal arc welding robotic arm will have the ability to perform a full circumferential weld with a rotational range greater than 360 degrees. All critical parameters will be recorded in process, and alarm or fault set points in the closure cell control system will notify the operator immediately of any parameter anomalies, and place a flag in the data stream. After welding the inner lid, the inner container will be evacuated and filled with inert helium gas via a purge port. The inner container will then be leak tested to confirm the integrity of the welds. The process sequence flowchart for disposal container closure (CRWMS M&O, 2001b) indicates DOE does not plan to conduct a nondestructive examination of the inner container lid weld.

The middle lid, made of Alloy 22, will be 10 mm [0.39 in]-thick and will be welded to the outer barrier using a partial penetration weld. The original square root partial penetration weld design may be modified to include a chamfer at the root of the weld. The gas tungsten arc welding method is presently being considered for remote welding of this lid (CRWMS M&O, 2001b). The welding sequence will be similar to that described in the previous two paragraphs. There will be a remote visual inspection of the weld preparation surfaces followed by a dimensional inspection using a tactile coordinate measuring system, tack welding, and then circumferential welding of the lid. Nondestructive evaluation of the weld will be performed to ensure acceptability. Laser peening will be used for stress relief of the weld, followed by a second nondestructive evaluation of the weld. There is no identifiable method for performing a volumetric inspection of the middle closure lid weld at present. It is expected, however, that a suitable process and tooling for this nondestructive evaluation will be developed later (CRWMS M&O, 2001b).

The extended outer shell lid is also made of Alloy 22. It will be tack welded and then circumferentially welded to the outer container using the narrow groove gas tungsten arc welding method. For the most part, the welding sequence will be similar to that described in the preceding paragraphs. Remote visual inspection of the weld preparation surfaces will be used to ensure that the surfaces are free of deposits and scale. The weld joint will be back purged using Argon, followed by tack welding, and then circumferential welding of the lid. Nondestructive evaluation of the weld will be performed to ensure acceptability. The inspection will require two passes (rotations). A surface examination will be performed using an alternating current field measurement probe, followed by a volumetric inspection using ultrasonic testing and a couplant. The weld will then be induction annealed, and the nondestructive evaluation will be repeated one final time. This strategy allows repairs to be made before the postweld heat treatment, ensuring the postweld heat treatment does not have to be repeated because it is thought that additional postweld heat treatments would be detrimental to the long-term performance of the waste package (CRWMS M&O, 2001b).

To experimentally determine the minimum detectable flaw size using ultrasonic testing, DOE fabricated two Alloy 22 mockups fabricated using 25-mm [1-in]-thick material. The plates were welded using gas tungsten arc welding with joint dimensions similar to that proposed for the outer closure weld. The Alloy 22 mockup was then machined so the dimensions were representative of the cross-sectional geometry of the extended outer shell lid of the waste package in the area of the closure weld. The lid configuration limits the available scan surfaces for ultrasonic testing to the top surface of the waste package on each side of the weld and to the side of the weld from the outside diameter of the waste package at the elevation of the weld. Geometric features in the weld area do not allow ultrasonic testing from other surfaces. The two mockups were constructed so they reflected these constraints to available ultrasonic testing locations. Each mockup contained five flaws of known dimension and location. There were two types of planar flaws, lack of fusion and lack of penetration. The third type of implanted flaw was porosity. Examinations were performed by scanning from the top of the mockup plate with 45 and 70°-angle beams directed toward the weld from each side of the weld. A straight beam scan was performed on the closure weld mockup specimen by placing the transducer on the crown of the weld. An additional straight beam scan was performed by placing the transducer on the side of the weld mockup specimen, which was machined so that the ultrasonic beam path was equivalent to the distance between the waste package outside diameter and the closure weld. The last scan orientation resulted in a sound beam traveling normal to the weld axis and was optimum for detecting fabrication flaws that follow the weld fusion line, such as lack of fusion and lack of penetration (CRWMS M&O, 2001b).

Results obtained from the scans indicated that the last scan orientation described in the preceding paragraph provided the greatest response from planar flaws. Also, planar type flaws (i.e., fusion and penetration flaws) with a minimum area of 16 mm^2 [0.025 in^2] can be detected in this weld joint geometry. Small volumetric porosity reflectors, however, were not detected, primarily because of the scattering of the sound wave from the round-shaped individual gas pores. The inability to detect small volumetric porosity reflectors may be acceptable (American Society for Mechanical Engineers, 1995e) because the geometric discontinuities associated with the individual gas pores do not cause localized increases in stress that appreciably affect the initiation of stress corrosion cracking or mechanical failure.

In summary, DOE agreed[21] to provide justification that the nondestructive evaluation methods used to inspect the Alloy 22 and Type 316 nuclear grade stainless steel plate materials and welds are sufficient and capable of detecting defects that may adversely affect waste package preclosure structural performance. Subsequent to the technical exchange agreement, DOE demonstrated, through an assessment of the ultrasonic inspection of the closure weld mockup, that flaws, such as lack of penetration and lack of fusion, can be detected (CRWMS M&O, 2001b). Further, information DOE provided subsequent to the technical exchange agreement suggests that, because of waste package weld geometry, a full volumetric inspection may not be suitable for the middle Alloy 22 closure lid. A demonstration of a suitable nondestructive evaluation of this closure lid weld will require some development and may require an adjustment to the joint geometry. Finally, for the inner Type 316 nuclear grade stainless steel closure weld, there does not appear to be a method or process to perform a remote nondestructive examination of this weld. The potential consequence of not performing a nondestructive examination of the inner disposal container closure lid has not been assessed.

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

Criticality Design Criteria

The general preclosure criticality control requirement is specified under 10 CFR 63.112(e)(6), which indicates that the structures, systems, and components must be designed in such a way that would " ... prevent and control criticality ... ". In its review of the preliminary preclosure safety assessment (DOE, 2001b), NRC identified the following concerns. The first was the DOE reliance on the level of the burnup in the commercial spent nuclear fuel assemblies for designing the criticality control systems of the waste packages. Another concern included consideration of events (e.g., internal and external flooding; spent nuclear fuel assembly misload events; events in the pools and storage racks; and, in general, Categories 1 and 2 events with respect to criticality), when designing the surface and subsurface facilities. Furthermore, the issues NRC identified when reviewing the DOE report (2001b) are briefly discussed.

According to NRC Regulatory Guide 3.71 (NRC, 1998), burnup of the spent nuclear fuel assemblies must be verified through measurements before they can be loaded into waste packages if the licensee chooses to take credit for the burnup when designing the criticality control system of the waste package. During the preclosure technical exchange,[22] DOE agreed to provide an approach for verification of fuel assembly burnup. DOE stated that burnup credit is only being sought for commercial spent nuclear fuel, and that burnup information for the majority of the fuel developed and available through reactor records maintained according to NRC-accepted quality assurance requirements is the best source of assembly burnup information. NRC agreed that reactor records are a more accurate source of fuel assembly burnup data than physical measurements. NRC stated that its current position, however, is that measurements are needed to verify the burnup indicated by reactor records.

Several waste package internal component configurations are considered in the determination of the effective neutron multiplication factor (i.e., k_{eff}): (i) an intact basket with a neutron absorber inside the waste package, (ii) a degraded basket with the neutron absorber flushed from the waste package and iron-oxide corrosion product uniformly distributed throughout the waste package, and (iii) a degraded basket with iron oxide settled to the lowest 3.5 rows of assemblies (CRWMS M&O, 2000f). Although the configurations with degraded baskets are more significant for postclosure performance than for preclosure performance, the analyses of the degraded configurations suggest that up to 11.2 percent of the pressurized water reactor fuel waste packages will need some additional criticality control measures. Several criticality control options have been considered including new reactor control rod assemblies, spent reactor control rod assemblies, and disposable control rod assemblies specifically manufactured for the waste packages. The zirconium clad B_4C disposable control rods are the preferred option for the site recommendation waste package design.

With respect to the consideration of events such as flooding, misload, and the like, DOE stated " ... established design requirements that preclude preclosure criticality unless two unlikely

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

independent events occur [e.g., CRWMS M&O (2000u)]. The probability of two unlikely independent events occurring will be less than 10^{-6}/yr." Staff believe the double-contingency principle (i.e., two unlikely events), which has been used historically in designing criticality control systems for facilities, storage, and transportation packages, does not require the licensee to quantify the probability of the unlikely events. According to 10 CFR Part 63, however, events must be identified, their probabilities quantified, and assigned designation as Categories 1 or 2 events. On the other hand, 10 CFR 63.112(e)(6) indicates that the structures, systems, and components must be designed in such a way that nuclear criticality is prevented. Therefore, as DOE has indicated, the repository preclosure structures, systems, and components will be designed to prevent criticality under normal operation and Categories 1 and 2 events.[23]

Waste Package Shielding

The current site recommendation waste package design does not provide additional shielding for personnel protection (CRWMS M&O, 1999b). It is intended that the waste package containment barriers provide sufficient shielding to protect the waste package materials from radiation-enhanced corrosion (CRWMS M&O, 2000f). The maximum dose rate on the external surfaces of the waste package with 21 pressurized water reactor fuel assemblies is 13.30 ± 0.60 Sv/hr [1,330 ± 60 rem/hr], whereas the maximum dose rate for a waste package with 44 boiling water reactor fuel assemblies is 14.09 ± 0.32 Sv/hr [1,409 ± 32 rem/hr] (CRWMS M&O, 2000e). Shielding for staff protection is to be achieved by operational procedures, in conjunction with other structures, systems, and components, during waste package handling and transport.

The current DOE waste package design description appears to adequately provide shielding to prevent radiolysis-induced corrosion. Additional protection for workers is provided by other structures, systems, and components. The design of the waste package is still being developed, so DOE will provide additional design information in future documents. These documents will be reviewed as they become available.

Designing for Normal Operation and Categories 1 and 2 Event Sequences

DOE identified event sequences presently being considered in establishing the design criteria and specifications for important to safety structures, systems, and components (DOE, 2001b). A detailed discussion of the DOE identification and categorization of event sequences that pertain to the preclosure period of the proposed repository can be found in Subsections 2.1.4 and 2.1.5. The discussion presented in this section is limited to the postulated waste package drop event. As more information becomes available the scope of this discussion will be expanded to include other relevant important to safety structures, systems, and components event sequence and consequence analyses.

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

The waste package drop event has been characterized as an internal event sequence that is not expected to result in a radiological release because it is prevented by the design of the waste package (CRWMS M&O, 2000h). Analyses intended to support this characterization have been performed (CRWMS M&O, 2000v). The scope of these analyses was limited to a single waste package drop orientation. It is not clear that a single drop orientation scenario is sufficient to bound the potential for waste package failure, considering the number of different waste package handling operations and the present lack of design detail for the various cranes and other devices that will be used to transfer the waste package from the waste handling building to its emplacement within the drift. DOE stated during the preclosure technical exchange[24] that, as part of the normal design process, design basis dynamic events will be reevaluated as the designs for both the surface and subsurface facilities mature. It should be noted that DOE does not consider the waste package to be breached if the inner disposal container remains intact.

No specific requirements are provided in 10 CFR Part 63 that mandate waste package drop tests or any other empirical evaluations that will demonstrate the structural integrity of the waste package subjected to other design basis events, such as those required by 10 CFR Part 71. As a result, the means used to demonstrate the ability of the waste package to withstand the postulated event sequences is at the discretion of DOE. In the case of demonstrating the ability of the waste package to withstand handling drops without breaching, DOE has chosen to use numerical simulations based on the finite element method as the sole basis for its safety case. Although DOE has not precluded the use of actual waste package drop tests in the future to demonstrate the structural integrity of the waste package, there are no specific plans to do so at this time.

Because of the reliance on computer simulations to demonstrate the performance capabilities of the waste package, the assumptions, boundary conditions, material characterization, numerical formulations (along with their inherent limitations), level of mesh discretization, and failure criteria will have to be scrutinized more rigorously. As a result, DOE agreed[25] to (i) demonstrate that the mesh discretizations of the finite element models used to simulate the effects of waste package drop events are sufficient to provide reasonably convergent results that can be used to assess potential failure, (ii) justify the constitutive models used to represent the response of the waste package materials to impact loads (e.g., the inclusion or exclusion of temperature and strain rate effects), (iii) provide documentation of all boundary conditions used for the numerical models and the technical basis or rationale for them, and (iv) provide evidence that the criterion used to establish failure adequately bounds the uncertainties associated with effects not explicitly considered in the simulation. Specific uncertainties not presently considered in the waste package drop analyses are (i) residual stresses arising from the closure weld fabrication process, (ii) dimensional and material variability, (iii) ground motion effects caused by a seismic event (waste package drops are more likely to occur during seismic events), (iv) sliding and inertial effects of the spent nuclear fuel, and related matters.

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

[25]Ibid.

The waste package drop analyses DOE performed (CRWMS M&O, 2000v) does not indicate whether the structural integrity of the spent nuclear fuel was considered when establishing allowable drop heights. At the preclosure technical exchange,[26] DOE stated that in case of a drop, an assessment would be made as to whether the waste form must be repackaged, but the primary consideration when establishing drop heights is the integrity of the waste package. DOE also noted that the repackaging requirements have not yet been established, but they will be based on long-term performance needs.

Fuel Cladding Thermal Control

Temperature control for commercial spent nuclear fuel waste packages after emplacement within the repository will be provided using a combination of drift spacing, waste package spacing, ventilation during the preclosure period, waste package configuration, and thermal blending of the spent nuclear fuel. The maximum allowed thermal output of any waste package is 11.8 kW [40,263 BTU/hr] (CRWMS M&O, 2000f). With the exception of waste packages with 24 boiling water reactor fuel assemblies, the waste packages containing commercial spent nuclear fuel have aluminum thermal shunts added to conduct heat from the interior of the waste package to the waste package inner container. The axial and radial gaps between the inner and outer containers after differential thermal expansion will affect the steady-state waste package temperatures. Larger gaps will tend to cause higher interior and lower exterior (i.e., outer container) temperatures. Aluminum Alloys 6061 and 6063 were chosen instead of copper because of concerns that copper may react with chloride introduced by water entering the waste package and cause accelerated degradation of the zirconium alloy cladding. For the commercial spent nuclear fuel waste package configurations, the 21 pressurized water reactor fuel waste packages with absorber plates have the highest heat output with an average of 11.33 kW [38,650 BTU/hr] (CRWMS M&O, 2000f). Peak cladding temperatures are calculated to be less than 300 °C [572 °F], even with close waste package spacing (CRWMS M&O, 2000e). The heating, ventilation, and air conditioning system within the waste handling building will maintain fuel cladding temperatures within acceptable limits before packaging and emplacement.

The current DOE waste package design description appears to include components to provide thermal control so the fuel cladding temperature will be maintained within acceptable limits. The design of the waste package is still under development, so DOE will provide additional design information in future documents. These documents will be reviewed as they become available.

Backfill Design

Backfill is not used in the present conceptual design of the proposed repository. As a result, no assessment is required.

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

Sorption Barrier Design

A sorption barrier is not used in the present conceptual design of the proposed repository. As a result, no assessment is required.

2.1.7.4 Status and Path Forward

Table 2.1.7-2 provides the status of the Design of Structures, Systems, and Components Important to Safety and Safety Controls. The table also enumerates the related DOE and NRC agreements pertaining to the Repository Design and Thermal-Mechanical Effects and Container Life and Source Term Key Technical Issues. The agreements listed in the table are associated with acceptance criteria discussed in Sections 2.1.7.3.3.2 and 2.1.7.3.3.3. Note that the status and the detailed agreements (or path forward) pertaining to all the key technical issue subissues are provided in Table 1.1-3 and Appendix A.

Table 2.1.7-2. Summary of Resolution Status for Design for Structures, Systems, and Components Important to Safety and Safety Controls Preclosure Topic

Preclosure Items	Status	Related Agreements*	Comments
Relationship between the Design Criteria and Design Basis and the Regulatory Requirements	Pending	†	Staff Review Incomplete
Geologic Repository Operations Area Design Methodologies	Pending	†	Staff Review Incomplete
Assumptions, Codes, and Standards for Surface Facilities Design	Pending	†	Staff Review Incomplete
Materials for Surface Facilities Design	Pending	†	Staff Review Incomplete
Load Combinations for Surface Facilities Design	Pending	†	Staff Review Incomplete
Surface Facilities Design Analyses and Documentation	Pending	†	Staff Review Incomplete
Assumptions, Codes, and Standards for Subsurface Facility Design	Pending	†	Staff Review Incomplete
Subsurface Operating Systems Design	Pending	†	Staff Review Incomplete
Material and Material Properties for Subsurface Facility Design	Pending	RDTME.3.01	Impact of corrosion on the effectiveness of ground-support system

Table 2.1.7-2. Summary of Resolution Status for Design for Structures, Systems, and Components Important to Safety and Safety Controls Preclosure Topic (continued)			
Preclosure Items	**Status**	**Related Agreements***	**Comments**
Load Combinations for Subsurface Facility Design	Pending	RDTME.2.01 RDTME.2.02 RDTME.3.02 RDTME.3.03	Seismic load characterization and critical combination of thermal and seismic loadings
Models and Rock Properties for Subsurface Facility Design	Pending	RDTME.3.04 RDTME.3.05 RDTME.3.07 RDTME.3.08 RDTME.3.10 RDTME.3.13	Rock properties and data sufficiency, rock strength, and fracture pattern analyses
Subsurface Ground-Support Systems Design	Pending	RDTME.3.06 RDTME.3.09	Drift invert stability and rock support system analyses
Subsurface Ventilation System Design	Pending	RDTME.3.14	Ventilation modeling and validation
Subsurface Power and Power Distribution Systems Design	Pending	†	Staff Review Incomplete
Maintenance Plan for Subsurface Facility	Pending	†	Staff Review Incomplete
Waste Package and Engineered Barrier Subsystem Design	Pending	PRE.07.01 through PRE.07.05	Criticality analysis, finite element modeling, weld filler material compatibility, nondestructive evaluation methods, and mechanical properties after welding

*Related DOE and NRC agreements are associated with one or more acceptance criteria.
†Not discussed at the first DOE and NRC Technical Exchange and Management Meeting on Preclosure Safety, July 24–26, 2001, Las Vegas, Nevada.

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

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