1 INTRODUCTION

1.1 Background and Report Structure

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

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

Because of the short time available to review the potential DOE license application, the NRC, consistent with the Nuclear Waste Policy Act (1982), made early identification of issues and issue resolution a major part of the prelicensing interactions. Prelicensing interactions include public meetings at which all stakeholders including the State of Nevada, Tribal governments, affected units of local governments, and interested members of the public have the opportunity to participate. Issue resolution is based on an in-depth review of DOE and contractor documents; the independent work of NRC and its contractor, the Center for Nuclear Waste Regulatory Analyses (CNWRA); published literature; and other publicly available information. Using risk insights information, the NRC staff has developed a risk insights baseline to focus reviews on issues most important to repository performance. The prelicensing consultations and the issue resolution process are in conformance with the NRC efforts to streamline its high-level waste program (NRC, 1999) and prepare for an efficient and competent review of any potential license application DOE may submit.

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

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

Table 1.1-1. Key Technical Issues and Associated Subissues

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

Table 1.1-1. Key Technical Issues and Associated Subissues (continued)						
Key Technical Issue	**Associated Subissues**					
Thermal Effects on Flow	TEF1—Features, Events, and Processes Related to Thermal Effects on Flow	TEF2—Thermal Effects on Temperature, Humidity, Saturation, and Flux	—	—	—	—
Repository Design and Thermal-Mechanical Effects	RDTME1—Design Control Process Implementation of an effective design control process within the overall Quality Assurance program	RDTME2—Seismic Design Methodology Design of the geologic repository operations area for the effects of seismic events and direct fault disruption [including implications for drift stability, key aspects of emplacement configuration (i.e., fault offset distance, retrievability, and waste package damage)]	RDTME3—Thermal-Mechanical Effects Consideration of thermal-mechanical effects on underground facility design and performance (including implications for drift stability, key aspects of emplacement configuration that may influence thermal loads and associated thermomechanical effects, retrievability, the change in geometry and flow into and out of emplacement drifts, and fault setback distance)	RDTME4—Design and Long-Term Contribution of Seals to Performance Design and long-term contribution of repository seals in meeting the postclosure performance objectives (including implications for inflow of water and release of radionuclides to the environment)	—	—
Total System Performance Assessment and Integration	TSPAI1—System Description and Demonstration of Multiple Barriers	TSPAI2—Scenario Analysis and Event Probability	TSPAI3—Model Abstraction	TSPAI4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards	—	—

Table 1.1-1. Key Technical Issues and Associated Subissues (continued)						
Key Technical Issue	**Associated Subissues**					
Unsaturated and Saturated Flow Under Isothermal Conditions	USFIC1—Climate Change What is the likely range of future climates at Yucca Mountain?	USFIC2—Hydrologic Effects of Climate Change What are the likely effects of climate change?	USFIC3—Shallow Infiltration What is the estimated amount and spatial distribution of present day shallow infiltration?	USFIC4—Deep Percolation What is the estimated amount and spatial distribution of percolation through the potentialrepository horizon (present day, and through the period of repository performance)?	USFIC5—Saturated Zone What are the ambient flow conditions in the saturated zone, and what are the likely dilution mechanisms?	USFIC6—Matrix Diffusion To what degree does matrix diffusion occur in the unsaturated and saturated zones?
Radionuclide Transport	RT1—Radionuclide Transport Through Porous Rock	RT2—Radionuclide Transport Through Alluvium	RT3—Radionuclide Transport Through Fractured Rock	RT4—Nuclear Criticality in the Far Field	—	—
Activities Related to Development of the U.S. Nuclear Regulatory Commission Yucca Mountain Regulations	—	—	—	—	—	—

Of the 10 key technical issues, the first 9 are directly related to the objective of this report; the last pertains to development of the NRC regulation in 10 CFR Part 63.[1] A brief discussion of 10 CFR Part 63, as well as other applicable regulations, is included in Section 2.1. Technical issues related to preclosure safety were not defined in the mid-1990s, but are included in this report.

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

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

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

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

To clarify the issue structure, charts were constructed to depict components of a safety review (Figure 1.1-1) and the relationships among various principal components of a postclosure performance assessment for the potential repository at Yucca Mountain (Figure 1.1-2). These charts show that an efficient way to review the DOE postclosure safety case and its associated performance assessment is to follow the partitioning depicted in Figure 1.1-2. This partitioning is based primarily on the natural progress of potential radionuclide release and transport to a reasonably maximally exposed individual at the Yucca Mountain site. The topics (14) at the most detailed level of review in Figure 1.1-2 are called integrated subissues or model abstractions, mainly because each integrated subissue draws information from multiple key technical issues. The integrated subissues represent an interdisciplinary and logical framework for assessing the DOE postclosure performance assessment.

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

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

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

The geologic repository would be a first-of-a-kind facility, and there is little experience regarding its postclosure long-term performance. For this reason, and also because significant experience already exists at NRC in regulating safety during construction and operation of other nuclear facilities, staff emphasized developing licensing review capabilities with respect to postclosure during the early years of the program. Beginning in fiscal year 2000, however, the importance of preclosure safety was elevated in view of the DOE plans to proceed with a design to support the site recommendation. Chapter 4 provides a status of the preclosure issues.

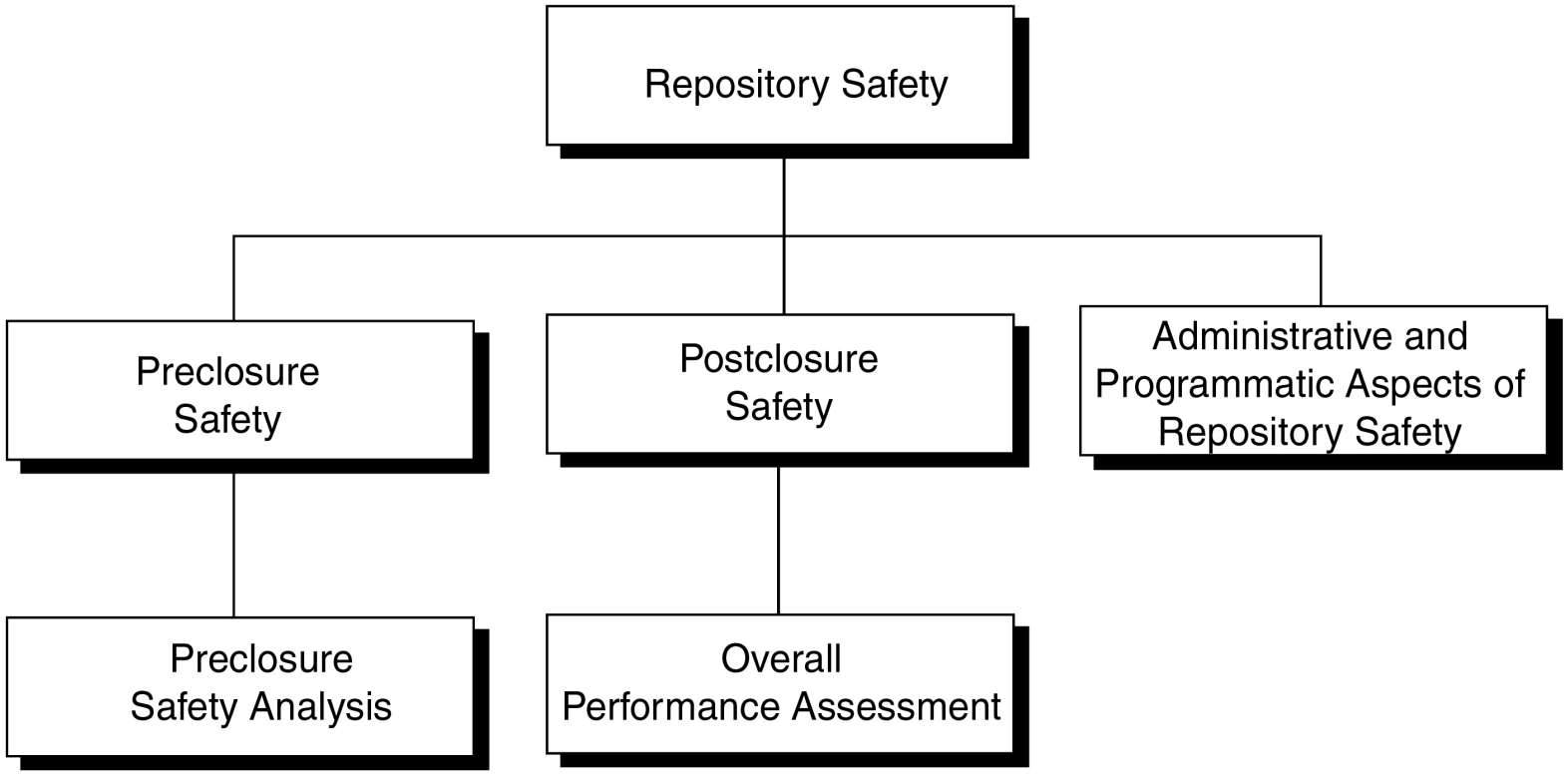

Figure 1.1-1. Review Components of Repository Safety

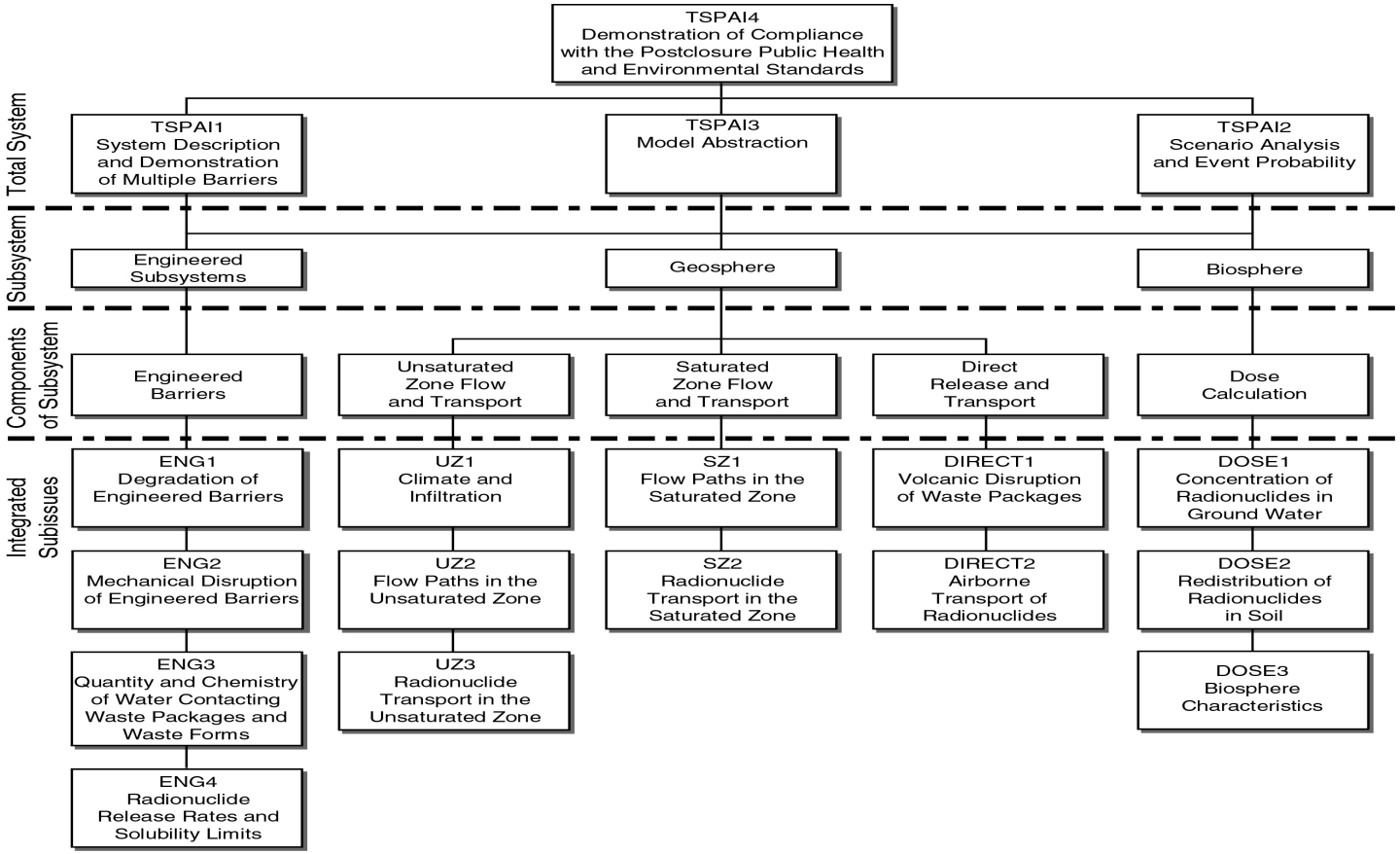

Figure 1.1-2. Components of Postclosure Performance Assessment Review

Based on NUREG–1804 (NRC, 2003), the 10 preclosure topics defined for this purpose are (i) Site Description As It Pertains to Preclosure Safety Analysis; (ii) Description of Structures, Systems, Components, Equipment, and Operational Process Activities; (iii) Identification of Hazards and Initiating Events; (iv) Identification of Event Sequences; (v) Consequence Analyses; (vi) Identification of Structures, Systems, and Components Important to Safety; Safety Controls; and Measures to Ensure Availability of the Safety Systems; (vii) Design of Structures, Systems, and Components Important to Safety and Safety Controls; (viii) Meeting the 10 CFR Part 20 As Low As Is Reasonably Achievable Requirements for Normal Operations and Category 1 Event Sequences; (ix) Plans for Retrieval and Alternate Storage of Radioactive Wastes; and (x) Plans for Permanent Closure and Decontamination, or Decontamination and Dismantlement of Surface Facilities (Reamer, 2001).

Chapter 5 of this report documents the status of issue resolution for the 14 integrated subissues associated with model abstraction for postclosure performance. To put review of the integrated subissues in the context of the total system performance assessment, four additional review issues are defined (Figure 1.1-2): (i) TSPAI1—System Description and Demonstration of Multiple Barriers, (ii) TSPAI2—Scenario Analysis and Event Probability, (iii) TSPAI3—Model Abstraction, and (iv) TSPAI4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards. These topics also are discussed in Chapter 5. The discussion of TSPAI3—Model Abstraction (Section 5.1.3) covers the 14 integrated subissues. Each integrated subissue draws information from various key technical issue subissues, which are clearly identified in the text; these relationships also are described in Table 1.1-2.

The NRC regulations call for DOE to conduct performance confirmation activities. The objective of performance confirmation is to acquire information by conducting monitoring, *in-situ* experiments, laboratory experiments, and analyses that will provide confidence that the repository will continue to perform in a safe manner during the preclosure and postclosure periods. Chapter 6 discusses this aspect of the repository program. The DOE research and development programs to resolve any safety questions also are discussed in Chapter 6.

Confidence in the estimated preclosure and postclosure safety indicators and performance measures will be based, in part, on the premise that data were collected and analyses conducted according to the Quality Assurance program required by NRC and akin to that set forth in Appendix B of 10 CFR Part 50. NRC has followed the development and implementation of the Quality Assurance program for the quality-affecting activities of the Yucca Mountain project. This was accomplished by participating as observers during quality assurance audits conducted by DOE and assessing the status of the Quality Assurance program through periodic meetings. Quality assurance and other administrative and programmatic aspects of the Yucca Mountain project are discussed in Chapter 7.

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

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

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

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

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

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

The report also provides a risk-informed context for the assessment by the NRC staff of the current information available to support a potential DOE license application. Review of the issues is intended to increase the likelihood that DOE will have information available to submit a high quality license application for a potential high-level waste repository at Yucca Mountain. The NRC acceptance review will determine if a potential license application contains the

information necessary for the NRC to docket the application and begin its technical review. Potential docketing of an application is not an NRC judgment regarding whether, for example, a construction authorization should be issued. Licensing decisions will only be made after review of any license application.

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

1.2 Prelicensing Issue Resolution Process

The NRC strategic plan (2000) calls for early identification and resolution of issues at the staff level before receipt of a potential license application to construct a geologic repository. The principal means for achieving this goal is through prelicensing interaction with DOE as mandated by the Nuclear Waste Policy Act (1982).

The purpose of issue resolution is to ensure that sufficient information is available to enable the NRC staff to review a potential license application. Resolution at the staff level does not preclude an issue from being raised and considered during the licensing proceedings and does not predecide the NRC staff evaluation of that issue after staff review of any license application. During prelicensing, issue resolution at the staff level is achieved when the staff has no further questions or comments at a point in time regarding how DOE is addressing an issue. The agreement items reached during the technical exchanges with DOE reflect the understanding by the NRC staff of issues most important to repository performance. This understanding is based on limited, focused, and risk-informed reviews of selected portions of information made publicly available by DOE. Depending on the DOE responses, agreement items are either closed or needs for additional information are identified by the staff. The availability of new or additional information (e.g., changes in design parameters) could raise new questions or comments regarding a previously resolved issue.

Three categories of issue resolution are defined by NRC: closed, closed-pending, and open. Issues are closed if the DOE approach and available information acceptably address staff questions such that no information beyond what is currently available will likely be required for regulatory decision making at the time of any license application. Issues are closed-pending if the DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (through specified testing or analysis), acceptably addresses the NRC questions so that no information beyond that provided, or agreed to, will likely be required at the time of a potential license application. Issues are open if NRC identifies questions regarding the DOE approach or information and DOE has not yet acceptably addressed the questions or agreed to provide the necessary additional information in a potential license application. As a result of recent technical exchanges, DOE and NRC reached agreements pertaining to a subset of the nine postclosure key technical issues and the associated subissues and preclosure issues. The status of each key technical issue subissue is presented in Table 1.2-1. The agreements reached during the technical exchanges are included in Appendix A.

Table 1.2-1. Status of Key Technical Issue Subissues Resolutions						
Key Technical Issue	**Subissue 1**	**Subissue 2**	**Subissue 3**	**Subissue 4**	**Subissue 5**	**Subissue 6**
Unsaturated and Saturated Flow Under Isothermal Conditions	Closed	Closed	Closed-Pending	Closed-Pending	Closed-Pending	Closed-Pending
Igneous Activity	Closed-Pending	Closed-Pending	N/A	N/A	N/A	N/A
Container Life and Source Term	Closed-Pending	Closed-Pending	Closed-Pending	Closed	Closed-Pending	Closed-Pending
Structural Deformation and Seismicity	Closed-Pending	Closed-Pending	Closed-Pending	Closed	N/A	N/A
Radionuclide Transport	Closed-Pending	Closed-Pending	Closed-Pending	Closed-Pending	N/A	N/A
Thermal Effects on Flow	Closed-Pending	Closed-Pending	N/A	N/A	N/A	N/A
Evolution of the Near-Field Environment	Closed-Pending	Closed-Pending	Closed-Pending	Closed-Pending	Closed-Pending	N/A
Repository Design and Thermal-Mechanical Effects	Closed	Closed-Pending	Closed-Pending	Closed	N/A	N/A
Total System Performance Assessment and Integration	Closed-Pending	Closed-Pending	Closed-Pending	Closed-Pending	N/A	N/A

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

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

1.3 References

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

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2 RISK-INFORMED REVIEW PROCESS

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

Following is a brief history of regulations and a discussion of the main principles included in the standards and regulations. Figure 2.1-1 provides a timeline for pertinent rulemaking (adapted from CRWMS M&O, 2000).

The Nuclear Waste Policy Act (1982) established the national policy and defined the responsibilities of various Federal agencies for the safe disposal of spent nuclear fuel and high-level waste generated mainly as a result of commercial power production and defense activities. As mandated by statute, the U.S. Department of Energy (DOE) is responsible for siting, building, operating, and closing an underground geologic repository; the U.S. Environmental Protection Agency (EPA) has the responsibility of setting generally applicable environmental radiation protection standards based on authority established according to other laws; and the U.S. Nuclear Regulatory Commission (NRC) must implement the EPA standards by incorporating them into NRC regulations, issue technical criteria for licensing a repository, and decide whether to authorize construction, operation, and closure of a repository.

In 1985, EPA established generic standards for the management, storage, and disposal of high-level waste in 40 CFR Part 191 (50 FR 38066, September 19, 1985). NRC developed its implementing regulations in 10 CFR Part 60. These standards and regulations were intended to apply to all appropriate facilities in the United States, including the potential high-level waste repository at Yucca Mountain, Nevada. In 1987, the U.S. Court of Appeals for the First Circuit Court invalidated the standards and remanded them to EPA (Natural Resources Defense Council, Inc., 1987). Also in 1987, the Nuclear Waste Policy Act (1982) was amended by, among other actions, designating Yucca Mountain, Nevada, as the only potential site to be characterized for a high-level waste repository.

In 1992, Congress directed EPA, in Section 801 of the Energy Policy Act of 1992 (1992), to contract with the U.S. National Academy of Sciences to advise EPA on the appropriate technical basis for public health and safety standards governing a potential repository at Yucca Mountain. On August 1, 1995, the U.S. National Academy of Sciences Committee on Technical Basis for Yucca Mountain Standards issued its report Technical Bases for Yucca Mountain Standards (National Research Council, 1995). EPA issued its final standards applicable to Yucca Mountain in 40 CFR Part 197 on June 13, 2001. NRC prepared its final regulations based on careful review and consideration of the public comments received on its proposed rule and the statutory direction for NRC to adapt its technical criteria to be consistent with final EPA standards. After considering public comments on the draft rule, NRC published its final regulations in a new 10 CFR Part 63 on November 2, 2001. These regulations include criteria for long-term repository performance as well as licensing procedures, records and reporting, monitoring and testing programs, performance confirmation, quality assurance, personnel training and certification, and emergency planning.

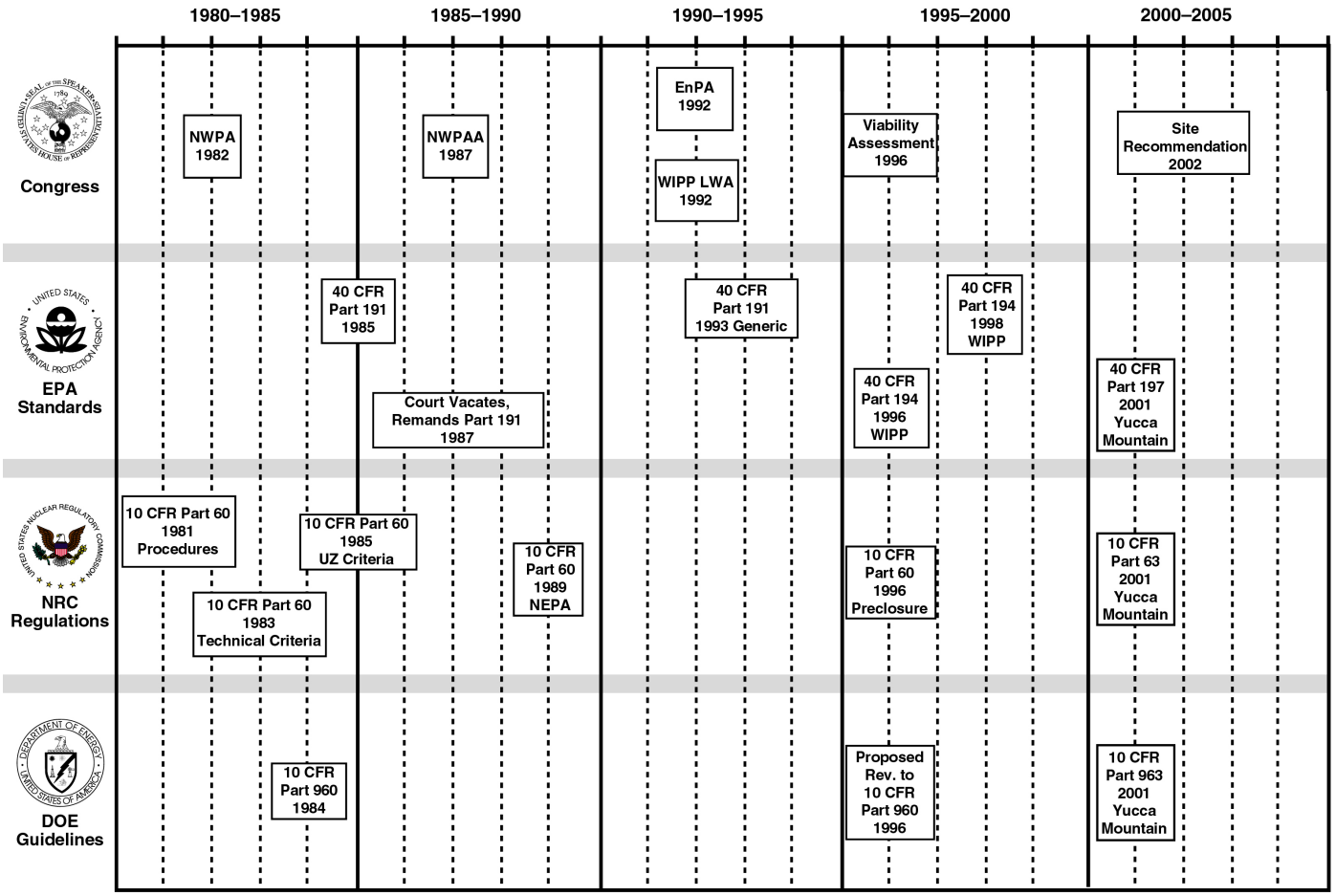

Figure 2.1-1. Timeline of Legislative and Regulatory Events, 1980–2005

EPA Standards

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

Radiation Standards: On June 13, 2001, EPA promulgated its final public health and environmental radiation standards (40 CFR Part 197) for the operational phase of repository development (i.e., the period of time during which waste is brought to the site and placed in the repository) and for permanent disposal (i.e., the period of time after permanent closure or sealing of the repository). The two phases are often referred to as the preclosure and postclosure phases. The preclosure or operational phase of the repository is limited by an annual individual dose limit of 0.15 mSv/yr [15 mrem/yr] for members of the public from normal operations at the repository.

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

Performance Assessments: Performance assessment is a systematic analysis that identifies the features, events, and processes (i.e., specific conditions or attributes of the geologic setting; degradation, deterioration, or alteration processes of engineered barriers; and interactions between the natural and engineered barriers) that might affect performance of the geologic repository; examines the effects on performance; and estimates the potential radiological consequences. DOE is required to use performance assessment to show compliance with the

postclosure performance objectives. To ensure DOE uses meaningful and reasonable calculations, EPA specified certain limitations for the performance assessment to preclude boundless speculation. The DOE performance assessments are not to include consideration of very unlikely features, events, and processes, which EPA defines to be those features, events, and processes that have less than 1 chance in 10,000 of occurring within 10,000 years of disposal. In addition, the EPA standards direct NRC to exclude unlikely features, events, and processes, or sequences of events and processes, from the required assessments for demonstrating compliance with the human intrusion and groundwater protection standards. EPA did not define unlikely features, events, and processes in its standards, and left the specific probability of the unlikely features, events, and processes for NRC to define. The EPA standards also specify criteria that pertain to the characteristics of a reference biosphere. The standards specify that the reference biosphere used in the performance assessments needs to be consistent with present conditions in the Yucca Mountain area, and speculation on changes in society, human biology, or increases or decreases in human knowledge or technology should not be considered.

NRC Regulations

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

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

Preclosure safety analysis is a systematic examination of the site, the design, and the potential hazards and initiating events as well as the resulting event sequences and potential radiological exposures to workers and the public. The regulations require DOE to identify the event sequences that might lead to radiological exposures. An event sequence means a series of actions or occurrences within the natural and engineered components of a geologic repository operations area that could potentially lead to exposure of individuals to radiation. An event sequence includes one or more initiating events and associated combinations of repository system component failures, including those produced by the action or inaction of operating personnel. The regulations classify the event sequences by two broad categories: Category 1 and Category 2. Those event sequences expected to occur one or more times before permanent closure of the geologic repository operations area are referred to as Category 1 event sequences. Consistent with the EPA final standards, Category 1 event sequences are

limited to an annual individual dose of 0.15 mSv/year [15 mrem/yr] for members of the public from normal operations at the repository. Other event sequences that have at least 1 chance in 10,000 of occurring before permanent closure are referred to as Category 2 event sequences. The repository operations area is to be designed such that any Category 2 event sequence (i.e., those event sequences representing off-normal or accident conditions) will not result in an individual dose larger than 0.05 Sv [5 rem] at or beyond the area boundary. The analysis of a specific Category 2 design basis event would include an initiating event and the associated combinations of repository system or component failures that can potentially lead to exposure of individuals to radiation. An example design basis event is a postulated earthquake (the initiating event) that results in (i) failure of a crane lifting a spent nuclear fuel waste package inside a waste handling building, (ii) damage to a building ventilation (filtration) system, (iii) drop and breach of a waste package, (iv) damage to spent nuclear fuel, (v) partitioning of a fraction of the radionuclide inventory to a building atmosphere, (vi) release of some radioactive material through a damaged ventilation (filtration) system, and (vii) exposure of an individual (either a worker or a member of the public) to the released radioactive material.

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

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

- DOE is required to consider uncertainty in its representation of the repository (uncertainty and variability in parameter values must be taken into account) and the events that can happen (consideration of potentially disruptive events with a probability of occurrence as low as 1 chance in 10,000 of occurring during 10,000 years). Also, DOE must provide further assurances that uncertainty in the information (e.g., evaluation

of site characterization data) used to develop the performance assessment has been evaluated by considering alternative conceptual models of features and processes that are consistent with available data and current scientific understanding. DOE also must supply its basis for including or excluding features, events, and processes that significantly affect performance.

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

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

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

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

Confidence in the safety of the repository can be increased further by a program of continued investigation of repository performance (i.e., performance confirmation program). The regulations provide for a performance confirmation program to confirm the assumptions, data, and analyses that led to the findings that permitted construction of the repository and subsequent emplacement of the wastes. General requirements for the performance

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

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

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

Identification of the NRC Policy Issues

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

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

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

2.2 Risk-Informing NRC Reviews

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

> LICENSEE RESPONSIBILITY embodies the principle that, although the U.S. Nuclear Regulatory Commission is responsible for developing and enforcing the standards governing the use of nuclear installations and materials, *it is the licensee who bears the primary responsibility for conducting those activities safely. The U.S. Nuclear Regulatory Commission's role is not to monitor all licensee activities but to oversee and audit them* [emphasis added]. This allows the agency to focus its inspection, licensing, and other activities on those areas where the need, and the likely safety and safeguards benefit, is [sic] greatest.

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

A risk-informed approach focuses the NRC prelicensing reviews on topics that, among other factors, are major potential contributors to safety or, alternatively, are likely to contribute most to risk reduction. These topics are selected based on information presented by DOE, independent staff investigations, published information, and experience gained through attending meetings of review committees and participating in site visits. The staff has developed a baseline of risk insights (Appendix D) to risk-inform their review. In its preclosure integrated safety analyses and postclosure performance assessments, DOE demonstrates major potential contributors to safety. Combined with the NRC staff independent analyses, these DOE analyses provide a reasonable framework for selecting items of high importance to system safety and waste isolation and, therefore, that should be subjected to a more thorough NRC review. This

approach of risk-informing reviews directly helps to meet the NRC strategic goal to enhance effectiveness, efficiency, realism, and timeliness.

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

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

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

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

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

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

The risk insights baseline (Appendix D) presents the current perspective of the NRC staff on the important parameters, models, and assumptions. The risk insights also reflect uncertainties in understanding the features, events, and processes relevant to waste isolation at Yucca Mountain. Generally, important uncertainties are addressed in a total system performance assessment through a variety of approaches such as parameter ranges (e.g., range of

retardation factors of radionuclides in alluvium) and conservative modeling (e.g., assume southerly blowing wind direction for igneous activity). The risk insights provide a basis for focusing on the more important technical issues relative to risk and indicate where staff can benefit most from additional information (e.g., reduction of uncertainty in dose estimates).

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

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

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

2.3 Preclosure and Postclosure Assessment Processes

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

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

(1) <u>Model Integration</u>

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

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

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

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

- Has sufficient justification been provided for any excluded coupling?

(2) Data and Model Justification

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

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

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

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

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

(3) Data Uncertainty

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

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

- Are any bounding assumptions technically defensible?

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

- Have any correlations between parameter values been appropriately considered?

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

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

(4) Model Uncertainty

- Has DOE considered plausible alternative models?
- Has DOE provided supporting information for the conceptual model(s) used in the safety case?
- Are the intermediate outputs of the engineered and natural system models produced by DOE consistent with the selected conceptual model(s)?

(5) Model Support

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

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

2.4 Updating the Integrated Issue Resolution Status Report

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

It is the responsibility of DOE to demonstrate compliance with the regulations at 10 CFR Part 63. The NRC will review the entire application to determine if DOE has satisfied the regulatory requirements. The Yucca Mountain Review Plan (NUREG–1804, NRC, 2003), the risk insights baseline (Appendix D), and the information contained in this update of this report will form the bases for the NRC staff to conduct a risk-informed review of a DOE license application for a potential repository at Yucca Mountain. The risk insights baseline will help focus the staff review, by guiding the depth of the staff review in particular areas, and helping develop requests for additional information. The relevance of the risk insights baseline is dependent on the DOE repository design and performance assessment approaches presented in the license application; however, the staff independent analyses provide additional

confidence for review of the strengths and limitations of the DOE demonstration of compliance. This approach is consistent with the NRC policy regarding risk-informed, performance-based regulations in which risk insights, engineering analysis, expert judgment, the principle of defense-in-depth, and safety margins are incorporated in licensing decisions.

It is emphasized that this update to the Integrated Issue Resolution Status Report tracks progress toward issue resolution during prelicensing interactions with the NRC staff. With a few exceptions as noted, the review is based on information available by March 2004. It is not a licensing review, and no conclusions are drawn with respect to whether or not the Yucca Mountain site is licensable or whether it meets applicable NRC regulatory requirements. The licensing review will begin only after a potential license application is submitted, and the NRC staff review will be documented in a safety evaluation report.

2.5 References

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3 GENERAL INFORMATION

3.1 General Description

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

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

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

3.3 Physical Protection Plan

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

3.4 Material Control and Accounting Program

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

3.5 Description of Site Characterization Work

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

4 REPOSITORY SAFETY BEFORE PERMANENT CLOSURE

4.1 Preclosure Safety Analysis

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

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

- A general description of the structures, systems, components, equipment, and process activities at the geologic repository operations area.[1]
- An identification and systematic analysis of naturally occurring and human-induced hazards at the geologic repository operations area, including a comprehensive identification of potential event sequences.
- Data pertaining to the Yucca Mountain site, and the surrounding region to the extent necessary, used to identify naturally occurring and human-induced hazards at the geologic repository operations area.
- The technical basis for either inclusion or exclusion of specific, naturally occurring, and human-induced hazards in the safety analysis.
- An analysis of the performance of the structures, systems, and components to identify those that are important to safety. This analysis identifies and describes the controls relied on to limit or prevent potential event sequences or mitigate their consequences. This analysis also identifies measures taken to ensure the availability of safety systems.
- A description and discussion of the design, both surface and subsurface, of the geologic repository operations area, including the relationship between design criteria and the requirements specified at 10 CFR Part 63.111(a) and (b); and the design bases and their relation to the design criteria.

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

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

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

[2] *Event sequence* means a series of actions and/or occurrences within the natural and engineered components of a geologic repository operations area that could potentially lead to exposure of individuals to radiation. An event sequence includes one or more initiating events and associated combinations of repository system component failures, including those produced by the action or inaction of operating personnel. Those event sequences expected to occur one or more times before permanent closure of the geologic repository operations area are referred to as Category 1 event sequences. Other event sequences that have at least 1 chance in 10,000 of occurring before permanent closure are referred to as Category 2 event sequences.

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

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

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

4.1.1 Site Description As It Pertains to Preclosure Safety Analysis

4.1.1.1 Areas of Review

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

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

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

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

This section of the Integrated Issue Resolution Status Report addresses assessment of the Yucca Mountain site description. Site description comprises (i) site geography, (ii) regional demography, (iii) local meteorology and regional climatology, (iv) regional and local surface and ground water hydrology, (v) site geology and seismology, (vi) igneous activity, (vii) site geomorphology, and (viii) site geochemistry. Adequacy of the site description is assessed based on information necessary for DOE to conduct its preclosure safety analysis and geologic repository operations area design. Section 2.1, Regulations Applicable to a Potential High-Level Waste Repository at Yucca Mountain, of this report discusses the methodology used by staff for this review.

4.1.1.2 Staff Review of Available Information

The DOE site description is primarily documented in Civilian Radioactive Waste Management System Management and Operating Contractor (CRWMS M&O) (2000a) and in DOE (1999a). Since then, DOE has published a detailed description of the geotechnical data used for preclosure earthquake ground motion analysis (Bechtel SAIC Company, LLC, 2002a). A summary evaluation of the geotechnical data from Bechtel SAIC Company, LLC (2002a) is provided in Section 4.1.1.2.5, Site Geology and Seismology. These reports, plus additional supporting DOE documents identified in the appropriate subsections that follow, are reviewed to the extent they contain site description information relevant to the preclosure safety analysis. Much site description information also pertains to repository safety after permanent closure and, where appropriate, this review cross-references appropriate sections of the postclosure review contained within this report. In addition, this preclosure review incorporates information previously evaluated within the key technical issue framework, including these Key Technical Issues: (i) Igneous Activity, (ii) Structural Deformation and Seismicity, (iii) Evolution of the Near-Field Environment, (iv) Thermal Effects on Flow, (v) Repository Design and

Thermal-Mechanical Effects, (vi) Unsaturated and Saturated Flow under Isothermal Conditions, and (vii) Total System Performance Assessment and Integration.

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

The Yucca Mountain site rests primarily within the westernmost parts of the Nevada Test Site. Parts of the potential repository are also within the Beatty District of the public lands administered by the Bureau of Land Management and U.S. Air Force (Nellis Air Force Range). The nearest population centers are Beatty, Nevada {28 km [17 mi] to the west-northwest}; Amargosa, Nevada {24 km [15 mi] to the south}; Pahrump, Nevada {83 km [52 mi] to the south-southeast}; and Las Vegas, Nevada {142 km [88 mi] to the east-southeast}.

Review of the site description is organized according to the review methods and associated review criteria identified in U.S. Nuclear Regulatory Commission (NRC) (NRC, 2003). These eight review methods are organized around general subsections of the site description identified in 10 CFR 63.112(c). The potential DOE license application should contain a description of the site geography adequate to permit evaluation of the preclosure safety analysis and the geologic repository operations area design.

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

4.1.1.2.1 Site Geography

4.1.1.2.1.1 Site Location

Yucca Mountain is located in Nye County, Nevada, approximately 142 km [88 mi] west-northwest of Las Vegas. The potential repository site would be on land controlled by the U.S. Air Force (Nellis Air Force Range), the DOE Nevada Test Site, and the U.S. Bureau of Land Management. The geographic location of the potential high-level waste repository at Yucca Mountain, Nevada, is adequately identified in CRWMS M&O (2000a).

4.1.1.2.1.2 Significant Natural and Manmade Features

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

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

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

4.1.1.2.1.3 Site Maps

CRWMS M&O (2000a) and DOE (2004) contain maps that adequately show (i) Yucca Mountain (Figures 1.1-1, 2.2-1, 2.2-2, 2.2-3), (ii) physiography (Figures 1.2-1 and 2.2-4), (iii) facilities and infrastructure, (iv) preclosure controlled area, and (v) potential withdrawal area. The maps and information conveyed are adequate to identify these features with regard to preclosure safety assessment in a potential license application.

4.1.1.2.2 Regional Demography

The regional demography is reviewed in CRWMS M&O (2000a) and DOE (1999a). In CRWMS M&O (2000a), population estimates are based principally on demographic reports (Nevada State Demographer, 1999a,b,c), and on estimates made by CRWMS M&O (1998a)

and by the U.S. Census Bureau (1996, 1993). These data are for the estimated population in 1998. The regional demographics are inadequate as they are based on outdated population estimates. DOE estimates in the potential license application should use the most recent census data compiled, such as the 2000 census or later census data.

4.1.1.2.3 Local Meteorology and Regional Climatology

4.1.1.2.3.1 Climate and Meteorological Conditions

The modern climatic and meteorological conditions at Yucca Mountain are influenced by a broad range of atmospheric conditions including global-scale processes, regional weather patterns, seasonal variations, and local topographically controlled weather patterns (CRWMS M&O, 2000a). The current climate in Central and Southern Nevada is generally arid to semiarid because of modern regional weather patterns, far-away moisture sources such as the Pacific Ocean (including the Gulf of California) or the Gulf of Mexico, and the numerous mountain ranges between Yucca Mountain and these moisture sources. The degree of aridity varies in space, mostly by elevation, and in time, seasonally and annually.

Present-day climate and meteorological conditions are discussed in CRWMS M&O (2000a). Discussions on the local meteorology are based on data acquired by the onsite meteorological monitoring network operated by the Yucca Mountain Radiological and Environmental Programs Department and selected regional National Oceanic and Atmospheric Administration meteorological stations (CRWMS M&O, 2000a). Information on the large-scale climatic factors affecting the Yucca Mountain area was obtained from textbooks and scientific literature as described in the CRWMS M&O (2000a).

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

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

As reported in CRWMS M&O (2000a, Table 6.2-11), the extreme minimum and maximum temperatures measured at the Radiological and Environmental Programs Department site located at Yucca Mountain were −12.5 °C [9.5 °F] and 39.9 °C [103.8 °F]. The monthly mean minimum and maximum temperatures for this measuring station were 2.9 °C [37.2 °F], which was calculated for the months of January and December, and 32.3 °C [90.1 °F], which was

calculated for the month of July. In addition, the yearly average number of days above 32 °C [90 °F] is 38.2 and, conversely, the average number of days below freezing {i.e., 0 °C [32 °F]} is 25.4 (CRWMS M&O, 2000a, Table 6.2-11). These temperatures are derived from measurements taken over a 12-year period, from 1986 to 1997.

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

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

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

4.1.1.2.3.2 Precipitation and Flooding

Precipitation is characterized in Section 6.2.3.1 of CRWMS M&O (2000a). Tables 6.2-3 and 6.2-4 summarize the precipitation statistics for five stations at and near Yucca Mountain; Tables 6.2-10 to 6.2-18 provide monthly and annual climatological summaries, including precipitation, for the local weather stations one to nine, within the Radiological and Environmental Programs Department Sites; Table 6.2-20 provides monthly climatology summaries for regional weather stations; Table 8.2-4 summarizes the annual precipitation for the National Weather Service Stations between 1921 and 1947; and Table 6.2-25 summarizes the annual precipitation for the National Weather Service Stations between 1948 and 1995. Average precipitation for Yucca Mountain ranges between 174 and 195 mm/yr [7 and 8 in/yr] compared with the 254 mm/yr [10 in/yr] average for the region with only 105 mm/yr [4 in/yr] in the Amargosa farms area. Average precipitation values are based on 30-year records.

Flooding is discussed in Section 7.3 of CRWMS M&O (2000a). This section summarizes local and regional flood studies in southern Nevada, as well as local studies in the Yucca Mountain region. Results of hydrologic engineering studies started in 1999 have not yet been reported by DOE or its contractors. The staff notes, however, that summaries of data from nearby regional meteorological stations, including the Amargosa Farms, Jackass Flat, and Area 12 Mesa, are not included, despite their relatively long rainfall records. The relative close proximity of Site 9 (Radiological and Environmental Programs Department Site), Jackass Flat, and Amargosa

Farms meteorological stations would provide additional support for meteorological data and models. DOE should provide this information in a potential license application.

4.1.1.2.3.3 Severe Weather

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

4.1.1.2.4 Regional and Local Surface and Ground Water Hydrology

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

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

This section reviews the characterization and analyses of surface and ground water interaction with the potential repository design. The focus is proportionately on features deemed to be structures, systems, and components important to safety. Accordingly, evaluation is needed for the (i) flood potential, catastrophic erosion, and drainage design for the facilities, systems, and components; (ii) flood and catastrophic erosion near expected transportation pathways, particularly near wash channels; and (iii) design modification and standoff distances from known faults crossing emplacement drifts and access tunnels. These three items are discussed in the context of Surface Waters and Ground Water.

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

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

4.1.1.2.4.1 Surface Waters

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

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

Multiple ventilation and exhaust shafts are part of the current potential repository design (McDaniel, 2004). The ventilation system consists of seven exhaust shafts, four inlet shafts, and the north and south portals of the Exploratory Studies Facility (McDaniel, 2004). The shafts will be vertical with sweeps near the ground surface. Based on CRWMS M&O (2000b), the shafts will not be sited over nor have direct pathways vertically to emplacement drifts. Hence, the safety concern is with operation of the ventilation systems and flooding of localized zones in the tunnels. The primary concern with the ventilation system is that the surface intersection of the shafts should avoid channels and side slopes prone to enhanced runoff and erosion. Shafts on the crest and east flanks of Yucca Mountain pose little risk because the catchment areas are small. It is difficult to determine the topographic locations of shafts northeast of Drill Hole Wash from McDaniel (2004). In addition to a clear delineation of topographic position, DOE should provide more detailed information on flood probabilities for the surface intersections of shafts northeast of Drill Hole Wash.

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

Direct precipitation during intense storms could lead to a flooding of facilities buildings and components. Drainage from the radiological control area will include an underground storm drainage system designed to protect this portion of the pad from a probable maximum flood (DOE, 2001). The drainage system for the remainder of the facility will be designed to handle the 100-year flood. CRWMS M&O (2000c) mentions roofs will be designed to withstand a 100-year precipitation event. The north portal itself will be protected by construction of open channels to divert water (DOE, 2001). The drainage design for the north portal pad is tied to the flood mitigation from washes in Midway Valley (part of the pad being below the 100-year flood area). Justification or clarification is needed for the implied use of the 100-year precipitation event for the critical buildings on the pad, specifically the Waste Handling and Waste Treatment Buildings.

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

Probable maximum flood is defined as the maximum runoff condition resulting from the most severe combination of hydrologic and meteorologic conditions considered reasonably possible for the drainage basin being studied. Probable maximum flood is derived using the probable maximum precipitation. A 100-year flood is the flood derived from a precipitation event having a recurrence interval of 100 years. By definition, recurrence interval is not associated with a probable maximum precipitation or flood.

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

CRWMS (2000b) and DOE (1995) refer to the results of Bullard (1992) and the addition of the bulking factor by Blanton (1992) in discussing probable maximum floods that might affect repository facilities. Portions of the north portal pad are within the area of the probable maximum flood. CRWMS M&O (2000c) and DOE (2001) note that critical buildings and systems, such as the Carrier Preparation Building, the Waste Handling Building, and the Waste Treatment Building, will be above the probable maximum flood. The rest of the facility buildings on the pad near the north portal will be designed to withstand the 100-year flood, which is implied to be approximately 2.5 feet below the probable maximum flood level (DOE, 2001). The choice of the 100-year flood leaves flooding as borderline between a category 1 or 2 design consideration (CRWMS M&O, 2000c) using probability criteria, but category 2 is selected (DOE, 2001). However, 10 CFR Part 63 specifies the use of event sequences for categorization

of initiating events. Thus, additional description of potential event sequences for flooding is needed for the portion of the north portal area below the probable maximum flood.

Muck piles developed during excavation of the drifts are currently sited in Midway Valley (McDaniel, 2004; CRWMS M&O, 1999, 1998b). Midway Valley has been in a state of sediment aggregation during modern climate conditions. There is little incision from ephemeral stream flow off the east flank of Yucca Mountain. A muck pile extending from approximately the south portal to the north portal might lead to a focusing of stream flow from Split, Coyote, Wren, and Drill Hole Washes. Coalescing stream flow into Midway Valley could incise and possibly erode facility systems. McDaniel (2004) noted several smaller muck piles south and east of the north portal of the Exploratory Studies Facility. DOE should provide criteria for siting the muck piles that include consideration of flooding.

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

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

4.1.1.2.4.2 Ground Water

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

Focused, fast pathway, downward percolation may occur along large fracture systems and faults (e.g., Drill Hole Waste fault). The chemistry of the perched water body and of the aquifer beneath Yucca Mountain suggests the likelihood of recharge by fast-pathway water flowing through faults and fractures. Portions of the repository access tunnels and emplacement drifts will intersect faults or underlie faults that cut the nonwelded Paintbrush tuffs. These areas may be prone to elevated water influx during periods of flooding at the ground surface. If standard mining practices that would alleviate the problems are to be used, then information on these or alternative practices should be given.

The depth of the aquifers and perched water beneath the site and the recharge and discharge features have been adequately described in CRWMS M&O (2000b). Evidence of past water table positions suggests maximum elevations in the repository footprint of 120 m [394 ft] above present-day elevations (CRWMS M&O, 2000b). Perched water has been found at the base of

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

4.1.1.2.4.3 Summary

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

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

4.1.1.2.5 Site Geology and Seismology

4.1.1.2.5.1 Site Geology

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

4.1.1.2.5.2 Regional Geologic Setting

As noted by DOE (CRWMS M&O, 2000a), Yucca Mountain lies within the Central Basin and Range physiographic province of the North American Cordillera. The region is characterized by complex interactions of strike-slip (shear faults) and extensional deformation, active since onset of the Cenozoic (65 million years). The region remains tectonically active as indicated by numerous Quaternary (in last 2 million years) faults [including evidence for Holocene (last 10,000 years) activity], historic seismicity (including the 1992 Little Skull Mountain earthquake activity), and volcanism (punctuated by the most recent volcanic eruption at Lathrop Wells Cone approximately 80,000 years ago).

In its description of geologic setting (CRWMS M&O, 2000a), DOE adopts a segmented regional framework in which the region is divided into three tectonic domains. Each tectonic domain is a structurally bounded section of Earth crust with relatively similar deformational characteristics within the domain compared with markedly different deformational characteristics in adjacent domains. These domains are the Walker Lane domain, which includes the site; the Basin and Range domain, which includes the areas to the north and east; and the Inyo-Mono domain, which includes regions to the west and south.

The stratigraphy of the geologic setting consists of igneous, sedimentary, and volcanic rocks that range in age from Proterozoic (2,500 million years) to the present. Pre-Cenozoic rocks (before 65 million years), which constitute the basement rocks of the regional geologic setting, primarily consist of Precambrian and Early Cambrian (approximately 2,500 to 500 million years) silica-rich strata overlain by a thick Paleozoic (approximately 500–245 million years) section of limestones and dolomite. The regional carbonate aquifer is within these Paleozoic limestones and dolomites. Cenozoic rocks of the Yucca Mountain geologic setting fall into three general groups: (i) pre-Middle Miocene (>16.5 million years) strata (including volcaniclastics) that predate the southwestern Nevada volcanic field, (ii) Middle to Late Miocene (16.6–5.3 million years) volcanic rocks that compose the southwestern Nevada volcanic field, and (iii) Plio-Pleistocene (5.3 million years to the present) basalts and valley fill sediments. The Cenozoic rocks overlie complexly deformed Paleozoic and Precambrian rocks on a regional erosional unconformity, suggesting significant uplift and erosion of the pre-Cenozoic rocks associated with extensional tectonics of the Basin and Range.

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

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

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

Since the 1999 staff review and summary of the site description (CRWMS M&O, 2000a), additional aeromagnetic data were acquired (Blakely, et al., 2000). Although the intent of the aeromagnetic data was to assess potential ground water flow paths in the Amargosa Desert and Death Valley, these new data also provide important new information on the regional geologic setting features such as faults and possible small basaltic volcanoes now buried within the thick accumulations of alluvial material in the basins.

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

4.1.1.2.5.3 Regional Tectonic Setting

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

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

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

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

Staff reviewed the development and application of tectonic models in postclosure performance assessments (including development of the probabilistic seismic hazard assessment) and have classified the subissue as closed for prelicensing (see Section 1.2 for definition of closed) (NRC, 1998). DOE has sufficient information for staff review with regard to the postclosure aspects of seismic and faulting hazards analyses. In that assessment, staff recommended that (i) the full range of tectonic models, as presented in the probabilistic seismic hazard assessment (CRWMS M&O, 1998c; Stepp, et al., 2001), should be applied uniformly and with continuity across the entire DOE analysis of Yucca Mountain, as appropriate; (ii) classification of specific models as preferred or favored should be avoided without full justification; and (iii) DOE should continue to evaluate new scientific information with regard to the regional tectonics as necessary. These recommendations also apply to the site descriptions of regional tectonic models as they relate to preclosure safety analyses.

The DOE findings (CRWMS M&O, 2000a) about the site geology are consistent with the regional tectonic models described in previous staff reviews (e.g., NRC, 1999a). In addition, the DOE review provides a comprehensive summary of data, results, and interpretations of tectonic models similar to previous staff reviews (e.g., NRC, 1999a). Thus, the DOE regional tectonic model summary adequately describes the technical bases for the descriptive and process

models used to assess the ability of the natural system to help meet preclosure safety performance objectives.

Since the 1999 staff review and summary of the site description (CRWMS M&O, 2000a), there are several new published regional reconstructions of Basin and Range extension (e.g., Snow and Wernicke, 2000). DOE should evaluate the new tectonic models as to their impact on DOE's current understanding of the site geology.

4.1.1.2.5.4 Quaternary Stratigraphy and Surficial Processes

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

Eight Quaternary alluvial units were recognized within the Yucca Mountain region (U.S. Geological Survey, 1996). These alluvial units range in age from 1.6 Ma to the present. The alluvial stratigraphy forms the basis for many paleoseismic interpretations in which ages and amounts of fault displacements were determined from relative juxtapositions of the eight alluvial units across active fault zones. This information was used by the DOE expert elicitation panel in its construction of the Yucca Mountain probabilistic seismic hazard assessment (CRWMS M&O, 1998c; Stepp, et al., 2001). Results from the probabilistic seismic hazard assessment are used for both postclosure performance assessment and preclosure safety analysis and as input to the preclosure seismic design.

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

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

DOE has not provided information that describe how these site data coupled with the results of the probabilistic seismic hazard assessment (CRWMS M&O, 1998a) will be used to develop

preclosure seismic response spectra and preclosure seismic design values. DOE originally established a timetable for the release of two documents for that purpose. The Seismic Design Inputs Report was to be made available in September 2001 and the Seismic Topical Report 3 in fiscal year 2002 (Schlueter, 2000; Reamer, 2001). The DOE schedule for release of that information has been delayed and changed. The current plans (Ziegler, 2004) indicate that DOE will instead release three documents to support site response and seismic design. Technical Basis Document No. 14: Low Probability Seismic Events will develop the preclosure seismic design and fault displacement inputs and the seismic hazard results used for postclosure performance assessment. DOE plans indicate that detailed information on seismic design inputs will be provided in upcoming documents. Topical Report 2 (DOE, 1997a) will be revised to incorporate the new seismic information and to update the seismic design methodology to be consistent with the performance objectives in 10 CFR Part 63.

In summary, staff consider this portion of the site description closed, pending submission of the necessary and promised information from DOE. Details of the application of DOE information on preclosure hazard assessments from natural surficial processes are provided within their respective sections of this report.

4.1.1.2.5.5 Site Stratigraphy

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

Yucca Mountain comprises a thick accumulation of volcanic tuff deposited on an irregular surface of eroded and deformed Paleozoic and Precambrian basement composed of highly faulted and folded sedimentary and metasedimentary rocks. These tuffs were erupted from a series of Middle to Late Miocene (15–9 million years) calderas that collectively form what has been defined as the Southwestern Nevada Volcanic Field. Sawyer, et al. (1994) provide the most recent comprehensive regional stratigraphy of the Miocene volcanic rocks in the Yucca Mountain region. Rocks of the Paintbrush Group, principally Tiva Canyon Tuff (12.7 million years), make up the main surface exposures of Yucca Mountain, whereas the repository horizon is within the Topopah Spring Tuff (12.8 million years). The Paintbrush Group tuffs rest on a sequence of older tuffs, including the Prow Pass and Bullfrog members of the Crater Flat Group.

Because of their importance for understanding geologic systems at Yucca Mountain, the volcanic rocks have been a major focus of stratigraphic studies being conducted as part of the site characterization program. Many investigations of the Yucca Mountain area have focused on mapable, lithostratigraphic, hydrogeologic, and thermal-mechanical properties of the tuffs. Each type of investigation has led to its own stratigraphic nomenclature (Scott and Bonk,1984; Buesch, et al., 1996; Flint, 1998; Ortiz, et al., 1985). Table 4.5-3 of CRWMS M&O (2000a) provides a cross-correlation of these different stratigraphic units. Different compositions of the igneous magma, eruption types (effusive versus explosive), cooling histories, and transport and

deposition mechanisms combine to produce the range of depositional features observed in the Yucca Mountain strata.

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

CRWMS M&O (2000a) and numerous references therein provide a detailed and comprehensive summary of the site stratigraphic work. The DOE regional geologic setting summary adequately describes the technical bases for the site stratigraphy used to assess the ability of the natural system to help meet preclosure safety performance objectives. However, DOE computer models of unsaturated zone flow, transport, and rockfall that reflect attributes of the stratigraphy often lump or generalize rock units and rock data (such as repository horizon, thickness, percent vitric verses percent zeolitic). This confuses between-model comparisons. For example, some models use data from thermal-mechanical unit TSw2 (Ortiz, et al., 1985) to represent the repository host horizon, but the repository host horizon includes the additional thermal-mechanical unit Tptpul. Also, the uncertainties associated with thickness and lateral continuity of key strata (e.g., Ptn, Chv, Tptpll) should be consistently explained for each model that abstracts data from these strata. For consistency, DOE should correlate the modeled stratigraphic or thermal mechanical units with the developed lithostratigraphy (e.g., Buesch, et al., 1996) for each model that assumes or uses stratigraphic data.

4.1.1.2.5.6 Site Structural Geology

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

Younger tuffs related to the Timber Mountain Group are locally exposed at Yucca Mountain in topographic lows between large block-bounding faults. This observation, along with evidence for growth faults in the Paintbrush rocks in Solitario Canyon (e.g., Carr, 1990; Day, et al., 1997), suggests that faulting and tuff deposition were synchronous at Yucca Mountain. Trenching studies of the Solitario, Paintbrush Canyon, and Bow Ridge faults also show sufficient evidence

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

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

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

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

An *en echelon* pattern of faulting is best expressed along the western edge of Yucca Crest and the fault line escarpment that follows the west-dipping Solitario Canyon, Iron Ridge, and Stagecoach Road faults (e.g., Simonds, et al., 1995). The geometry of faults and ridges defines a scallop trend composed of linear, north-trending fault segments connected by discrete curvilinear northwest-trending fault segments. For example, the ends of the northwest-trending curvilinear Iron Ridge fault bend to the northwest near its overlap with both the Stagecoach Road and Solitario Canyon faults. Yucca Mountain also contains numerous swarms of small northwest-trending faults that connect the large north-trending faults. One example is at West Ridge, which is cut by numerous small faults that connect segments of the Windy Wash and Fatigue Wash faults. This geometry strongly suggests that the entire Yucca Mountain fault system is an *en echelon* branching fault system (Ferrill, et al., 1999) in which slip on the large block-bounding fault triggers relatively widespread, but predictable, secondary deformation on connecting and linking faults. Linkage of the *en echelon* system is either by lateral propagation of curved fault tips or formation of connecting faults that breach the relay

ramps (Ferrill, et al., 1999, Figure 1; Peacock and Sanderson, 1994; Trudgill and Cartwright, 1994). More importantly, from this interpretation of *en echelon* faulting, it follows that locally developed faults and fractures were produced by local variations of the stress field (e.g., Crider and Pollard, 1998) rather than dramatic swings of the regional extension direction (Throckmorton and Verbeek, 1995). The amount, orientation, and degree of faulting directly depend on the relative position of the rock within the *en echelon* fault system, either in relay ramps that connect overlapping *en echelon* fault segments or in the hangingwall or footwall blocks of the block-bounding faults.

Fracturing of the volcanic rocks at Yucca Mountain started soon after deposition of the volcanic tuffs about 11–13 million years ago. The first fractures of the volcanic rocks were probably cooling fractures (also commonly referred to as cooling joints). Soon after deposition of the tuffs, tectonic and gravitational forces caused additional fracturing of the tuffs. Cooling, tectonic, and unloading fractures constitute the naturally occurring fracture system at Yucca Mountain. Because the region is tectonically and geomorphically active, both tectonic and unloading joints continue to form. Manmade fractures in drifts at Yucca Mountain are also present, formed by excavation of the tunnels and drifts. DOE should consider the effects of excavation-induced fractures or other skin effects in pre- and postclosure performance analyses, especially if risk-significant processes are effected (e.g., rockfall). As discussed in the preceding paragraphs, faults are also prominent features of the structural framework at Yucca Mountain. Small faults and shear joints (up to meters in length and of small displacement) grade upward in scale to large features (hundreds of meters, in the case of joints, and tens of kilometers, in the case of faults). NRC (1999a) provides a comprehensive discussion of fractures and fracture studies at Yucca Mountain.

For preclosure safety analysis, the most critical aspect of fracture characterization is the statistical representation of the various fracture sets. The statistical properties of fractures (most notably fracture intensity and orientation) are used to assess the stability of subsurface openings and potential rockfall characteristics, especially the size of rock blocks that may fall on the waste packages. Azimuthal orientation of the drifts within the potential repository is optimized to ensure that large unstable blocks are minimized (i.e., drifts perpendicular to the dominant fracture orientation). DOE has not fully utilized fracture data in preclosure repository design, especially within the larger lithophysal units of the Topopah Spring Tuff. Most fracture studies have focused on the middle nonlithophysal units of the Topopah Spring Tuff (e.g., Gritto, et al., 2004).

The staff analyses (e.g., NRC, 1999a) have shown that characterization of fracture networks at Yucca Mountain is impaired by several important sampling biases common to fracture analyses. If left uncorrected, these sampling biases lead to underrepresentation of fracture intensity and misrepresentation of fracture-set orientations. For example, because of the limited diameter of the Exploratory Studies Facility {7 m [23 ft]}, the lengths of the longest fractures are often unconstrained. The ends of the fracture are simply obscured in unexposed rock. In addition, the orientation of a one-dimensional sampling line (e.g., borehole or detailed line survey scanline) or two-dimensional sampling surface (e.g., pavement, roadcut, or tunnel surface) inherently biases sampling against discontinuities parallel to the sampling line or surface and in favor of sampling discontinuities at a high angle to the sampling line or surface. Mathematical corrections (Terzaghi, 1965) can partially compensate for this sampling bias. Finally, because measuring every fracture from the microscale to megascale is impractical or impossible for large sample areas, fracture studies usually invoke a size (e.g., length) cutoff. This was commonly

1 m [3 ft] in the Yucca Mountain studies. Fractures smaller than that cutoff dimension are not routinely counted. Consequently, small fractures are underrepresented in fracture characterizations. Exclusion of small fractures can skew fracture-intensity determinations.

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

Of particular importance to preclosure safety and design is the potential for sampling bias of fracture orientations. For example, DOE developed a drift layout plan of the potential repository (azimuths of drifts) based on assumptions of the measured fracture orientations at Yucca Mountain. DOE wants to minimize block volumes of potential rockfalls by aligning the drifts perpendicular to the azimuth of the dominant fracture set. The staff has previously commented that the statistical representation of fracture orientations, based on the measured fractures at Yucca Mountain may contain a sampling bias such that the actual fracture orientations are different from those used in the DOE design calculation (NRC, 1999a). DOE agreed to provide that information prior to submitting a potential license application (Schlueter, 2000). While the DOE structural geology summary has not yet adequately described the technical bases for the descriptive and process models used to assess the ability of the natural system to help meet preclosure safety performance objectives, DOE has agreed to provide the needed information. Overall, the available information, along with key technical issue agreements between DOE and NRC (Section 5.1.3.2.4.1), is sufficient to expect that the information necessary to assess site structural geology with respect to preclosure safety will be available at the time of a potential license application.

4.1.1.2.5.7 Seismic Hazard Assessment

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

Seismic Source Characterization. In the seismic source characterization, six teams of experts were used. Each team consisted of three specialized geoscientists with expertise in either paleoseismology, Basin and Range structural geology, or Basin and Range seismology. To assess seismic sources, the teams relied on information provided by the U.S. Geological Survey; DOE; other project-specific Yucca Mountain studies; and published geological, geophysical, and seismological literature. In addition, the teams were assembled for six workshops, held between April 1995 and June 1997, at which the experts exchanged information on seismic sources and participated in additional discussions with other external experts. Details of the workshops are given in CRWMS M&O (1998c).

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

relied, to varying degrees, on two tectonic models: (i) seismogenic detachment faults as potential seismic sources (i.e., Deep Detachment Fault Tectonic Model) and (ii) hidden or buried strike-slip faults with associated cross-basin faults as potential seismic sources (i.e., Amargosa Desert Fault Model). In addition, planar-block bounding faults were also considered in the assessments made by the six expert elicitation teams. Although presented to the experts at the workshops, strain rate values derived from global positioning satellite measurements were not explicitly considered by any teams as a viable alternative to estimations of the seismic hazard.

Seismic sources in CRWMS M&O (1998c) and in Stepp, et al. (2001) consisted of two types, fault sources and areal source zones. The approach used by DOE to identify potential seismic sources follows standard practice for seismic hazard assessments of sites west of the Mississippi River, where better exposure of bedrock and greater tectonic activity make identification of fault sources easier to discern.

Fault sources are used in the hazard assessment to account for expected seismicity on known or suspected fault traces. Uncertainty in fault sources is accounted for by alternative interpretations of fault length, fault dip, closest approach to the site, depth within the seismogenic crust, and possible kinematic linkage with other faults. In the probabilistic seismic hazard analysis calculations, earthquakes are assumed to occur randomly along the fault surface, constrained by the size of the rupture area. Rupture area and rupture dimensions are specified by empirical relationships based on magnitude (e.g., Wells and Coppersmith, 1994).

Fault sources were identified by the expert teams from published U.S. Geological Survey and DOE maps and reports (U.S. Geological Survey, 1996; Piety, 1995; Anderson, et al., 1995a,b; Simonds, et al., 1995), published scientific literature (Scott, 1990; Zhang, et al., 1990; Reheis and Dixon, 1996; Reheis and Sawyer, 1997), and CNWRA publications (Ferrill, et al., 1996; McKague, et al., 1996). In addition, the experts benefitted from detailed discussions at several of the probabilistic seismic hazard analysis workshops, in which summaries of fault sources and tectonic models were presented by project and external scientific experts. The expert teams also visited many of the sources during a field trip held as part of Probabilistic Seismic Hazard Analysis Workshop 3.

Local and regional Yucca Mountain tectonics also were considered when identifying potential fault sources. Considerations included sources from proposed buried or otherwise cryptic strike-slip faults (Schweickert and Lahren, 1997) and seismogenic detachment faults (Wernicke, 1995). Uncertainty in the sources, both in geometric characteristics and likelihood of activity, was accounted for by the logic tree structure of the probabilistic seismic hazard analysis, in which various models of faulting and fault activity were weighted according to the opinions of the experts.

The expert teams considered 87 fault sources or combinations of fault sources (CRWMS M&O, 1998c, Table 4-2). These sources included 30 faults or combinations of fault sources local to Yucca Mountain (within Yucca Mountain or in the adjacent basins), 51 regional faults or combinations of faults in the Yucca Mountain region {generally within a radius of approximately 100 km [62 mi] of the site}, and 6 faults or combinations of fault sources inferred from the tectonic models. Included in this list are faults identified through independent analysis of Type I faults by NRC and CNWRA staffs (McKague, et al., 1996, Section 4.1.1). For example, one of the expert teams considered 41 faults as individual fault sources (CRWMS M&O, 1998c, Tables AAR–1 and AAR–4). All are Type I faults. This same expert

team also demonstrated how nonindividual Type I fault sources contribute to seismicity as background or areal seismic sources.

In contrast to fault sources, areal sources represent areas of distributed or background seismicity in which no geologic or geophysical evidence can tie earthquakes to known faults. In this way, areal sources account for earthquakes that occur on unidentified or unidentifiable fault sources. Areal sources are typically developed to represent earthquakes with magnitudes that may not necessarily cause surface rupture. In the DOE probabilistic seismic hazard analysis (CRWMS M&O, 1998c; Stepp, et al., 2001), experts relied on empirical relationships that relate surface rupture to earthquake magnitude (e.g., Wells and Coppersmith, 1994). Given these data, there is greater than an 80-percent probability that **M**6.5 earthquakes will rupture the surface, while there is less than a 20-percent chance that **M**5.5 earthquakes will rupture the surface.

The boundaries of areal sources are drawn to define areas with relatively uniform seismicity and maximum magnitude, generally defined by the historic seismic record. All expert teams considered one to three areal source zones. For most teams, the source zones were used to capture background seismicity; thus, the maximum magnitude for areal sources close to Yucca Mountain was less than for those sources farther away; thus, the expert teams felt the fault source characterization at Yucca Mountain was superior to that in the surrounding regions. Some of the expert teams also included an explicit volcanic areal source term to explicitly account for seismic activity related to volcanism.

The recurrence rates for the faults were estimated using either recurrence intervals or slip rates. Recurrence and slip rates were primarily derived from paleoseismic data obtained by the U.S. Geological Survey in detailed investigations of faulting in the Yucca Mountain region (CRWMS M&O, 1998c). Additional constraints were derived from geologic data that estimate longer-term slip rates (e.g., Stamatakos, et al., 1997).

For fault sources, two methods were used by the experts to estimate recurrence. The first was to estimate the frequency of the largest earthquakes on the fault, and then specify the magnitude distribution function for the remaining earthquakes based on a particular recurrence model. The experts used three such recurrence models: (i) characteristic (Schwartz and Coppersmith, 1984), (ii) truncated exponential (Gutenberg and Richter, 1954), and (iii) modified truncated exponential. The second approach was to translate the slip rate into a seismic moment rate, and then partition the moments into earthquakes of various magnitudes according to a magnitude distribution model (Wesnousky, 1986).

For areal sources, the expert teams used the earthquakes in the catalog of historic earthquakes. The distribution of earthquake magnitudes in each areal source zone was interpreted following an exponential distribution (Gutenberg and Richter, 1954). Recurrence relationships for each zone were then estimated following a truncated exponential magnitude distribution to account for the maximum magnitude earthquake (Cornell and Van Marke, 1969).

The maximum magnitude earthquake is the largest earthquake that can be produced on a fault or in an areal source, regardless of its frequency of occurrence. For fault sources, the expert teams used empirical scaling relationships that relate maximum magnitude to the physical dimensions of the fault. Maximum magnitude was derived from fault length, rupture area, maximum surface displacement, and average surface displacement. In some cases, the expert

teams modified their maximum magnitude estimate by considering slip rate as well as rupture dimensions following Anderson, et al. (1996). In addition, the experts considered rupture area and average slip on the fault to estimate seismic moment, which was then converted to maximum magnitude using the relationships in Hanks and Kanamori (1979). For areal sources, the experts estimated the maximum magnitude earthquake based on the largest fault in the areal source not explicitly modeled as a fault source. Alternatively, the experts relied on the empirical relationships that relate surface rupture to earthquake magnitude based on empirical data (e.g., Wells and Coppersmith, 1994).

To quantify the historical seismicity of the region, the DOE facilitation team provided a single earthquake catalog to the expert teams. This catalog was compiled from 12 regional catalogs (CRWMS M&O, 1998c, p. G-2). The initial catalog contained 271,223 earthquakes of **M**0.5 and larger for the period 1868–1996. This initial catalog was modified in three ways. First, all the magnitudes were converted to moment magnitude ($\mathbf{M_W}$). Second, information on earthquakes from nuclear testing was removed based on compilations of all known nuclear tests. Third, foreshocks and aftershocks information was removed using two standard declustering methods (Youngs, et al., 1987; Veneziano and van Dyck, 1985). The Little Skull Mountain sequence was used to test the effectiveness of the two declustering techniques. Results show that the Veneziano and van Dyck (1985) method was better able to isolate foreshocks and aftershocks. After modifications, the resulting catalogs contained between 26,250 [Veneziano and van Dyck (1985) method] and 31,147 [Youngs, et al. (1987) method] earthquakes covering a circular area with a 300-km [186-mi] radius centered on Yucca Mountain.

The elicitation used a standard logic tree approach to delineate the alternative interpretations into a coherent framework and to incorporate uncertainty. The first branch of the tree identified alternatives of faults based on different interpretations of local and regional tectonics derived from the suite of viable tectonic models. Subsequent branches evaluated alternatives in fault-specific characteristics such as fault linkage, segmentation, maximum magnitude, activity rate, and seismogenic depth (CRWMS M&O, 1998c, Figures 4-2 and 4-3, example logic tree representations).

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

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

In the Yucca Mountain probabilistic seismic hazard analysis, the experts were to provide input (i.e., data, scientific interpretations, and estimates of parameter uncertainties) as part of the development of a probabilistic ground motion attenuation model. The ground motion attenuation models are mathematical relationships between ground motion and earthquake magnitude, distance, site conditions, and style of faulting and are used to estimate the levels of ground motion that may occur at a site. Consistent with the overall approach in the probabilistic

seismic hazard analysis, the probabilistic ground motion attenuation model includes estimates of aleatory and epistemic uncertainties in ground motion levels.

During review of the probabilistic seismic hazard analysis, specific issues were raised regarding the definition of the shallow crustal velocity near the free surface and the value of crustal kappa used for ground motion estimation at Yucca Mountain. These issues were raised because of the differences between the site condition at Yucca Mountain and the representations of the empirical strong motion database used (mainly California). There is a great difference in shear wave velocities, deep crustal damping [Q(f)], and shallow crustal {top 1–2 km [0.62–1.24 mi]} damping value (kappa) between California and Yucca Mountain. Kappa, defined as the spectral decay, is primarily caused by subsurface geological structures near the site. It is a smaller value for hard rock sites than for soft rock sites. The value of kappa estimated by Su, et al. (1996) for the southwestern part of the Nevada Test Site ranged from 0.005 to 0.024 second. In the probabilistic seismic hazard analysis, a value of 0.0186 second was used. DOE agreed (Schlueter, 2000) that if new studies find that the median value of kappa for material with shear wave velocity below 1,900 m/s [6,234 ft/s] is different from 0.0186 second, the median attenuation model will be adjusted.

Vibratory Ground Motion Hazard Results. Median and fractile ground acceleration and aleatory and epistemic uncertainties for various earthquake magnitudes, sources-to-site distances, and different fault styles were estimated by the experts. Uncertainties in seismic source characterization and ground motion attenuation relations were quantified by considering inputs from six seismic source fault displacement expert teams and seven ground motion experts. Each team and each expert provided their own assessment of uncertainty. The moment magnitude, $\mathbf{M_w}$, used in the probabilistic seismic hazard analysis ranged from 5.0 to 8.0 for normal and strike-slip faulting, and the distances examined were from 1 to 160 km [0.62 to 99 mi].

The probabilistic hazard for vibratory ground motion was calculated for peak ground acceleration, peak ground velocity, uniform hazard spectrum, and spectral accelerations at frequencies ranging 0.3–20 Hz. It was found that at 5–10 Hz, or high frequencies, the ground motions are dominated by earthquakes of magnitudes less than 6.5 and distances less than 15 km [9.3 mi]. At lower frequencies, 1–2 Hz, the ground motions are dominated by large events beyond distances of 50 km [31 mi]. The recurrence models contributed most to the uncertainty in the ground motion hazard, while geometric fault parameters were minor contributors to uncertainty. It was found that at 10 Hz, the dominant sources for seismic hazard ground motion are Paintbrush Canyon, Iron Ridge, and Solitario Canyon faults, and the host areal seismic source zone. For 1-Hz ground motion, the dominant seismic hazard comes from Death Valley–Furnace Creek faults.

The vibratory ground motion hazard calculations were performed for each expert-proposed attenuation equation and seismic source parameters. Combining the experts' hazard curves, giving each expert equal weight, a set of integrated hazard curves were produced. The integrated results, based on input from the six expert teams and the seven ground motion experts represent the seismic hazard and its associated uncertainty at Yucca Mountain. The separation between the 15th- and 85th-percentile curves conveys the effects of the epistemic uncertainty on the calculated hazards. In general, the greatest contributors to uncertainty in the hazard were within expert uncertainty, rather than expert-to-expert uncertainty. Additionally, the

total uncertainty caused by ground motion is larger than the uncertainty caused by the seismic source characterization.

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

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

4.1.1.2.5.8 Faulting Hazard Assessment

Appropriate design parameters for faulting, including setback distances, were derived using results from the DOE fault displacement hazard assessment. The probabilistic fault displacement hazard assessment was constructed through the expert elicitation used by DOE to develop a probabilistic seismic hazard analysis (CRWMS M&O, 1998c; Stepp, et al., 2001). The expert elicitation results were based on the findings of six expert teams, each consisting of three geoscientists. Fault displacement analyses evaluate the potential hazards of an active fault intersecting vital components of the engineered barrier system, especially waste packages.

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

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

to expect that the necessary information will be available to assess the probability of faulting affecting the preclosure repository system at the time of a potential license application. However, revised design plans for the surface facilities (McDaniel, 2004) show several potentially important systems, structures, or components are to be placed atop faults in Midway Valley, including the Midway Valley splay and possibly the Bow Ridge fault. DOE should develop information regarding the potential hazards these faults pose to the surface facilities.

4.1.1.2.5.9 Site Geoengineering Properties

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

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

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

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

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

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

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

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

4.1.1.2.6 Igneous Activity

The following sections on igneous activity refer to the requirements of 10 CFR 63.112(c). The potential DOE license application should contain a description of regional igneous activity adequate to permit evaluation of the preclosure safety analysis and the geologic repository operations area design.

Distributed basaltic volcanism is a long-lived characteristic of the Yucca Mountain region. Since the end of large-scale silicic caldera activity around 11 million years ago, approximately 12 basaltic volcanoes are known to have formed within 30 km [19 mi] of the potential repository site. Each of these volcanoes represent igneous events that consisted of one to four scoria cones and multiple subsurface intrusions that extend for kilometers away from the volcano. In addition to these known volcanoes, recent geophysical surveys (Blakely, et al., 2000) indicate that approximately 20 magnetic anomalies could be interpreted as more basaltic volcanoes buried in alluvial basins around the potential repository site (O'Leary, et al., 2002; Hill and Stamatakos, 2002).

Basaltic scoria cone volcanoes form during eruptions that typically have 2–8-km [1–5-mi]-high eruption columns. These eruption columns can disperse fragments of quenched magma (i.e., tephra) tens of kilometers from the vent, depending on the strength and direction of prevailing winds. At distances of approximately 20 km [12 mi] from the volcano, basaltic tephra

fall deposits are generally 1–100 cm [0.4–39 in] thick with bulk densities of 1,000–1,700 kg/m^3 [62–106 lb/ft^3] (e.g., Hill, et al., 1998; NRC, 1999b).

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

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

Available analysis or data do not support the basis for the DOE conclusion that a 3-cm [1-in]-thick volcanic tephra deposit represents a worst-case event. The 3-cm [1-in]-thick deposit cited in DOE (1999b) applies only for a volcanic eruption occurring 150 km [93 mi] from the potential repository site (i.e., Perry and Crowe, 1987). Basaltic volcanic eruptions have an annual probability of occurrence that exceeds 1×10^{-6} within 10 km [6 mi] of the potential repository site (e.g., NRC, 1999b). Tephra fall deposits measured about 10 km [6 mi] from volcanoes analogous to those within 20 km [12 mi] of Yucca Mountain are on the order of 1–100 cm [1–39 in] thick (e.g., NRC, 1999b). In addition, Perry and Crowe (1987) conclude that a 1-m [3-ft]-thick tephra-fall could occur approximately 3 km [2 mi] from a basaltic volcanic event. Noncompacted, dry basaltic volcanic tephra has bulk deposit densities that can range 1,000–1,700 kg/m^3 [62–106 lb/ft^3] (e.g., Hill, et al., 1998; NRC, 1999b). These deposit densities can increase by an approximate factor of two when wet, depending on average grain size and sorting of the deposit. Thus, a basaltic volcanic eruption located 1–10 km [0.6–6 mi] from Yucca Mountain represents a Category 2 event that could deposit on the order of 1–100 cm [0.4–39 in] of dry tephra on surface structures, resulting in dry loads between 10 and 1,700 kg/m^2 [2 and 348 lb/ft^2]. In addition, DOE has not provided a technical basis to determine the analogy of wind-blown sands to volcanic tephra particles. Volcanic tephra fall deposits may have larger amounts of fine-grained particles (i.e., ash) than wind-blown sands. Thus, the possible effects of volcanic ash on air filters and ventilation systems may be different than the effects on these systems from wind-blown sand.

The DOE summary of igneous activity relevant to preclosure safety (DOE, 1999b) does not provide sufficient information to evaluate potential effects on the performance of surface facilities. DOE should provide additional information on the amount and character of potential tephra deposits that could fall on surface facilities from basaltic volcanic eruptions located within areas where the annual probability of a new volcano forming is greater than 1 chance in 10,000

to occur before permanent closure. DOE should then evaluate the potential effects of these tephra fall deposits on structures and systems important to safety.

4.1.1.2.7 Site Geomorphology

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

4.1.1.2.8 Site Geochemistry

4.1.1.2.8.1 Geochemistry of Subsurface Waters

The unsaturated zone at Yucca Mountain contains pore waters, fracture waters, and isolated perched water (CRWMS M&O, 2000a; Bechtel SAIC Company, LLC, 2003b). Yang, et al. (1998, 1996) measured chemical compositions of ambient pore water and perched water from Yucca Mountain and its vicinity. More recent data on perched and pore waters are reported in Technical Basis Document No. 5 (Bechtel SAIC Company, LLC, 2003c, Section 3.2.1); DOE considers the latter to be more representative of the waters that may enter the drift. Pore waters have been extracted from unsaturated zone borehole core samples using high-pressure uniaxial compression and ultracentrifuge techniques. Perched water and pore water compositions were measured using inductively coupled plasma spectroscopy and ion chromatography. Stratigraphic units penetrated by the boreholes are (in descending order) the Paintbrush Group (composed of Tiva Canyon Tuff, Yucca Mountain Tuff, Pah Canyon Tuff, and Topopah Spring Tuff), the Calico Hills Formation, and the Prow Pass Tuff. The newer ambient pore water compositions include samples from the potential repository horizon in the Topopah Spring Tuff. There are no measured fracture water compositions from Yucca Mountain because of the difficulty of collecting fracture water samples. However, fracture water has been collected from Rainier Mesa (White, et al., 1980) and appears to be similar in composition to perched and saturated zone waters collected at Yucca Mountain.

The pore water analyses of Yang, et al. (1998, 1996) provide valuable characterizations of unsaturated zone ground water chemistry at Yucca Mountain, but there are indications that aspects of these data are unreliable. Yang, et al. (1998, 1996) noted charge imbalances in the chemical analyses. In addition, Apps (1997) concluded that measured pH values are inaccurate, based on inconsistencies of pH measurements of water from the J–13 Well. Browning, et al. (2000) noted that the range of analytical pH for pore waters extracted from similar depths within individual boreholes appears unreasonably wide, suggesting that measured pH values are unreliable. Browning, et al. (2000) noted similar abrupt variations in some reported major aqueous species concentrations. Potassium occurs in primary and secondary phases at Yucca Mountain and is an important component of Yucca Mountain waters, but Yang, et al. (1998, 1996) did not always report potassium concentrations. Finally,

particulate aluminum in filtered samples resulted in unreliable aluminum concentrations (Yang, et al., 1996). Clearly, there are significant uncertainties in the pore water analyses of Yang, et al. (1998, 1996) that compromise the utility of these data. Apps (1997) and Browning, et al. (2000) propose different sets of assumptions for revising/improving these data using aqueous speciation calculations.

DOE used little or none of the ground water compositional data provided by Yang, et al. (1998, 1996); Apps (1997); or Browning, et al. (2000) in any process-level models providing input into the Total System Performance Assessment–Site Recommendation. For the Total System Performance Assessment–License Application, DOE has indicated it will base characterization of unsaturated zone water chemistry on the more recent pore water data obtained by ultracentrifugation of samples from potential repository horizons in the Cross-Drift and in boreholes SD–9 and NRG–7/7A (Bechtel SAIC Company, LLC, 2003c). The analyses show a great deal of diversity in terms of major ion chemical characteristics. This formed the basis for representative waters used in calculating the range of possible seepage water compositions to support a potential license application. The unsaturated zone waters tend to be calcium-sulfate-chloride type, rather than sodium-bicarbonate type, but a range of compositions with varying $(Ca^{2+} + Mg^{2+})/(Na^{+} + K^{+})$ ratios is observed. As discussed in Section 5.1.3.3 of this report (Quantity and Chemistry of Water Contacting Engineered Barriers and Waste Forms), it appears DOE has obtained sufficient data on unsaturated zone water compositions. However, the report detailing the data and their interpretation was not available at the time of this assessment; therefore, the staff has not yet evaluated whether the new analyses have addressed the uncertainties attending the Yang, et al. (1998, 1996) data.

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

4.1.1.2.8.2 Geochemistry of Rock Strata

CRWMS M&O (2000a) and Bechtel SAIC Company, LLC (2003b) summarize data provided by DOE on geochemical composition of the rock strata at Yucca Mountain through the year 2000. X-ray diffraction techniques were used to characterize the mineralogy of core samples from boreholes in the vicinity of Yucca Mountain, as well as fracture samples collected within the Exploratory Studies Facility. These data were combined with information from stratigraphic and potentiometric surfaces and incorporated into the three-dimensional mineralogic model part of the Geologic Framework Model. The mineralogic model was designed as a resource to interpolate information about mineral assemblages between boreholes where measurements were made, and this model has been a useful effort. Although DOE provided sufficient information on matrix mineralogy via developing the mineralogic model, staff considered at the time that DOE should provide additional support to characterize the mineralogy of fractures and lithophysal cavities for numerical modeling efforts, such as reactive transport modeling.

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

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

4.1.1.2.8.3 Geochemical Alterations

The chemical compositions of ambient ground water from Yucca Mountain are expected to evolve significantly before contacting the engineered barrier system. Several different factors will control the composition of water as it infiltrates through the overlying rock toward the drift, including temperature, the types of materials that interact chemically with the water along the flow pathway, and flow velocity versus reaction rate. Thermal-hydrological models suggest that temperatures at the drift crown will remain above nominal boiling for approximately 1,000 years (CRWMS M&O, 2000d). These models suggest that ambient ground water compositions should adequately characterize seepage compositions for a period of almost 10,000 years, but more information concerning the technical basis should be provided. It is unlikely that ambient pore water will ever drip in significant volumes from the drift crown at the potential Yucca Mountain repository because fractures are expected to be the predominant flow pathway to the drift. Even if ambient pore water drips in significant volumes, the effects likely would be unimportant to the lifetime of the drip shield/waste package because corrosion is enhanced in higher-temperature, more saline solutions. After water seeps out of the porous rock, its chemical composition continues to evolve through evaporation and salt formation processes in the engineered barrier system. Thus, ambient ground water above the potential repository will be subjected to thermal perturbations in several different environments that will change its chemical compositions during time. Predictions of the quantity and chemistry of water contacting the drip shields and waste packages for a period of thousands of years for the potential Yucca Mountain repository are thus difficult and must be accomplished by considering both analytical data and numerical models.

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

4.1.1.3 Summary and Status

Table 4.1.1-1 provides a summary of the status of the preclosure items related to the site description with cross-references to related agreements in the postclosure Key Technical Issues. The table forms the basis for pending discussion with DOE regarding preclosure site description. Those items considered pending involve either additional review by staff or additional information from DOE.

Table 4.1.1-1. Summary of Resolution Status of Site Description Preclosure Topic			
Preclosure Item	**Status**	**Related Agreement**	**Note**
Site Geography	Pending	None	Current information sufficient, but site location information may need updates given recent changes to surface facility design.*
Regional Demography	Pending	None	Demographic information should be updated to include most recent census data.
Local Meteorology and Regional Climatology	Pending	None	Current information lacks sufficient information on site insolation. Cask decay heat removal capabilities for the aging-facility pads should be determined. Updated information to include regional data from Amargosa Farms and Nevada Test Site should be provided.
Regional and Local Surface and Ground Water Hydrology	Pending	None	Additional information should be provided to evaluate potential water and debris flows: maximum versus 100-year flood, siting criteria for ventilation shafts, aging-facility pads, and muck piles in Midway Valley, transportation near active drainages, and water influx along faults.
Site Geology and Seismology	Pending	RDTME.2.01 RDTME.2.02 RDTME.3.03 RDTME.3.04 SDS.1.02 SDS.2.01 SDS.2.02 SDS.2.03	Current information on regional geologic and tectonic setting as well as site stratigraphy is sufficient. Correlation of rock and model units should be provided. Because of changes to surface facility design, additional site soil data should be provided for seismic response models and site design. Additional information on dynamic properties should also be provided to complete evaluation of DOE site response analysis.

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

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

4.1.2.1 Areas of Review

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

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

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

4.1.2.2 Staff Review of Available Information

4.1.2.2.1 General Information

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

Approximately 70,000 metric tons heavy metal [77,162 tons] of high-level waste will be received, processed, and emplaced during the proposed operational period of 24 years (CRWMS M&O, 1999a). This high-level waste includes commercial spent nuclear fuel, DOE spent nuclear fuel (including naval reactor spent nuclear fuel), vitrified high-level radioactive waste, and immobilized plutonium. The geologic repository operations area is categorized into surface and subsurface facilities. The surface facilities will be used to receive spent nuclear fuel and defense high-level waste shipments, provide capability to age waste as necessary, and prepare and package the wastes for underground emplacement (McDaniel, 2004; Board, 2004). The surface facilities will house radiological protection, utilities, and ventilation for the underground facilities and also provide other supporting functions. The surface facilities consist of four primary functional areas: (i) the waste receiving and inspection area, where incoming trucks and rail cars are inspected, received, and temporarily staged; (ii) aging areas, where the received wastes are placed for cooling and radiological decay until ready for disposal; (iii) the surface portion of the waste handling operations area, which includes all buildings where

radioactive material is handled for packaging; and (iv) the general support facilities, consisting of administrative buildings, security stations, and warehouses (McDaniel, 2004).

4.1.2.2.2 Surface Facilities

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

The waste will arrive at the repository by rail or truck and be received at the radiologically controlled area 24 hours a day. The rail shipment will arrive at the site as a unit train consisting of one or two locomotives, three to five rail cars carrying one cask per rail car, and buffer rail cars between the rail cars with casks. The truck shipment will arrive in legal-weight trucks. DOE developed a schedule of receipt based on a reference design (CRWMS M&O, 1999a). The reference design is based on an approximated annual receipt rate of 3,000 metric tons [3,307 tons] of uranium waste for an operational period of 24 years. The annual rate of receiving and handling casks, canisters, fuel assemblies, and disposal canisters in the facility will vary. In the preclosure safety analysis, however, it is important to know the maximum handling rate because under 10 CFR 63.21(c)(5), the preclosure safety analysis must assume that operations will be carried out at the maximum capacity and rate of receipt of waste stated in the license application.

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

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

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

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

4.1.2.2.3 Subsurface Facilities

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

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

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

The emplacement drifts will be an array of horizontal tunnels trending approximately east-northeast-west-southwest (252 azimuth) between the east and west mains. Each drift will have a diameter of 5.5 m [18.5 ft] and will be separated from the adjacent drifts by a center-to-center distance of 81 m [265.7 ft]. The transition from the east and west mains to the emplacement drifts (which are nearly perpendicular to the mains) will be provided through the emplacement-drift turnouts (CRWMS M&O, 2000b).

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

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

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

Construction of underground openings and waste emplacement operations will proceed concurrently, and development of underground openings will not interfere with the waste emplacement operations (CRWMS M&O, 1999b; DOE, 1998). The repository openings are constructed to serve a variety of functions. The main access (shafts and ramps) provides facilities for ventilating the subsurface, emplacing waste, removing excavated material, performing maintenance, and transporting staff and materials. A conveyor belt will transport excavated rock from the subsurface to the surface. A tunnel boring machine will be used for most underground excavations. Mechanical methods, such as road-header machines or the drill-and-blast excavation method, may be used where tunnel boring machine operation is not feasible. Other construction-related activities will include installation of ground supports and transportation of excavated rock from the subsurface to the surface. A general description of the construction of the repository surface and subsurface facilities has been provided in various reports (CRWMS M&O, 1999a; DOE, 2001, 1998).

The subsurface transporter is used to transport the waste package to the emplacement drifts. The subsurface transporter is a shielded cask mounted on a rail car. A locomotive will be coupled to each end of the subsurface transporter at the waste handling loading facility. The two locomotives will move the transporter down the north ramp (sloping at a 2.15-percent grade) and along the access main tunnel to reach the emplacement drift turnout. At the selected emplacement drift, one locomotive will be uncoupled. The remaining locomotive will push the transporter against the transfer dock at the emplacement drift entrance. The transporter is equipped with a self-contained mechanism that will push the rail car through the emplacement drift door and position it for unloading. A self-propelled, remotely operated emplacement gantry, which is stationed in the emplacement drift during active emplacement operations, will move into position over the rail car. The gantry lifting arms will engage the pallet structure to lift the pallet and waste package off the transporter. After raising the pallet to a desired elevation, the gantry would move the waste package to its emplacement location in the drift and lower the waste package and the pallet onto the drift invert. The gantry would disengage from the pallet and return to a position near the emplacement drift door. If the waste package is to be moved during or after emplacement, it will be removed from the emplacement drift by following the emplacement operations in reverse order.

4.1.2.3 Summary and Status

The staff is reviewing information provided by DOE describing the structures, systems, components, equipment, and operational process activities. This review is coordinated with the

review of information to be provided in the preclosure safety analysis and will focus on the following areas:

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

As discussed earlier, to conduct a meaningful preclosure safety analysis on the design and operations to determine the structures, systems, and components important to safety and the safety measures, the structures, systems, components, equipment, process activities, and sources of hazardous materials involved in the safety analysis need to be sufficiently described. The extent of the description should be detailed enough for the staff to obtain a clear understanding of the design and operations and consistent with the level of the preclosure safety analysis performed. Consequently, the sufficiency of this subsection should be evaluated in conjunction with other subsections relevant to the preclosure safety analysis, including repository design. Review and evaluation of descriptions of the structures, systems, components, equipment, and operational process activities will continue as the DOE design and preclosure safety analysis are made available.

4.1.2.4 References

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

4.1.3 Identification of Hazards and Initiating Events

4.1.3.1 Areas of Review

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

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

- 10 CFR 63.112(b) requires the preclosure safety analysis include an identification and systematic analysis of naturally occurring and human-induced hazards at the geologic repository operations area.

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

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

4.1.3.2 Staff Review of Available Information

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

Major design changes to the surface facility are including a dry transfer system for handling spent nuclear fuel assemblies and surface aging facilities in place of the pool-based transfer system and adding a staging system to the site recommendation design (McDaniel, 2004). For fuel handling operations, the new design includes two dry transfer buildings, one canister transfer building, and a transporter cask buffer staging area proposed to be constructed in a phased manner. The proposed design would involve substantial changes in facility process activities and requirements of structures, systems, and components for the surface facilities. Additionally, the drift layout, construction phases, and ventilation design for the subsurface facilities have

been modified; however, the subsurface operations and systems required for emplacement activities have not changed (Board, 2004). These proposed modifications in design and operations, especially for surface facilities, may significantly affect the preclosure safety analyses submitted earlier (CRWMS M&O, 1999a; DOE, 2001a).

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

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

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

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

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

- Event frequency is greater than or equal to 10^{-6} per year. Any event with a probability of at least 1 in 10,000 of occurring during the 100-year preclosure period is included for further consideration.

- Event is not bounded by analysis of another event.

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

Table 4.1.3-1. Preliminary Operational Hazard Analysis

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

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

Table 4.1.3-1. Preliminary Operational Hazard Analysis (continued)				
No.	**Functional Area**	**Generic Hazard*†**	**Potential Event*†**	**Staff Remark**
3	Waste Handling–Assembly Transfer	Collision/Crushing	Transportation Cask: drop, slap down, collision, handling equipment drop on cask Spent Nuclear Fuel Assembly: drop on floor, slap down, collision, drop onto spent nuclear fuel assembly staging rack, drop onto waste package Waste Package before and after closure weld: drop, slap down, drop onto sharp object, collision, handling equipment drop	Staff reviewed the information provided by DOE and has no further questions at this time.
		Chemical Contamination/Internal Flooding	Leakage, uncontrolled draw-down, or filling of remediation pool	
		Explosion/Implosion	None	
		Fire/Thermal	Spent nuclear fuel overheating resulting in excessive clad temperature and zircalloy cladding fire due to loss of water in remediation pool Fuel damage by burn-through during welding process, spent nuclear fuel overheating in waste package resulting in excessive cladding temperature and possible zircalloy cladding fire	

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

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

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

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

No.	Hazard	Hazard Definition	DOE Assessment	Staff Remark
1	Avalanche	A large mass of snow, ice, soil, or rock or mixtures of these materials falling, sliding, or flowing under gravity	Not applicable to the hazards list • High mountain ranges do not exist at Yucca Mountain	Staff has reviewed information provided by DOE and has no further questions at this time.
2	Coastal Erosion	Wearing away of soil and rock by waves and tidal action	Not applicable to the hazards list • Coastline does not exist at Yucca Mountain	Staff has reviewed information provided by DOE and has no further questions at this time.
3	Dam Failure	Failure of a large human-made barrier that creates and restrains a large body of water	Not applicable to the hazards list • No dam of sufficient size exists in proximity to Yucca Mountain	Staff has reviewed information provided by DOE and has no further questions at this time.
4	Debris Avalanche	Sudden and rapid movement of debris down steep slopes resulting from intensive rainfall	Applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficient to affect 100-year preclosure period • Consequence of process is significant • Annual event frequency $\geq 10^{-6}$ • Not included in another analysis	Staff has reviewed information provided by DOE; however, staff is waiting for further information from DOE regarding this hazard. Excavated rock debris placed next to surface facilities should be examined for its effects on structures during a seismic event.
5	Denudation	Sum of processes that result in wearing away or progressive lowering of Earth's surface by weathering, mass wasting, and transportation	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is slow enough to affect during 100-year preclosure period	Staff has reviewed information provided by DOE and has no further questions at this time.

Table 4.1.3-2. List of Natural Hazards with DOE Assessment*† (continued)				
No.	**Hazard**	**Hazard Definition**	**DOE Assessment**	**Staff Remark**
6	Dissolution	Processes of chemical weathering by which mineral and rock materials pass into solution	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficiently high to affect 100-year preclosure period and may create rockfall • Consequence is indeterminant; assumed to be equivalent to significant enough to affect 100-year preclosure period • Annual event frequency is indeterminant; assumed $\geq 10^{-6}$ • Key Block Analysis Report will address rockfall issue	Staff has reviewed information provided by DOE and has no further questions at this time
7	Eperogenic Displacement	Geomorphic processes of uplift and subsidence that produce broader features of continents and oceans	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is not sufficient to pose credible hazard during 100-year preclosure period	Staff has reviewed information provided by DOE and has no further questions at this time.
8	Erosion	Slow wearing of soil and rock by weathering, mass wasting, and action of streams	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is not sufficient to pose credible hazard during 100-year preclosure period	Staff has reviewed information provided by DOE and has no further questions at this time.
9	Extreme Weather Fluctuations	Various types of weather fluctuations that pose unusual design challenges	Not applicable to the hazards list • No potential exists at Yucca Mountain	Staff has reviewed information provided by DOE and has no further questions at this time

Table 4.1.3-2. List of Natural Hazards with DOE Assessment*† (continued)				
No.	**Hazard**	**Hazard Definition**	**DOE Assessment**	**Staff Remark**
10	Extreme Wind	Fastest mile of wind with 100-year return period	Applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficient during 100-year preclosure period • Potential consequence is indeterminant; assumed equivalent to true • Annual event frequency $\geq 10^{-6}$ • Not included in another analysis	In the report, Extreme Wind/Tornado/ Tornado Missiles‡, DOE used a 50-year return period for design-basis wind speed, not 100 years. In addition, Yucca Mountain and surrounding region, according to SEI/ASCE 7–02§, are a special area requiring site-specific measurement of wind speed. Staff review is provided in this report.
11	Flood (Storm and River Diversion)	Area covered with water from storm or river diversion caused by inadequate drainage	Applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficiently high during 100-year preclosure period • Consequence of process are sufficiently high • Annual event frequency $\geq 10^{-6}$ • Not included in another analysis	Staff has reviewed information provided by DOE and has no further questions at this time.
12	Fungus, Bacteria, and Algae	General class of microorganisms that may be present in subsurface environment	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficiently high during 100-year preclosure period • Consequence of process not significant to affect 100-year preclosure period	Staff has reviewed information provided by DOE and has no further questions at this time.
13	Glacial Erosion	Lowering of Earth's surface due to grinding and scouring by glacier ice incorporated with rock fragments	Not applicable to the hazards list • No potential exists at Yucca Mountain for a glacier	Staff has reviewed information provided by DOE and has no further questions at this time.

Table 4.1.3-2. List of Natural Hazards with DOE Assessment*† (continued)				
No.	**Hazard**	**Hazard Definition**	**DOE Assessment**	**Staff Remark**
14	Glaciation	Formation, movement, and recession of glaciers or ice sheets	Not applicable to the hazards list • No potential exists at Yucca Mountain for a glacier and associated climate change	Staff has reviewed information provided by DOE and has no further questions at this time.
15	High Lake Level	Potential overflow or flooding of lake	Not applicable to the hazards list • No potential exists at Yucca Mountain because there is no lake nearby	Staff has reviewed information provided by DOE and has no further questions at this time.
16	High Tide	High tide in water connected with ocean having potential for flooding inland areas	Not applicable to the hazards list • No potential exists at Yucca Mountain because there is no ocean or coastal area	Staff has reviewed information provided by DOE and has no further questions at this time.
17	High River Stage	Potential flooding of river or natural permanent or seasonal surface stream with considerable volume	Not applicable to the hazards list • No potential exists at Yucca Mountain because there is no river nearby	Staff has reviewed information provided by DOE and has no further questions at this time.
18	Hurricane	Intense cyclone that forms over tropical oceans	Not applicable to the hazards list • No potential exists at Yucca Mountain because it is located approximately 360 km [225 mi] inland from nearest ocean, northeast of Santa Monica Bay near Los Angeles; based on ANSI/ANS 2.8–92‖, site needs to be within 160–320 km [100–200 mi] from ocean for hurricane to be potential natural hazard	Staff has reviewed information provided by DOE and has no further questions at this time.

Table 4.1.3-2. List of Natural Hazards with DOE Assessment*† (continued)				
No.	**Hazard**	**Hazard Definition**	**DOE Assessment**	**Staff Remark**
19	Landslide	Wide variety of mass movement of land forms and processes involving downslope transport with gravitational influence	Applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficiently high to affect 100-year preclosure period • Consequence is indeterminant; assumed equivalent to true • Annual event frequency $\geq 10^{-6}$ • Not part of another analysis	Staff has reviewed information provided by DOE and has no further questions at this time.
20	Lightning	Flashing of light produced by discharge of atmospheric electricity between charged cloud and Earth	Applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficiently high during 100-year preclosure period • Consequence is indeterminant; assumed equivalent to true • Annual event frequency $\geq 10^{-6}$ • Not part of another analysis	Staff has reviewed information provided by DOE and has no further questions at this time.
21	Low Lake Level	Low level of lake water used for cooling	Not applicable to the hazards list • No potential exists at Yucca Mountain because there is no lake nearby	Staff has reviewed information provided by DOE and has no further questions at this time.
22	Low River Level	Low level of river water used for cooling	Not applicable to the hazards list • No potential exists at Yucca Mountain because there is no river nearby	Staff has reviewed information provided by DOE and has no further questions at this time.
23	Meteorite Impact	Impact of meteoroid reaching Earth's surface without completely vaporizing	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficiently high during 100-year preclosure period • Consequence is indeterminant; assumed equivalent to true • Annual event frequency $\leq 10^{-6}$	Staff has reviewed information provided by DOE and has no further questions at this time.

Table 4.1.3-2. List of Natural Hazards with DOE Assessment*† (continued)				
No.	**Hazard**	**Hazard Definition**	**DOE Assessment**	**Staff Remark**
24	Orogenic Diastrophism	Movement of Earth's crust produced by tectonic processes where structures within fold-belt mountain areas formed, including thrusting, folding, and faulting	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is too low to affect 100-year preclosure period	Staff has reviewed information provided by DOE and has no further questions at this time.
25	Rainstorm	Storm that produces 100-year or greater maximum rainfall rate occurring for 1 day	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficiently high during 100-year preclosure period • Consequence is indeterminant; assumed significant • Annual event frequency $\geq 10^{-6}$ • Bounded by debris avalanche, flooding, and landslide events for which this is an initiator	Staff has reviewed information provided by DOE and has no further questions at this time.
26	Range Fire	Combustion of natural vegetation external to repository that propagates to combustible materials within operations area	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficiently high during 100-year operational period • Consequence is significant • Annual event frequency $\geq 10^{-6}$ • Will be addressed in fire hazard analyses	This hazard will be addressed in the fire hazard analyses of the potential facilities. DOE has not provided the fire hazard assessment for the current facility design.

Table 4.1.3-2. List of Natural Hazards with DOE Assessment*† (continued)				
No.	**Hazard**	**Hazard Definition**	**DOE Assessment**	**Staff Remark**
27	Sandstorm	Extreme wind capable of transporting sand and other unconsolidated surficial materials	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficient during 100-year preclosure period • Consequence is indeterminant; assumed significant • Annual event frequency $\geq 10^{-6}$ • Bounded by extreme wind and tornadoes events • Potential filter clogging is screened out from further consideration because of capability for orderly facility shutdown through technical specification—a to-be-verified item	Staff has reviewed information provided by DOE and has no further questions at this time.
28	Sedimentation	Process of forming or accumulating sediment in layers	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is too low in 100-year preclosure period	Staff has reviewed information provided by DOE and has no further questions at this time.
29	Seiche	Free or standing wave oscillation of water surface in enclosed or semienclosed basin	Not applicable to the hazards list • No potential exists at Yucca Mountain because there is no large body of water nearby	Staff has reviewed information provided by DOE and has no further questions at this time.
30	Seismic Activity (Uplifting)	Structurally high area in the crust produced by positive movements for long time periods resulting in faults giving rise to upthrust of rocks	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is too slow in 100-year preclosure period	Staff has reviewed information provided by DOE and has no further questions at this time.

Table 4.1.3-2. List of Natural Hazards with DOE Assessment*† (continued)				
No.	**Hazard**	**Hazard Definition**	**DOE Assessment**	**Staff Remark**
31	Seismic Activity (Earthquake)	Earthquakes, including those artificially induced	Applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficiently high during 100-year preclosure period • Consequence is significant • Mean annual probabilities of Frequency Categories 1 and 2 design-basis ground motions are 1×10^{-3} and 1×10^{-4}; structures, systems, and components important to safety will be designed to withstand design-basis earthquake (Frequency Categories 1 and 2), as appropriate • Not bounded by another analysis	DOE has not yet submitted agreed-to information on either seismic inputs or seismic design.
32	Seismic Activity (Surface Fault Displacement)	Fracture or zone of fractures along which there is potential for displacement of sides relative to each other parallel to fracture	Applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficiently high during 100-year preclosure period • Mean annual probabilities of Frequency Categories 1 and 2 design-basis fault displacements are 1×10^{-4} and 1×10^{-5}; structures, systems, and components important to safety will be designed to avoid or withstand design-basis fault displacements (Frequency Categories 1 and 2), as appropriate • Not bounded by another analysis	DOE has not yet provided information on faulting hazard or design. DOE should consider new faults at the surface facility site, given the new layout.

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

No.	Hazard	Hazard Definition	DOE Assessment	Staff Remark
33	Seismic Activity (Subsurface Fault Displacement)	Fracture or zone of fractures along which there is potential for displacement of sides relative to each other parallel to fracture	Applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficiently high during 100-year preclosure period • Mean annual probabilities of Frequency Categories 1 and 2 design-basis fault displacements are 1×10^{-4} and 1×10^{-5}; structures, systems, and components important to safety will be designed to withstand design-basis fault displacements (Frequency Categories 1 and 2), as appropriate • Not bounded by another analysis	DOE has not yet provided information on faulting hazard or design.
34	Static Fracturing	Break in rock due to mechanical failure by stress	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficiently high to affect 100-year preclosure period • Consequence is indeterminant; assumed significant • Annual event frequency $\geq 10^{-6}$ • Will be addressed in Key Block Analysis Report	Potential degradation of the emplacement drifts during preclosure period should be assessed. DOE has committed to address this concern in a separate report.¶
35	Stream Erosion	Progressive removal of bedrock, overburden, soil, or other exposed matters from stream channel surface	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is too slow to affect 100-year preclosure period	Staff has reviewed information provided by DOE and has no further questions at this time.

Table 4.1.3-2. List of Natural Hazards with DOE Assessment*† (continued)				
No.	**Hazard**	**Hazard Definition**	**DOE Assessment**	**Staff Remark**
36	Subsidence	Sudden sinking or gradual downward settling of Earth's surface with little or no horizontal motion	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficiently high to affect 100-year preclosure period • Consequence is indeterminant; assumed significant • Annual event frequency $\geq 10^{-6}$ • Screened out because subsurface fault displacement will be the only natural phenomenon that would result in collapse of underground excavations leading to subsidence; emplacement levels would be at least 200 m [656 ft] below the directly overlying ground surface; emplacement drifts will be supported by rock bolts, steel mesh, and steel sets; no surface-handling facilities will be directly over emplacement drifts	Staff has reviewed information provided by DOE and has no further questions at this time.
37	Tornado	Small cyclone generally less than 500 m [1,650 ft] in diameter with extremely strong winds	Applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficient to affect 100-year preclosure period • Consequence is indeterminant; assumed significant • Annual event frequency $\geq 10^{-6}$ • Not bounded by another analysis	Staff has reviewed information provided by DOE in the report Extreme Wind/Tornado/Tornado Missiles‡ and has no further questions at this time.
38	Tsunami	Gravitational sea wave produced by large-scale, short-duration disturbance on ocean floor	Not applicable to the hazards list • No potential exists at Yucca Mountain because there is no coastal region	Staff has reviewed information provided by DOE and has no further questions at this time.

Table 4.1.3-2. List of Natural Hazards with DOE Assessment*† (continued)				
No.	**Hazard**	**Hazard Definition**	**DOE Assessment**	**Staff Remark**
39	Undetected Geologic Features	Geologic features of concern to the 100-year preclosure period include natural events such as faults and volcanoes	Not applicable to the hazards list • No potential exists at Yucca Mountain because site characterization provided sufficient assurance these types of activities would have been detected	Staff has reviewed information provided by DOE and has no further questions at this time.
40	Undetected Geologic Processes	Geologic processes of concern to the 100-year preclosure period include events such as erosion, tectonic, and seismic processes	Not applicable to the hazards list • No potential exists at Yucca Mountain because site characterization provided sufficient assurance these types of activities would have been detected	Staff has reviewed information provided by DOE and has no further questions at this time.
41	Volcanic Eruption	Magma and associated gases rise into the crust and are extruded onto Earth's surface and into atmosphere	Not applicable to the hazards list • No potential exists at Yucca Mountain because there is no potential for volcanic center at the site	Staff has reviewed information provided by DOE and has no further questions at this time. DOE should provide technical basis to exclude hazards from lava flows in the potential repository surface operations area from the Category 2 event sequence list.
42	Volcanism (Intrusive Magmatic Activity)	Development and subsurface movement of magma and mobile rock materials	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficiently high to affect 100-year preclosure period (indeterminant; assumed true) • Consequence is indeterminant; assumed significant • Annual event frequency $\leq 10^{-6}$	Staff has reviewed information provided by DOE and has no further questions at this time.

Table 4.1.3-2. List of Natural Hazards with DOE Assessment*† (continued)				
No.	**Hazard**	**Hazard Definition**	**DOE Assessment**	**Staff Remark**
43	Volcanism (Ash Flow and Extrusive Magmatic Activity)	Highly heated mixture of volcanic gases, magma, mobile rock material, and ash traveling down flank of volcano or along ground surface	Not applicable to the hazards list • No potential exists at Yucca Mountain for silicic volcanism	Staff has reviewed information provided by DOE and has no further questions at this time.
44	Volcanism (Ash Fall)	Airborne volcanic ash falling from eruption cloud	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is indeterminant; assumed significant • Consequence not significant to affect 100-year preclosure period because — Worst-case ash fall depth is 3 cm [1.2 in] — Worst-case live load on flat roof is 868.5 Pa [18.14 lb/ft^2], which is less than minimum 1997 Uniform Building Code requirements# • Filter clogging due to ash fall is bounded by filter clogging by sandstorm event	Staff disagrees with the DOE assessment because potential effects of basaltic tephra fall have not been bounded by existing DOE analyses, and realistic effects may exceed current design bases for roof loads and air circulation system performance. Staff review is provided in this section of the Integrated Issue Resolution Status Report.

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

No.	Hazard	Hazard Definition	DOE Assessment	Staff Remark
45	Waves	Oscillatory movement of water manifested by alternate rise and fall of water surface	Not applicable to the hazards list • No potential exists at Yucca Mountain because there is no large body of water nearby	Staff has reviewed information provided by DOE and has no further questions at this time.

*CRWMS M&O. "MGR External Events Analysis." ANL–MGR–SE–000004. Rev. 00. Las Vegas, Nevada: CRWMS M&O. 1999.
†DOE. "Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation." TDR–MGR–SE–000009. Rev. 00 ICN 03. Las Vegas, Nevada: DOE. 2001.
‡CRWMS M&O. "Extreme Wind/Tornado/Tornado Missile Hazard Analysis." CAL–WHS–MD–000002. Rev. 00B. Las Vegas, Nevada: CRWMS M&O. 2003.
§Structural Engineering Institute/American Society of Civil Engineers. "Minimum Design Loads for Buildings and Other Structures." SEI/ASCE 7-02. Rev. of ASCE 7-98. Reston, Virginia: American Society of Civil Engineers. 2003.
‖American National Standards Institute/American Nuclear Society. "Determining Design Basis Flooding at Power Reactor Sites, An American National Standard." ANSI/ANS 2.8-92. La Grange, Illinois: American Nuclear Society. 1992.
¶Bechtel SAIC Company, LLC. "Bounding Characteristics of Credible Rockfalls of Preclosure Period." 800–00C–MGR0–00200–000–00A. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2004. (Current staff understanding)
#International Conference of Building Officials. "Uniform Building Code." Whittier, California: International Conference of Building Officials. 1997.

Table 4.1.3-3. List of Human-Induced Events with DOE Assessment*†				
No.	**Hazard**	**Hazard Definition**	**DOE Assessment**	**Staff Remark**
1	Aircraft Crash	Accidental impact of aircraft on site facilities	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process (i.e., impact of the crash) is immediate • Consequence is significant • Event frequency $\leq 10^{-6}$ per year	Staff reviewed two DOE documents submitted as responses to agreement PRE.03.01 and had a technical exchange on September 30, 2003.‡ DOE committed to collect additional information on aircraft-related activities to assess the potential hazard. Additionally, the reports will be revised as new information is obtained. Staff review of this hazard is provided in this section.
2	Inadvertent Future Intrusions (Human-Induced)	Human-induced inadvertent future intrusions with regard to 100-year preclosure period involve undetected surface access into potential repository facilities	Applicable to the hazards list* • Potential exists at Yucca Mountain • Rate of process is sufficient to affect 100-year preclosure period • Consequence is indeterminant; assumed significant • Annual event frequency is indeterminant; assumed significant • Will be considered in future safeguards and security analyses—a to-be-verified item	Staff has reviewed information provided by DOE and has no further questions at this time.
3	Intentional Future Intrusions (Human-Induced)	Human-induced intentional future intrusions with regard to 100-year preclosure period involve undetected surface access, sabotage, or both to potential repository facilities	Not applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficient to affect 100-year preclosure period • Consequence is indeterminant; assumed significant • Annual event frequency is indeterminant, assumed significant • Will be considered in future safeguards and security analyses—a to-be-verified item	Staff has reviewed DOE assessment and has no further questions at this time.

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

No.	Hazard	Hazard Definition	DOE Assessment	Staff Remark
4	Industrial Activity-Induced Accidents	Accidents resulting from industrial or transportation activities unrelated to the potential repository	Applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is sufficient to affect 100-year preclosure period • Consequence is indeterminant; assumed significant • Annual event frequency is indeterminant at this time; assumed significant • Not bounded by another analysis	DOE has not submitted the updated report about Nearby Facilities.
5	Loss of Offsite/ Onsite Power	Loss of electric power either generated or controlled by persons outside repository system or loss of power within the potential repository	Applicable to the hazards list • Potential exists at Yucca Mountain • Rate of the process is indeterminant at this time; assumed significant • Consequence is indeterminant; assumed significant • Annual event frequency is indeterminant at this time; assumed significant • Not bounded by another analysis	Staff has reviewed DOE assessment and has no further questions at this time.
6	Military Activity-Induced Accidents	Accidents resulting from military activities Nevada Test Site or Nellis Air Force Range	Applicable to the hazards list • Potential exists at Yucca Mountain • Rate of process is indeterminant at this time; assumed significant • Consequence of the process is indeterminant at this time; assumed significant • Annual event frequency is indeterminant at this time; assumed significant • Not bounded by another analysis	DOE has not submitted the updated report about Nearby Facilities.
7	Pipeline Accidents	Industrial pipeline transporting hazardous materials	Not applicable to the hazards list • No potential exists at Yucca Mountain because no industrial activities requiring pipelines containing hazardous materials exist or are planned to be located near the site	Staff has reviewed DOE assessment and has no further questions at this time.

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

No.	Hazard	Hazard Definition	DOE Assessment	Staff Remark
8	Undetected Past Intrusions	Past intrusions involve mining activities where deep shafts, drill holes, or tunnels may have been excavated	Not applicable to the hazards list • No potential exists at Yucca Mountain because site characterization provided sufficient assurance these types of activities would have been detected	Staff has reviewed DOE assessment and has no further questions at this time.

*CRWMS M&O. "MGR External Events Analysis." ANL–MGR–SE–000004. Rev. 00. Las Vegas, Nevada: CRWMS M&O. 1999.
†DOE. "Preliminary Preclosure Safety Assessment for Monitored Geologic Repository Site Recommendation." TDR–MGR–SE–000009. Rev. 00 ICN 03. Las Vegas, Nevada: DOE. 2001.
‡Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange on Aircraft Hazards Analysis, September 30, 2003." Letter (October 7) to J. Ziegler, DOE. Washington, DC: NRC. 2003. <www.nrc.gov/waste/hlw-disposal/public-involvement/mtg-archive.html#KTI>
§Bechtel SAIC Company, LLC. "Preclosure Safety Analysis Guide." TDR–MGR–RL–000002. Rev. 00. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2002
Note: CRWMS M&O* states this hazard is not applicable to the hazards list requiring additional analysis; however, Bechtel SAIC Company, LLC§ includes this hazard requiring additional analysis.

Some potential hazards are bounded by the analysis carried out for another hazard. For example, potential effects of a rainstorm are bounded by the analysis for potential flooding and its associated effects. Sandstorm effects are included with extreme wind and tornado wind. Effects of subsidence are included in seismic activity—surface and subsurface fault displacement. Using the screening process and bounding analyses, DOE reduced the list of possible natural hazards to the potential repository during the preclosure period to nine events: (i) debris avalanche; (ii) extreme wind, including sandstorms; (iii) flooding, including rainstorm and river diversion; (iv) landslide; (v) lightning; (vi) seismic activity, earthquake; (vii) seismic activity, surface fault displacement; (viii) seismic activity, subsurface fault displacement, including subsidence; and (ix) tornado winds and tornado missiles. DOE (2001b) stated that the site for surface facilities and the North Portal will be stabilized against debris avalanche and landslide. For preclosure safety analysis, these events have been grouped with flooding. Additionally, DOE grouped tornado wind loading with the extreme wind event and classified it as a tornado wind event. Tornado missile has been separately classified as a potential hazard.

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

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

4.1.3.2.1 Aircraft Crash Hazard

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

DOE conducted an analysis to estimate hazards to the potential repository at Yucca Mountain from potential aircraft crashes (CRWMS M&O, 1999c). DOE (CRWMS M&O, 1999c) used the suggested methodology of NRC (1981a) to estimate the probability of crash of an aircraft onto the potential high-level waste repository. Additionally, CRWMS M&O (1999c) used the methodology suggested in DOE–STD–3014–96 (DOE, 1996) to estimate the effective area of a particular structure and the crash rate data for different aircraft, developed by Kimura, et al. (1996). These guidance documents are commonly used for estimating the aircraft crash hazard to a facility and are sufficient for use in developing a potential license application.

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

(a) The facility-to-airport distance D is between 8 and 16 km [5 and 10 statute mi] and the projected annual number of operations is less than $500 \times D^2$, or the facility-to-airport distance D is greater than 16 km [10 statute mi] and the projected annual number of operations is less than $1{,}000 \times D^2$.

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

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

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

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

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

types of aircraft engaged in different operations. DOE agreed to develop a detailed analysis of the aircraft crash hazard using all types of aircraft flying in the vicinity of the potential site.

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

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

$$F = \frac{T\beta}{A_f} A_{\text{eff}} \tag{4.1.3-1}$$

where

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

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

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

$$F = \frac{N\lambda\pi}{L_f} A_{\text{eff}} \tag{4.1.3-2}$$

where

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

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

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

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

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

$$P_{FA} = N \times C \times \frac{A_{eff}}{W} \qquad (4.1.3\text{-}3)$$

where

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

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

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

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

where

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

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

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

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

where

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

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

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

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

- Nevada Test and Training Range

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

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

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

4.1.3.2.1.2.1 Nevada Test Site

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

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

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

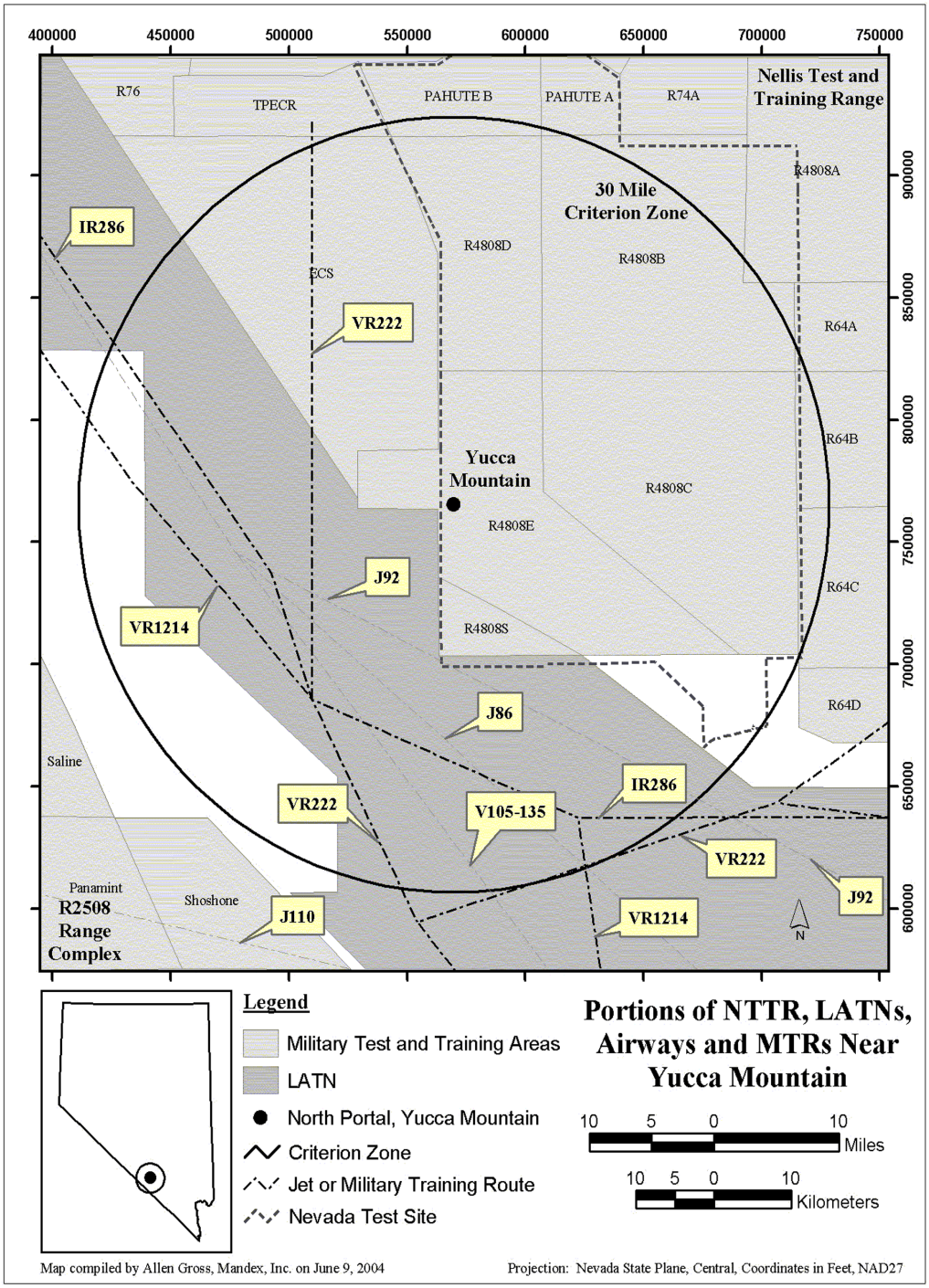

Figure 4.1.3-1. Portions of Nevada Test and Training Range (NTTR), Federal Airways, Military Training Routes (MTRs), and Low-Altitude Training Navigation Area (LATNs) (Adopted from Bechtel SAIC Company, LLC, 2002b, Figure 16)

4.1.3.2.1.2.2 Nevada Test and Training Range

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

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

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

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

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

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

4.1.3.2.1.2.3 R–2508 Range Complex

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

4.1.3.2.1.2.4 Airspace Supporting Nevada Test and Training Range

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

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

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

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

4.1.3.2.1.2.5 Airports and Airfields

Bechtel SAIC Company, LLC (2002b) listed all the airports within 160 km [100 mi] of the North Portal of the potential repository at Yucca Mountain. Airports and airfields with a high volume of traffic and within reasonable proximity to the potential repository site have been discussed with

more details about flight operations. Discussions of flight operations are given for Indian Springs Air Force Auxiliary Field, Tonopah Test Range Airfield, Nellis Air Force Base, Desert Rock Airport, Pahute Mesa Airstrip, Yucca Airstrip, Beatty Airport, Jackass Aeropark, Furnace Creek Airport, Imvite Airfield, McCarran International Airport, and North Las Vegas Airport.

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

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

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

4.1.3.2.1.2.6 Federal Airways

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

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

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

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

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

4.1.3.2.1.2.7 Other Activities

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

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

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

4.1.3.2.1.2.8 Summary

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

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

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

- Bechtel SAIC Company, LLC (2002b) does not provide any information regarding the number of each type of weapon used annually and safety precautions taken to ensure

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

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

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

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

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

  > "… it is expected that in a controllable situation at high altitudes, the pilot would eject between 3,048 and 4,572 m [10,000 and 15,000 ft] above mean sea level {approximately 1,524 and 3,048 m [5,000 and 10,000 ft] above ground level assuming a ground elevation of 1,524 m [5,000 ft] after unsuccessful restart." No basis for such an expectation has been presented.
  >
  > "… if the aircraft is at a high altitude and not in vertical descent, the pilot will regain control and a crash is averted." No basis for such an expectation has been presented.

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

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

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

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

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

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

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

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

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

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

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

- Basis for Assumption 3.15 in CRWMS M&O (2003a) that aircraft on the Nevada Test and Training Range flying near the proposed surface facility would be represented by "small attack, fighter, trainer aircraft" is not provided. DOE should clarify whether trainer aircraft

fly routinely near the potential repository and identify their missions. DOE should clarify also why the category of small military aircraft (all small attack, fighter, and trainer aircraft) would be more conservative when crash rates for F–16s, all single-engine, and all attack and fighter aircraft are higher. DOE should clarify whether uncertainties associated with the determination of the aircraft type flying in the vicinity of the proposed surface facilities have been appropriately considered in estimating the effective area of the buildings and in selecting the appropriate crash rate for the aircraft in the analysis.

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

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

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

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

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

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

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

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

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

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

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

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

Commercial aircraft use both McCarran International and North Las Vegas Airports. Limited chartered aircraft use Tonopah Airport (CRWMS M&O, 1999c). All three airports are more than 48 km [30 mi] from the potential site. Consequently, more than 900,000 annual takeoff and landing operations would be necessary at these airports to have a crash probability of 10^{-7} per year to the potential repository site. The number of commercial and general aviation aircraft currently taking off and landing at these airports is small and less than $1{,}000D^2$, where D is the distance between an airport and the site (NRC, 1981a). DOE estimated that the crash probability at the potential site from aircraft takeoff and landing at these three airports would be negligible.

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

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

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

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

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

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

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

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

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

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

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

4.1.3.2.2 Tornado Wind

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

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

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

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

CRWMS M&O (2003b) provided an analysis to develop the design basis straight and tornado wind speed. The design basis straight wind was developed based on limited site-specific data available for only 4 years (CRWMS M&O, 1997a) and compared with the DOE Standard DOE–STD–1020–2002 (DOE, 2002). Information contained in this report on development of

design basis straight and tornado wind speeds is classified as Official Use Only. The region is identified in SEI/ASCE 7–02 (Structural Engineering Institute/American Society of Civil Engineers, 2003) to be a special region requiring site-specific data to account for local topographical conditions. Therefore, DOE should use site-specific data, qualified in accordance with 10 CFR Part 63, Subpart G, for additional years to better quantify the design basis wind speed for structures, systems, and components important to safety.

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

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

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

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

4.1.3.2.3 Tornado Missiles Hazard

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

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

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

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

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

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

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

DOE has provided an analysis that is sufficient for use in developing a potential license application, including (i) selection of the design basis tornado, together with the supporting technical basis; (ii) selection of credible tornado missile characteristics for the waste package and other structures, systems, and components, together with the technical bases; and (iii) analysis of the effects of impact of the design basis tornado missiles or justification for excluding such tornado missiles as credible hazards.

4.1.3.2.4 Volcanic Hazards

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

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

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

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

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

DOE analyzed possible hazards of volcanic ash (i.e., tephra) to the potential repository and concluded that future eruptions could form a maximum 3-cm [1.2-in]-thick deposit at the potential repository site. The 3-cm [1.2-in] thickness is based on the assumption that a possible volcanic eruption during the preclosure period would be located at least 150 km [94 mi] from the potential repository site [i.e., Perry and Crowe (1987)]. Although large volcanoes located this distance away from the potential repository site could create deposits less than 3 cm [1.2 in] thick (e.g., Hoblitt, et al., 1987), new volcanoes characteristic of the area within 20 km [12.5 mi] of the site could produce appreciably thicker deposits. Based on analyses presented in Section 5.1.2.2.4.1, the probability of a new volcano forming within approximately 10 km [6.3 mi] of the potential repository site exceeds 1×10^{-6} per year. Tephra-fall deposits measured approximately 10 km [6.3 mi] from volcanoes analogous to those within 20 km [12.5 mi] of Yucca Mountain are on the order of 1–100 cm [1–39 in] thick (e.g., Sagar, 1997). These analog deposits appear reasonably comparable to poorly preserved deposits within several kilometers of Lathrop Wells volcano at the southern terminus of Yucca Mountain (e.g., Sagar, 1997). This topic was not discussed at the first technical exchange and management meeting for preclosure safety (Reamer, 2001). DOE has not presented information on the characteristics of tephra-fall deposits sufficient to support the analogy to sandstorms or to evaluate the potential effects on air circulation systems important to safety.

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

For the analysis of potential natural hazards to the potential repository, DOE concluded that a 3-cm [1.2-in]-thick volcanic tephra deposit is the worst-case event. However, the basis for this conclusion is not supported by available analyses or data. The 3-cm [1.2-in]-thick deposit cited by CRWMS M&O (1999b) applies only for a volcanic eruption occurring 150 km [94 mi] from the potential repository site (i.e., Perry and Crowe, 1987). DOE has not quantified the probability for this type of eruption to occur, but state that this probability is 1×10^{-6} per year (CRWMS M&O, 1999b).

DOE has not evaluated the probability for a basaltic volcanic eruption to occur at a location close enough to the potential repository site to possibly affect safe operation of surface facilities. Staff has performed an independent analysis of this probability for potential tephra-fall hazards. Tephra-fall deposits measured approximately 10 km [6.3 mi] from volcanoes analogous to those within 20 km [12.5 mi] of Yucca Mountain are on the order of 1–100 cm [0.4–39 in] thick (e.g., Sagar, 1997). These deposits increase in thickness to approximately 400 cm [157 in] within 1 km [0.6 mi] of the volcanic vent. Perry and Crowe (1987) also conclude that a 1-m [3.3-ft]-thick tephra deposit could occur approximately 3 km [1.9 mi] from a basaltic volcanic vent. As a first approximation, a new basaltic volcano located within 10 km [6.3 mi] of the potential repository site appears capable of creating a tephra deposit 1–100 cm [0.4–39 in] thick at the surface facility site.

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

Noncompacted, dry basaltic volcanic tephra has a bulk deposit density that can range 1,000–1,700 kg/m^3 [62–106 lb/ft^3] (e.g., Hill, et al., 1998; NRC, 1999). The density of these deposits can increase by roughly a factor of two when wet, depending on average grain size and sorting of the deposit (e.g., Sarna-Wojcicki, et al., 1981). Thus, a 1–100-cm [0.4–39-in]-thick tephra deposit could result in a load of 10–1,700 kg/m^2 [2–348 lb/ft^2] when dry, which could increase to 20–3,400 kg/m^2 [4–697 lb/ft^2] if the deposit was wet. Most of these potential roof loads are significantly larger than assumed to occur during the DOE screening of volcanic ash-fall as an applicable natural hazard for the potential repository site (CRWMS M&O, 1999b). This topic was outside the scope of the first technical exchange and management meeting for preclosure safety (Reamer, 2001).

DOE has not evaluated the probability for a basaltic volcanic eruption to occur at a location close enough to the potential repository site for lava flows to possibly affect safe operation of surface facilities (CRWMS M&O, 1999b). The probability of potential lava-flow hazards depends on important assumptions regarding location and extent of surface facilities, area upslope of these facilities capable of potential volcanic activity, and distribution of potential lava-flow lengths (e.g., Volcanism Working Group, 1990). Current uncertainties on the number, age, and location of possible volcanic events in the Yucca Mountain area (e.g., Hill and Stamatakos, 2002) likely affect the probability estimates for new volcano formation in the area upslope from potential

repository surface facilities. In addition, the physical characteristics of possible lava flows appear to exceed the design basis characteristics for other hazardous events (e.g., CRWMS M&O, 1999b). The current information has not been presented in a way to clearly support screening of lava flow hazards based on either low probability or negligible consequence. Thus, DOE should present a specific, transparent technical basis to justify exclusion of this event from the Category 2 event sequences list, or include this potential hazard in preclosure safety analyses.

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

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

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

4.1.3.2.5 Operational Hazards

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

The DOE operational hazard analysis methodology is documented in CRWMS M&O (1999a) and Bechtel SAIC Company, LLC (2002a). DOE used a combination of three hazard evaluation

techniques, namely, Energy Analysis, Energy Trace and Barrier Analysis, and Energy Trace Checklist (System Safety Society, 1997), to develop the generic checklist of hazards applicable to preclosure operations. The operational hazards have been categorized as (i) Collision/ Crushing, (ii) Chemical/Contamination/Flooding, (iii) Explosion/Implosion, (iv) Fire, (v) Radiation/Magnetic/Electrical/Fissile Materials, and (vi) Thermal. DOE divided the surface and subsurface facilities into several functional areas and applied the checklist to identify hazards in each functional areas.

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

The methodology proposed by DOE does not identify potential hazards resulting from failure of the software and hardware systems used in the remote operations. During the preclosure period, surface and subsurface facility operations are expected to be remotely controlled for various equipment (DOE, 2001c). Software reliability may be a significant factor in the safe operation of the potential Yucca Mountain repository (Eisenberg, 2001b). DOE should identify hazards and initiating events associated with reliability of hardware and software including reliability of remote systems used in the operations in preclosure safety analysis.

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

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

The NRC staff has reviewed (Lesher, et al., 2003) the DOE evaluation of the hazard potential of an underground transporter under runaway condition (CRWMS M&O, 2000c) in the North Ramp. Waste packages would be transported from the surface through the North Ramp to the emplacement drifts. The North Ramp is more than 2 km [1.24 m] long at a slope greater than −2 percent (CRWMS M&O, 2000c). The long downward slope of the North Ramp is the primary

concern for loss of braking and subsequent runaway of the transporter, leading to a derailment. CRWMS M&O (2000c) presented the results of a preliminary evaluation of the potential hazards of a tip-over and derailment of a transporter train while descending the North Ramp. CRWMS M&O (2000c) analyzed different scenarios in which transporter speed for runaway conditions initiated at different locations along the ramp was evaluated at the curve and before the transporter reaches the access main tunnel. This analysis confirmed that the maximum velocity of the transporter in the event of a runaway would exceed the speed at which the transporter would tip over and, therefore, runaway of the transporter is a potential hazard. This analysis also confirmed that the runaway transporter would derail only under worn track conditions. In addition, the staff review of the methodology used by CRWMS M&O (2000c) for calculating the transporter tipover speed indicates that the methodology is consistent with accepted industry and engineering practices.

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

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

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

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

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

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

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

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

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

addition, DOE should investigate whether failure of the speed controller would initiate a runaway event. The analysis presented in CRWMS M&O (2000c, 1997c,d) considered only point estimates for data used in the analysis. Future analysis should consider uncertainties in input data to assess sensitivity of the top event to these uncertainties.

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

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

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

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

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

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

4.1.3.3 Summary and Status

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

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

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

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

*The first Technical Exchange and Management Meeting for Preclosure Safety focused only on Aircraft Crash and Tornado Missiles Hazards [Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. ML021340719. Washington, DC: NRC. 2001. <www.nrc.gov/ waste/hlw-disposal/public-involvement/mtg-archive.html#KTI>]

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4.1.4 Identification of Event Sequences

4.1.4.1 Areas of Review

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

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

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

According to 10 CFR 63.2, those event sequences expected to occur one or more times before permanent closure of the geologic repository operations area are referred to as Category 1 event sequences. Other event sequences that have at least 1 chance in 10,000 of occurring before the permanent closure are referred to as Category 2 event sequences. DOE is required to demonstrate that for normal operations, Category 1 event sequences, and Category 2 event sequences meet the preclosure performance objectives stated in 10 CFR 63.111.

Event sequence analyses are based on development of scenarios that include an initiating event and the subsequent sequence of events associated with the failure of structures, systems, or components, including those produced by human actions or inactions. The scenarios are analyzed using event trees that results in a series of event sequences. DOE should ensure in its preclosure safety analysis that all possible event scenarios are considered and all event tree analysis account for uncertainty and variability in the estimated frequency and probability data. The probability of events that appear in event tree are quantified using actuarial data, fault trees, Bayesian analyses, expert judgement or other methods of estimation.

4.1.4.2 Staff Review of Available Information

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

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

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

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

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

- The DOE approach to categorize event sequences in low-temperature facility design is inconsistent and unclear. DOE states that if the preclosure period is extended beyond 100 years for a low-temperature operating mode, the preclosure period could be divided into two phases (Bechtel SAIC Company LLC, 2002). The Phase 1 period would consider waste-handling operations at the surface facility and waste-emplacement operations in the subsurface facility. Phase 2 would be the period after the emplacement and before final closure of the repository. The guide does not elaborate on how the two-phase preclosure period will affect the probabilities of occurrence or the associated event sequence categorization in the preclosure safety analysis. DOE should clearly present information on the categorization of the event sequences for the low-temperature facility design in a form consistent with the event sequence definition in 10 CFR 63.2 presented in Section 4.1.4.1.

4.1.4.2.2 Category 1 and 2 Event Sequences

Based on the preliminary design of the potential repository, DOE identified some event sequences reported in DOE (2001a) and associated reports (CRWMS M&O; 2000a, 1998, 1997a). As discussed in Section 4.1.2, the DOE facility design is being modified from the design presented in the site suitability report (DOE, 2001b), with significant changes in layout,

design, and functionality (DOE 2004a,b). Information is not available about the DOE analysis of event sequences from surface and subsurface operations for the revised facility design.

The DOE identification of event sequences that could potentially release radioactive material to the members of the public and facility workers is presented in DOE (2001a–c). The DOE preliminary hazards analysis identified nine natural and human-induced initiating events that could potentially cause radiological release (DOE, 2001a,Table 5-4). Adequacy of DOE identification natural and human-induced hazards are discussed in Section 4.1.3. DOE did not develop scenarios from these initiating events because DOE proposed to design, construct, and operate the potential repository to withstand these events (DOE, 2001b). In the future, when DOE submits the design, staff will review and evaluate adequacy of the DOE design, construction, and operations to prevent or mitigate natural and human-induced initiating events.

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

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

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

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

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

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

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

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

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

4.1.4.3 Summary and Status

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

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

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

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

*Limited general concerns were discussed in the DOE and NRC Technical Exchange and Management Meeting on Preclosure Safety [Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. ML021340719. Washington, DC: NRC. 2001. <www.nrc.gov/waste/hlw-disposal/public-involvement/mtg-archive.html#KTI>]. No agreements were reached.
†Not discussed at the DOE and NRC Technical Exchange on Preclosure Safety [Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. ML021340719. Washington, DC: NRC. 2001. <www.nrc.gov/waste/hlw-disposal/public-involvement/mtg-archive.html#KTI>].

4.1.4.4 References

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4.1.5 Consequence Analyses

4.1.5.1 Areas of Review

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

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

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

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

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

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

4.1.5.2 Staff Review of Available Information

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

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

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

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

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

DOE (2001a, Section 5.3.5.3) states, "… administrative controls will be in place to evacuate any members of the public who could potentially be located within the Yucca Mountain Project

Withdrawal Area but outside of the Preclosure Controlled Area Boundary (Figure 5-4) following a Category 2 (Design Basis Event, also referred to as an event sequence)." Because emergency planning after a Category 1 event sequence has not been addressed, there is a possibility the public could be present within the 11-km [6.8-mi] withdrawal area boundary. If evacuation plans are not established for Category 1 event sequences and members of the public are present within the 11-km [6.8-mi] withdrawal area boundary, Category 1 consequence analyses would be required to consider these individuals {i.e., dose calculations for members of the public within 11 km [6.8 mi]}. DOE should justify whether an emergency plan for members of the public is needed for a Category 1 event sequence. Considering that members of the public could be located within the withdrawal area boundary, DOE should provide additional justification for the selection of the 11-km [6.8-mi] distance to the withdrawal area boundary as the closest point that any member of the public could be located at the time of a postulated radiological release. This topic was discussed with DOE at a technical exchange on a preclosure safety analysis guide (Schlueter, 2002). DOE stated any members of the public located within the withdrawal area boundary would be evacuated to outside the site boundary. Staff has no additional comments at this time.

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

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

In analyzing radiation doses from Category 1 event sequences, DOE proposed to use input parameters based on long-term average data, such as annual average atmospheric dispersion factors and average waste characteristics for the source term (Bechtel SAIC Company, LLC, 2002). These long-term average data are appropriate for evaluating the chronic releases from normal operations of the surface and subsurface facilities. Releases from Category 1 event sequences will occur for a period of time that is short with respect to the time for which the parameter data were averaged (i.e., not chronic). Because 10 CFR 63.111(a)(2) refers to a preclosure standard in 10 CFR 63.204 of an annual dose to any real member of the public from Category 1 event sequences from normal operations that must not be exceeded in any year, parameters based on appropriate short-term data may have to be used to enable a demonstration with reasonable assurance the parameters used in the calculations are appropriate for the scenario used. DOE may have to use short-term data for atmospheric dispersion and other parameters and provide a technical justification for the appropriateness of all data used for the dose calculations. This topic was discussed with DOE at a technical exchange on a preclosure safety analysis guide (Schlueter, 2002). DOE stated that acute 2-hour dispersion factors will be used to demonstrate that no single Category 1 event sequence will exceed the preclosure standard in 10 CFR 63.204. Annual average dispersion factors, however, would be used when evaluating the aggregate sum of Category 1 event sequences and normal operational events. DOE has indicated that it intends to model Category 1 events as chronic releases.

CRWMS M&O (2000, Attachment IV, Section 2.2) states the dose coefficients for external exposure are based on soil contaminated to a depth of 15 cm [5.9 in]. Using this contamination depth may underestimate the external doses from increased self-attenuation by the contaminated soil, compared with a thinner contamination layer. Each airborne release would result in surface depositions of radionuclides, which will slowly migrate deeper into the soil with time. Attachment IV presents the dose calculation methodology for Category 1 event sequences, for which an exposure of 1 year is assumed. Studies of the depth distribution of radionuclides in soil for depositions less than 1 year shows most of the radionuclide inventory is contained within the upper few centimeters of soil (International Commission on Radiation Units and Measurements, 1994). Although the deeper contaminated layer would seem appropriate for plowed fields, a thinner contaminated layer should be considered for the external dose calculations. Selection of a normalized dose conversion (Sv yr^{-1} per Bq m^{-3}) based on a 15-cm [5.9-in] contaminated layer in U.S. Environmental Protection Agency (EPA) (1993) could be reasonable and considered to be conservative because a thicker contaminated layer added to the source term increases the normalized dose conversion factor. The uniform distribution assumption, however, would reduce the activity concentration (Bq m^{-3}) and result in lower estimates of the external dose. It is unclear if the expected activity of radionuclides deposited on the soil is distributed uniformly to a depth of 15 cm [5.9 in]. This topic was discussed with DOE at a technical exchange on a preclosure safety analysis guide (Schlueter, 2002). DOE stated that, for consequence calculations, radionuclides deposited on the soil are assumed to be distributed uniformly to a depth of 15 cm [5.9 in]. DOE should provide a technical basis to support its assumption.

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

CRWMS M&O (2000, Page 11) describes the local deposition factor as the fraction of the airborne release fraction that is deposited locally within the Waste Handling Building. It appears from this definition, a local deposition factor value of 1 would be equal to 100 percent of the material released being deposited in the Waste Handling Building, implying zero release from the Waste Handling Building. The local deposition factor is set at a value equal to 1 to maximize releases from the Waste Handling Building as part of Assumption 3.20, which was inconsistent with this definition. Furthermore, Eq. (11) of CRWMS M&O (2000) calculates the total release fraction to the environment and uses the local deposition factor directly to calculate the release fraction instead of using one minus the local deposition factor. Staff would prefer (i) defining the local deposition factor as a release or leakage factor rather than a deposition factor or (ii) modifying Eq. (11) and Assumption 3.20 to be consistent with the actual definition of the local deposition factor.

4.1.5.2.1.3 Assessment of the Methodology for Compliance with Regulatory Requirements

Although the DOE approach for demonstrating compliance applies a frequency weighting to the doses from Category 1 event sequences, the approach does not consider the potential for multiple Category 1 event sequences occurring in a single year. 10 CFR 63.111(a)(2) refers to

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

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

DOE (2001a, Section 5.3.5.3) states staff located on the Nevada Test Site and the Nellis Air Force Range are government workers on government property, subject to evacuation if required, and, therefore, not considered part of the public. 10 CFR 20.1003 defines occupational dose as "… the dose received by an individual in the course of employment in which the individual's assigned duties involve exposure to radiation or to radioactive material from licensed and unlicensed sources of radiation, whether in the possession of the licensee or other person." 10 CFR 20.1003 defines member of the public as any individual except when that individual is receiving an occupational dose. It is acknowledged that administrative controls should be more effective for individuals on government property compared with those not on

government property. Unless the assigned duties of all staff located on the Nevada Test Site and the Nellis Air Force Range involve exposure to radiation or to radioactive material, however, those staff should be considered members of the public. If the duties of those workers are deemed to involve exposure to repository-related radiation, the survey and monitoring requirements of Subpart F to 10 CFR Part 20 and the reporting requirements of Subpart M to 10 CFR Part 20 should be complied with. Consequently, staff located on the Nevada Test Site and the Nellis Air Force Range should be treated as members of the public unless trained, monitored, and protected by an established radiation protection program, or DOE should provide additional information about the classification of government workers as radiation workers. This topic was discussed with DOE at a technical exchange on a preclosure safety analysis guide (Schlueter, 2002). Based on information available at that time, DOE stated the staff located on the Nevada Test Site and the Nellis Air Force Range would be monitored and trained in accordance with an established radiation protection program.

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

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

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

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

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

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

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

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

Failed fuel (e.g., with cladding damage, debris, or pieces of fuel present) is to be placed in single-element disposable canisters. The consequences from failed fuel is assumed to be bounded by the radiological consequences from commercial spent nuclear fuel. The release fraction calculations do not consider failed fuel (CRWMS M&O, 1999b), which may have different particulate release fractions and source terms. The potentially different particulate release fractions from failed fuel should be addressed to support the argument that failed fuel was bounded by commercial spent nuclear fuel. For significant releases of radioactive material, credit can be taken for the mitigation of doses from pathways associated with radionuclides deposited on the ground and potential long-term exposure (NRC, 2003, Section 2.1.5.2.3, Review Method 2, page 2.1-38). This topic was discussed with DOE at a technical exchange on a preclosure safety analysis guide (Schlueter, 2002). DOE stated an update to the calculation is in progress which will address the release fractions for failed commercial spent nuclear fuel. In this update, DOE should evaluate the release fractions for wasteforms other than commercial spent nuclear fuel and include the potential differences in release fractions. In addition, DOE could provide a comparison of these alternative source terms with those used for commercial spent nuclear fuel.

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

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

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

4.1.5.2.2.3 Assessment of the Methodology for Compliance with Regulatory Requirements

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

4.1.5.3 Summary and Status

4.1.5.3.1 Normal Operations and Category 1 Event Sequences

At the first Technical Exchange and Management Meeting for Preclosure Safety (Reamer, 2001), the NRC staff agreed with the DOE general methodology for consequence analyses. Because the meeting focused on general methodologies, many specific comments were not raised. No specific agreements were reached on the consequence analyses.

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

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

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

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

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

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

- A technical basis to support the assumption that any airborne radionuclides deposited on the soil are uniformly distributed to a depth of 15 cm [5.9 in]
- Direct exposure dose calculations from external sources not related to the release of radioactive material
- The method for determining which combinations of Category 1 event sequences will be compared with the preclosure standard or regulatory limits

4.1.5.3.2 Category 2 Event Sequences

At the first Technical Exchange and Management Meeting for Preclosure Safety (Reamer 2001), the NRC staff agreed with the DOE general methodology for consequence analyses. Because the meeting focused on general methodologies, many specific comments were not raised. No specific agreements were reached on the consequence analyses.

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

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

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

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

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

4.1.5.4 References

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

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

4.1.6.1 Areas of Review

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

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

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

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

According to 10 CFR 63.2, structures, systems, and components important to safety are those engineered features whose functions are to (i) provide reasonable assurance that high-level waste can be received, handled, packaged, stored, emplaced, and retrieved without exceeding the requirements of 10 CFR 63.111(b)(1) for Category 1 event sequences or (ii) prevent or mitigate Category 2 event sequences that could result in radiological exposures exceeding the values specified in 10 CFR 63.111(b)(2) to any individual located on or beyond any point on the boundary of the site. In addition, structures, systems and components, must be identified as important to safety if they are required to comply with 10 CFR Part 20 dose limits for Category 1 event sequences. Structures, systems, and components required to maintain compliance with 10 CFR Part 20 and 10 CFR Part 63 limits during normal operations, but not during Category 1 event sequences, are not considered important to safety. For determining those structures, systems, and components important to safety, Table 4.1.6-1 presents the dose limits required by 10 CFR 63.111 for Categories 1 and 2 event sequences for both public and worker safety.

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

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

4.1.6.2 Staff Review of Available Information

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

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

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

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

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

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

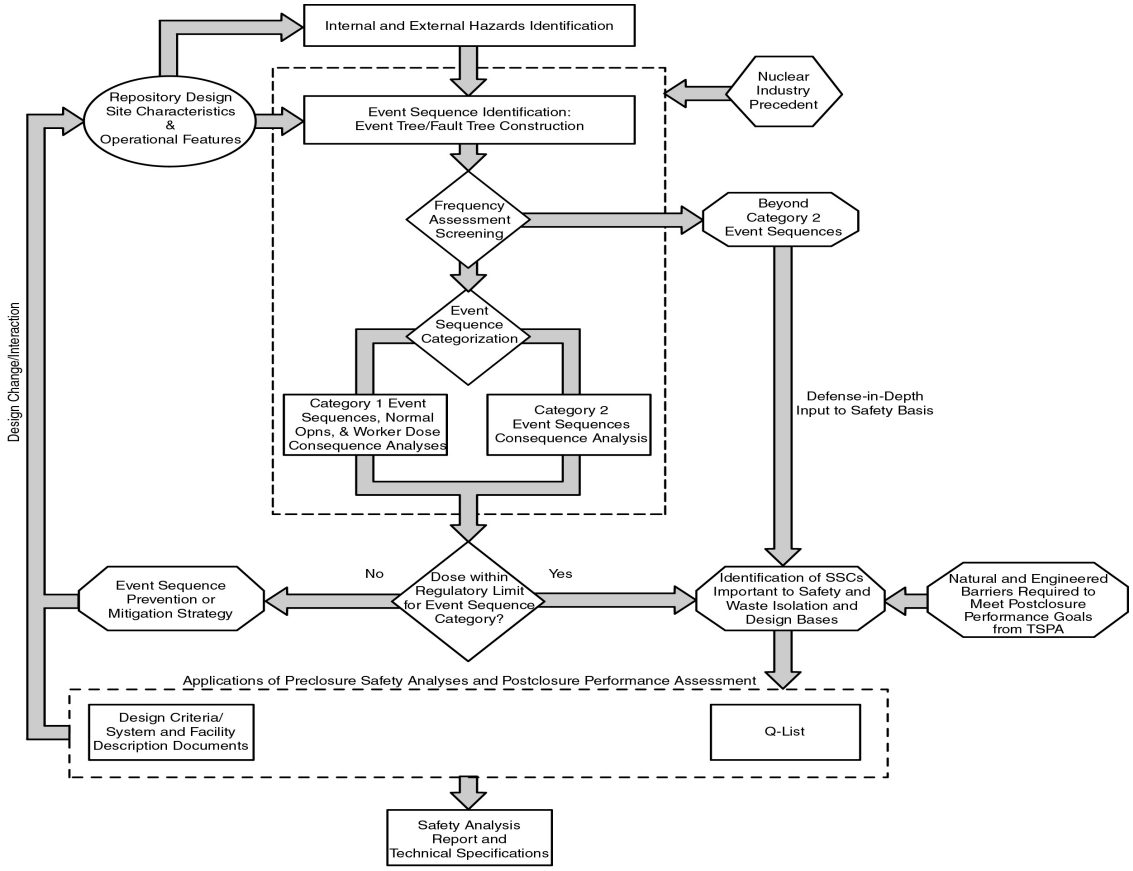

Figure 4.1.6-1. Overview of the DOE Preclosure Safety Analysis Process (DOE, 2003)

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

Based on frequencies for the Category 1 event sequences (DOE, 2001a), it can be expected that, for the entire preclosure operational period, more than one Category 1 event sequence will occur within a single year. An annual dose limit of 0.15 mSv [15 mrem] is specified in 10 CFR Part 63 for members of the public and the aggregated radiation exposure limits in the restricted and unrestricted areas must meet the dose limits in 10 CFR 63.111(a).

DOE proposes to classify individual structures, systems, and components for Category 1 event sequences based on the summation of three terms (Bechtel SAIC Company, LLC, 2002): (i) annual dose from normal operations of the surface and subsurface facilities; (ii) the frequency-weighted dose from all Category 1 event sequences; and (iii) the dose from a single Category 1 event sequence involving the failure of that particular structure, system, or component.

The current Q-List (Bechtel SAIC Company, LLC, 2003) includes structures and systems used in the geologic repository operations area. The systems are divided further to consider systems or operations. The current Q-List of structures, systems, and components important to safety is based on a design that is evolving. As discussed in Section 4.1.2, the DOE facility design is

being modified from that presented in the site suitability report (DOE, 2001b) with significant changes in layout, design, and functionality (McDaniel, 2004; Board, 2004). Consequently, this Q-List may change as the design and the preclosure safety analysis mature. Lacking the final design interaction, no attempt has been made to review the adequacy and completeness of the Q-List until DOE submits the potential license application.

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

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

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

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

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

structures, systems, and components required to provide the credited function(s) (Reamer, 2004).

4.1.6.3 Summary and Status

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

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

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

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

*Limited general concerns were discussed in the DOE and NRC Technical Exchange and Management Meeting on Preclosure Safety in Las Vegas, Nevada. [Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. ML021340719. Washington, DC: NRC. 2001. <www.nrc.gov/waste/hlw-disposal/public-involvement/mtg-archive.html#KTI>], and Important to Safety Technical Exchange [Reamer C.W., "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Important to Safety Technical Exchange, (May 12, 2004)." Letter (June 18) to J.D. Ziegler, DOE. ML041700192. Washington, DC: NRC. 2004. <www.nrc.gov/reading-rm/adams.html>], in Las Vegas, Nevada. No agreements were reached. The DOE facility design is not finalized; therefore, the Q-List was not discussed.
†Not discussed at the DOE and NRC Technical Exchange on Preclosure Safety [Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. ML021340719. Washington, DC: NRC. 2001. <www.nrc.gov/waste/hlw-disposal/public-involvement/mtg-archive.html#KTI>].

4.1.6.4 References

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

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

4.1.7.1 Areas of Review

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

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

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

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

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

  (3) Suitable shielding

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

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

  (6) Means to prevent and control criticality

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

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

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

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

(11) Means to provide reliable and timely emergency power to instruments, utility service systems, and operating systems important to safety if there is a loss of primary electric power

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

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

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

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

4.1.7.2 Staff Review of Available Information

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

4.1.7.2.1 Surface Facilities

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

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

- Design Methodologies

- Geologic Repository Operations Area Design and Design Analyses

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

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

4.1.7.2.1.2 Design Methodologies

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

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

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

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

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

    — In the first additional analysis, “beyond design basis ground motions,” the structures, systems, and components will be evaluated at larger ground motion levels (2,000 and 10,000-year return period ground motions for design basis ground motion levels 1 and 2 structures, systems, and components). The beyond design basis analyses will compare the resulting linear/nonlinear elastic

seismic demands with high-confidence strength capacities. If seismic demands exceed the strength capacities, the structures, systems, and components will be redesigned.

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

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

4.1.7.2.1.3 Geologic Repository Operations Area Design and Design Analyses

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

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

4.1.7.2.1.3.2 Aging Facilities

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

4.1.7.2.2 Subsurface Facilities

Subsurface facilities consist of (i) portals and access ramps, (ii) access mains, (iii) emplacement drifts, (iv) openings to support the subsurface ventilation, and (v) openings to support monitoring and performance confirmation testing (CRWMS M&O, 2000a). The portals and access ramps (North Portal, South Portal, North Ramp, and South Ramp) of the existing Exploratory Studies Facility would be integrated into the potential repository and would connect the surface and subsurface facilities through the access mains (CRWMS M&O, 2000a). The meeting minutes of the February and May 2004 technical exchanges (Schlueter, 2004; Reamer, 2004) contain conceptual drawings of these facilities.

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

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

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

The three specific areas of staff review are

- Relationship Between the Design Criteria and Design Bases and the Regulatory Requirements
- Design Methodologies
- Design and Design Analyses for Structures, Systems, and Components Equipment, and Safety Controls

4.1.7.2.2.1 Waste Transportation and Emplacement Equipment

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

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

The design of the subsurface facilities incorporates subject matter previously reviewed within the framework of two subissues of the Repository Design and Thermal-Mechanical Effects Key Technical Issue (NRC, 2000a): Subissue 2, Seismic Design Methodology; and Subissue 3, Thermal-Mechanical Effects on Underground Facility Design and Performance. In the subsequent sections, applicable portions of these subissues are considered but are not specifically identified.

The emplacement drifts are an array of horizontal tunnels trending approximately east-northeast–west-southwest (252° azimuth) between the east and west mains. Each drift will have a diameter of 5.5 m [18.5 ft] and will be separated from the adjacent drifts by a center-to-center distance of 81 m [265.7 ft]. The transition from the access mains to the

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

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

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

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

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

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

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

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

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

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

DOE has not provided NRC with information regarding

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

4.1.7.2.2.3 Ventilation System

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

4.1.7.2.3 Waste Package and Other Engineered Barriers

In addition to the waste package, other components of the engineered barrier system that may be used during preclosure operations at the potential geologic repository include a drip shield, drift invert, waste package pallet, and backfill. However, it is not clear whether these barriers will be considered as important to safety during the preclosure period. Designs of the waste package and engineered barrier system components incorporate subject matter previously reviewed within the framework of four subissues of the Container Life and Source Term Key Technical Issue (NRC, 2001) and Subissue 1, System Description and Demonstration of Multiple Barriers, of the Total System Performance Assessment and Integration Key Technical Issue (NRC, 2000b). The specific applicable Container Life and Source Term Key Technical Issue subissues are Subissue 1, Effects of Corrosion Processes on the Lifetime of the Containers; Subissue 2, Effects of Phase Instability of Materials and Initial Defects on the Mechanical Failure and Lifetime of the Containers; and Subissue 6, Effects of Alternate Engineered Barrier Subsystem Design Features on Container Lifetime and Radionuclide Release from the Engineered Barrier Subsystem. The Design of Structures, Systems, and Components and Safety Controls that are safety related for the waste package and engineered barrier system is also related to Container Life and Source Term Key Technical Issue Subissue 5, Effect of In-package Criticality on Waste Package and Engineered Barrier Subsystem Performance.

The DOE site recommendation reference design (CRWMS M&O, 1999a) indicates several variations of the basic waste package design will have to be implemented to accommodate the different types of spent nuclear fuel and high-level waste glass. The basic waste package design concept uses two concentric cylinders of different metallic materials. The outer container or barrier will be made from a corrosion-resistant nickel alloy referred to as Alloy 22,

surrounding an inner container made of Type 316 nuclear grade stainless steel (CRWMS M&O, 2000f). Fabrication processes used to construct the waste packages (e.g., forming, welding, and stress-relieving operations) may alter performance of the container materials. The waste packages will be supported by pallets and emplaced in a horizontal orientation within the repository drifts. In addition to the spent nuclear fuel and high-level waste, the waste packages also will contain several of engineered components designed to provide criticality control, provide structural support, and transfer heat from the waste package interior to the waste package surface (CRWMS M&O, 2000g). Each waste package will rest on an Alloy 22 emplacement pallet made of two V-shaped supports connected by hollow tubes with square cross sections. The waste package pallets will, in turn, rest on the drift invert. An inverted U-shaped drip shield, fabricated with titanium-palladium alloys (Titanium Grades 7 and 24), will be placed over the waste packages and, by interlocking the individual drip-shield units, will extend continuously over the entire length of the emplacement drifts. The drip shields will rest on the drift invert and provide shielding for both the top and sides of the waste packages (CRWMS M&O, 2000h). The current repository reference design does not include engineered backfill. Drift degradation, however, may produce natural backfill in the postclosure period (Ofoegbu, 2000).

Microstructural changes (e.g., ordering transformation, intermetallic precipitation, and metalloid segregation) that may affect the mechanical properties of the containers could result from welding operations, weld repairs, and postweld treatments. The ductility, fracture toughness, and impact strength of Alloy 22 are unlikely to be significantly affected by the fabrication processes necessary to construct the waste package outer container (Dunn, et al., 2004). Mechanical failure of the container and subsequent penetration of water are necessary conditions for a criticality event. At present, criticality has been screened out on the basis of low probability. The technical basis for this screening argument is the anticipated long life of the waste packages. In subsequent sections of this report, applicable portions of these subissues are considered, and the current resolution status is provided.

Design descriptions as well as details of the fabrication, inspection, repair, and emplacement of the waste package and engineered barrier system components are necessary to evaluate the DOE preclosure safety strategy. DOE provides information for the current designs of the waste packages and engineered barrier system components in CRWMS M&O (2000 f–h). Fabrication methods that may be used to construct the waste packages and engineered barrier system components also are provided in Bechtel SAIC Company, LLC (2001b,c). DOE has indicated that the potential license application design will use a Type 316 nuclear grade stainless steel inner container constructed to the requirements of the 2001 (with 2002 addenda) ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC (ASME, 2001a). The Alloy 22 outer container will not be an ASME stamped vessel (Brown, 2003a).

This section of the Integrated Issue Resolution Status Report has been prepared based on a review of these reports, other DOE documents, and discussions at the first preclosure technical exchange (Reamer, 2001a). Agreements were reached on specific topics concerning waste package design, inspection methods, variations in the mechanical properties of the waste package materials, and the effects of fabrication and repair on waste package performance. The postclosure performance of the engineered barrier system is addressed in Sections 5.1.3.1 and 5.1.3.2.

Overall, the available information, along with agreements reached between DOE and NRC (Schlueter, 2000; Reamer, 2001a,b,c), is sufficient to expect that the necessary information needed to assess the design of the waste package and engineered barrier system structures, systems, and components and safety controls will be available at the time of a potential license application. The designs of the waste package, drip shields, and the waste package pallet have yet to be finalized. In addition, the fabrication, remediation, and waste package and drip shield emplacement methods are currently being developed.

4.1.7.2.3.1 Waste Package Design Description

The waste package design consists of two concentric cylinders (i.e., disposal containers, fabricated from plate material). The meeting minutes of the February and May 2004 technical exchanges (Schlueter, 2004; Reamer, 2004) contain conceptual drawings of the waste package. The inner disposal container will be fabricated using Type 316 nuclear grade stainless steel, a minimum 50 mm [1.97 in] thick (Bechtel SAIC Company, LLC, 2001b). The inner disposal container will fit inside the outer disposal container constructed from Alloy 22. A radial gap of 0–4 mm [0–0.16 in] will be used between the inner and outer disposal containers to allow for differential thermal expansion to occur without introducing thermally induced stresses. The axial gap between the inner and outer disposal containers, which may be more important as far as differential thermal expansion stresses are concerned, is 10 mm [0.39 in] (CRWMS M&O, 2000f). The technical basis for the temperature used to establish these gaps, however, is not provided. Type 316 nuclear grade stainless steel is selected for the inner disposal container to provide mechanical integrity to the waste package during both the preclosure and postclosure periods of the potential repository. The selection of Alloy 22 as the outer disposal container material is based on the resistance of this nickel-chromium-molybdenum-tungsten alloy to both localized corrosion and stress corrosion cracking in chloride-containing environments. Placement of the corrosion-resistant Alloy 22 container on the outside of the Type 316 nuclear grade stainless steel is designed to provide long-term protection of the inner container (CRWMS M&O, 2000g; DOE, 2002).

Several waste package configurations are needed to encapsulate the various commercial spent nuclear fuel wasteforms (CRWMS M&O, 2000g). These configurations include designs for pressurized water reactor fuel containing either 12 or 21 pressurized water reactor assemblies with absorber plates and 21 pressurized water reactor assemblies with control rods. Two waste package configurations are required for boiling water reactor fuel that contains either 44 boiling water fuel assemblies with absorber plates or 24 boiling water reactor fuel assemblies with thick absorber plates. Moreover, additional waste package configurations are for the disposal of defense high-level waste and DOE-owned spent nuclear fuel.

The waste package will be constructed by rolling the plate materials into cylinders. A longitudinal weld will be used to complete the cylinder. Welding also will be used to connect two cylinders together to provide sufficient length for the spent nuclear fuel and high-level waste. The bottom lids of the disposal containers also are welded in place. Although the Type 316 nuclear grade stainless steel inner disposal container provides mechanical integrity to the waste package, the Alloy 22 outer disposal container will be required to sustain loads during lifting and transport. Lifting trunnions will be attached to the outer surface of the Alloy 22 disposal container to facilitate the necessary lifting and transport operations. Design of the inner disposal container will be specific to the waste package contents. Unique internal support

structures are required for pressurized water reactor fuel, boiling water reactor fuel, and high-level waste glass (CRWMS M&O, 2000g; DOE, 2002). After the internal support structure is constructed inside the inner disposal container, the inner Type 316 nuclear grade stainless steel container will be inserted into the Alloy 22 outer disposal container. After loading, the disposal containers will be sealed. The single closure lid for the Type 316 nuclear grade stainless steel inner container is held using a spread ring with a seal weld. A dual-closure lid design is used for the Alloy 22 outer disposal container (Brown, 2003b).

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

4.1.7.2.3.2 Waste Package Internal Components Design Description

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

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

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

4.1.7.2.3.3 Drip Shield Design Description

The description of the drip shield, its fabrication sequence, and the emplacement methods are not complete. The design of the drip shield is still being developed (Bechtel SAIC Company, LLC, 2001b). The current drip shield design calls for a Titanium Grade 24 support structure

covered with 15-mm [0.59-in]-thick Titanium Grade 7 plate (CRWMS M&O, 2000h). Individual segments of the drip shield are connected using a vertically sliding interlock configuration. The drip shield will be installed at the end of the preclosure period. The intended function of the drip shield is to divert any dripping water from contacting the waste packages and to protect the waste package against rockfall during the postclosure period (Bechtel SAIC Company, LLC, 2001b; DOE, 2002).

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

4.1.7.2.3.4 Waste Package Pallet

The waste package pallet is designed using Alloy 22 plate material (CRWMS M&O, 2000h). Each waste package pallet has two V-shaped supports connected by hollow, square cross-sectional tubing. Two sizes of emplacement pallets will be required to accommodate the different waste package lengths.

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

4.1.7.2.3.5 Disposal Container Fabrication and Closure

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

Filler materials used in welding processes should conform to the requirements specified in ASME (2001c, Section II, Part C) or equivalent. For the Type 316 nuclear grade stainless steel inner container, the filler material will be selected to control the delta ferrite content of the as-deposited weld metal. A ferrite number between 5 and 15, determined by Magna-gage measurements, is required in the inner disposal container fabrication welds (Bechtel SAIC Company, LLC, 2001a). The weld filler material for the Alloy 22 outer container will be ERNiCrMo-14 (Brown, 2003c).

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

The remaining fabrication steps for the disposal containers are specific to the inner and outer containers. For the Type 316 nuclear grade stainless steel inner container, the bottom lid and the internal parts, such as baskets, corner guides, and separator plates, are installed. For the Alloy 22 outer container, an assembly support ring used to support the Type 316 nuclear grade stainless steel inner containers is welded into place, and the welds are machined to allow the inner cylinder to be properly installed inside the outer container. The bottom lid is then fit and welded in place. The trunnion collar sleeve is installed on the outside of the Alloy 22 outer container and welded in place. Solution annealing is performed at approximately 1,125 °C [2,057 °F] to eliminate residual stresses created during fabrication processes. Water quenching will be used to reduce the temperature of the Alloy 22 outer container to below 800 °C [1,472 °F] in approximately 4 minutes (Bechtel SAIC Company, LLC, 2001b). DOE has not developed design criteria for residual stress mitigation by solution annealing.

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

The combination of cold work used in forming and machining operations and elevated temperature exposures caused by welding and annealing processes of the Alloy 22 waste package outer container may precipitate topologically close-packed phases. During the solidification of the weld metal, molybdenum and tungsten segregate to the interdendritic regions leaving the dendrite core rich in nickel (Cieslak, et al., 1986a,b). The depletion of nickel and enrichment of molybdenum and tungsten in the interdendritic regions promote the precipitation of topologically close-packed phases. The composition of all the topologically close-packed phases, including σ, μ, and P phases, can contain more than 30-percent molybdenum (Raghavan, et al., 1984). The high concentration of molybdenum in these phases results in a depletion of molybdenum adjacent to the precipitates that reduces the resistance of the alloy to localized corrosion. Because the formation of the precipitates preferentially occurs in the weld regions and in the intergranular regions of the heat-affected zone adjacent to the welds, localized corrosion in the form of interdendritic and intergranular corrosions may be a consequence of the precipitation of topologically close-packed phases (Heubner, et al., 1989). The ductility of σ, μ, and P phases is typically low compared with the austenitic matrix of the nickel-base alloy (Matthews, 1976; Tawancy, 1996). As a result, the precipitation of

topologically close-packed phases may reduce the ductility and impact strength of the alloy, particularly in welds or in the heat-affected zones of the welds. DOE has indicated that during the preclosure period, however, corrosion may not be a significant issue.

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

The effect of topologically close-packed phase precipitation on the mechanical properties of Alloy 22 has been reported at temperatures in the range 593–760 °C [1,099–1,400 °F] (CRWMS M&O, 2000k; Rebak, et al., 2000; Summers, et al., 2002). Table 4.1.7-1 combines the mechanical properties and corrosion rates reported by Rebak, et al. (2000) with the microstructural observations of the material after isothermal exposures. It is apparent the corrosion rate increases in response to partial grain boundary precipitation. In contrast, the Charpy impact energy for Alloy 22, after thermal aging that results in partial coverage of the grain boundaries with topologically close-packed phase precipitates, is quite high and similar to

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

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

*ASTM International. "Standard Test Methods of Detecting Susceptibility to Intergranular Corrosion in Wrought, Nickel-Rich, Chromium-Bearing Alloys." ASTM G 28-97. 2001 Annual Book of ASTM Standards. Vol. 3.02. West Conshohocken, Pennsylvania: ASTM International. 2001.

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

At 760 °C [1,400 °F], an exposure of 10 hours is required to decrease the Charpy impact energy to 203 J [150 ft·lb] (Rebak, et al., 2000). Assuming the extrapolation of activation energy is valid at temperatures greater than 760 °C [1,400 °F], an isothermal exposure after 1 hour at 870 °C [1,598 °F] would decrease the Charpy impact energy from 360 to 203 J [266 to 150 ft·lb]. Dunn, et al. (2004) reported welded Alloy 22 remains ductile and resistant to fracture even after thermal aging at 870 °C [1,598 °F] for a period of 1 hour. Solution annealed welds contained a slight volume fraction of topologically close-packed phases but retained high fracture toughness and impact strength. These results have led DOE to conclude that waste package fabrication processes will not significantly degrade the mechanical properties.

Variations in the composition of the Alloy 22 plate and the filler metal used in the welding process may alter the kinetics of topologically close-packed phase precipitation. Systematic studies about the effects of compositional variations of Alloy 22 on thermal stability show that molybdenum, tungsten, and iron decrease the phase stability of the alloy and increase the precipitation kinetics of topologically close-packed phases (Heubner, et al., 1989). The compositional specifications for Alloy 22 include 12.5 to 14.5-percent molybdenum, 2.5 to 3.5-percent tungsten and 2 to 6-percent iron. Trace elements such as sulfur, cobalt, and carbon also may alter thermal stability and mechanical properties of the base and filler metals.

Additional information is needed to determine the effects of microstructural and compositional variations of the plate and filler materials on the thermal stability and mechanical properties of the Alloy 22 waste package outer container. This information may result in unanticipated variations in waste package corrosion resistance and mechanical properties. To address these concerns, DOE agreed (Reamer, 2001b) to provide justification that the ASME Boiler and Pressure Vessel Code case for the use of Alloy 22 results in acceptable waste package mechanical properties considering allowed microstructural and compositional variations of the Alloy 22 base metal and the allowed compositional variations in the weld filler metals used in fabrication of the waste packages. In addition, DOE agreed (Reamer, 2001b) to provide justification that the mechanical properties of the disposal container fabrication and waste package closure welds are adequately represented considering the (i) range of welding methods used to construct the disposal containers, (ii) postweld annealing and stress mitigation processes, and (iii) postweld repairs. DOE indicates future work will include developing and testing of welding, heat treating, and inspecting equipment and processes.

In summary, microstructural and compositional variations of the plate material and filler metals may alter the kinetics of topologically close-packed phase precipitation because of welding and thermal exposures. Fabrication processes can alter the microstructure decrease localized corrosion resistance. The formation of topologically close-packed phases as a result of fabrication processes is unlikely to significantly alter the mechanical properties of the waste packages, however, the effects of compositional variations in the base and filler metals should be evaluated. With the DOE agreement to provide the additional information, sufficient information should be available at the time of a potential license application.

4.1.7.2.3.6 Nondestructive Evaluation of the Disposal Container and Closure Welds

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

The waste package design for the potential license application will have a single Type 316 nuclear grade stainless steel closure lid and dual Alloy 22 closure lids (Brown, 2003b). Because of the high radiation fields that will be present after the containers are loaded, remote welding processes are required to close the disposal containers. Before installation of the closure lid, the prepared surfaces will be visually inspected using a remote camera, followed by a tactile coordinate measurement using a coordinate measuring machine. The coordinate measuring machine will locate the center of the disposal container, relative to the closure gantry manipulator coordinate system, and determine disposal container cylindricity. This machine will provide a redundant check of the visual inspection for the weld preparations. Three remote cameras (lead, trail, and inspection) on the robotic arm welder will provide real time weld inspection with digital image processing and machine vision techniques that will identify problems with the welding process. It may be possible to perform some repairs at the weld station, and then resume the welding process. If the repair requires extensive machining, the disposal container will be moved to a repair station (Bechtel SAIC Company, LLC, 2001c).

The inner disposal container lid, made of Type 316 nuclear grade stainless steel, will be held using a shear ring with a seal weld. The shear ring will be assembled from three or four segments and welded in place. The welding robotic arm will have the ability to perform a full circumferential weld with a rotational range greater than 360 degrees. All critical parameters will be recorded in process, and the closure cell control system will notify the operator immediately of any parameter anomalies. After welding the inner lid, the inner container will be evacuated and filled with inert helium gas via a purge port. The inner container will then be leak tested to confirm the integrity of the welds. The process sequence flowchart for disposal container closure (Bechtel SAIC Company, LLC, 2001c) indicates DOE does not plan to conduct a nondestructive examination of the inner container lid weld.

The middle lid, made of Alloy 22, will be welded to the outer barrier using a fillet weld. The gas-tungsten arc welding method is presently being considered for remote welding of this lid (Bechtel SAIC Company, LLC, 2001c). There will be a remote visual inspection of the weld preparation surfaces followed by a dimensional inspection using a tactile coordinate measuring system, tack welding, and then circumferential welding of the lid. Nondestructive evaluation of the weld will be done by visual inspection and eddy current testing.

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

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

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

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

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

In summary, DOE agreed (Reamer, 2001b) to provide justification that the nondestructive evaluation methods used to inspect the Alloy 22 and Type 316 nuclear grade stainless steel plate materials and welds are sufficient and capable of detecting defects that adversely may affect waste package preclosure structural performance. Subsequent to the technical exchange agreement, DOE demonstrated, through an assessment of the ultrasonic inspection of the closure weld mockup, that flaws, such as lack of penetration and lack of fusion, can be detected (Bechtel SAIC Company LLC, 2003b, 2001c). The effects of recent waste package design changes on the use of ultrasonic testing to detect flaws in the Alloy 22 closure welds should be evaluated.

4.1.7.2.3.7 Criticality Design Criteria

10 CFR 63.112(e)(6) requires that the preclosure safety analysis include an analysis of the performance of the structures, systems, and components that provide means to prevent and control criticality. 10 CFR 63.112(f) requires a description and discussion of the design and the relationship between the design bases, the design criteria, and the preclosure performance objectives. In its review of the preliminary preclosure safety assessment (DOE, 2001), the staff identified two general issues. The first is the DOE reliance on the level of the burnup in the commercial spent nuclear fuel assemblies for designing the criticality control systems of the waste packages. The second includes consideration of events (e.g., internal and external flooding; spent nuclear fuel assembly misload events; events in the pools and storage racks; and, in general, Category 1 and 2 events with respect to criticality) when designing the surface and subsurface facilities.

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

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

With respect to the consideration of events such as flooding, misload, and the like, DOE states that established design requirements "… preclude preclosure criticality unless two unlikely independent events occur [e.g., CRWMS M&O (2000l)]. The probability of two unlikely independent events occurring will be less than 10^{-6}/yr." While the double-contingency principle (i.e., two unlikely events), which has been used historically in designing criticality control systems for facilities, storage, and transportation packages, may not require the licensee to quantify the probability of the unlikely events, under 10 CFR Part 63 events must be identified, their probabilities quantified, and designations assigned as Category 1 or 2 events. 10 CFR 63.112(e)(6) also requires an analysis of the performance of structures, systems, and components to control and prevent nuclear criticality. Therefore, DOE has indicated that the repository preclosure structures, systems, and components will be designed to prevent criticality under normal operation and Categories 1 and 2 events (Reamer, 2001b).

4.1.7.2.3.8 Waste Package Shielding

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

The current DOE waste package design description appears to provide information on the shielding to prevent radiolysis-induced corrosion that is sufficient to use in developing a potential license application. Protection for workers is provided by other structures, systems, and components. The design of the waste package is still being developed. DOE should provide additional information as the design of the waste package is further developed.

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

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

The waste package drop event has been characterized as an internal event sequence that is not expected to result in a radiological release because it is prevented by the design of the waste package (CRWMS M&O, 2000f). Analyses intended to support this characterization have been performed (CRWMS M&O, 2000m). These analyses are limited to a single waste package drop orientation. It is not clear that a single drop orientation scenario is sufficient to bound the potential for waste package failure, considering the number of different waste package handling operations and the present lack of design detail for the various cranes and other devices that will be used to transfer the waste package from the Waste Handling Building to its emplacement within the drift. DOE stated during the preclosure technical exchange

(Reamer, 2001b) that, as part of the normal design process, design basis dynamic events will be reevaluated as the designs for both the surface and subsurface facilities mature.

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

DOE agreed (Reamer, 2001b) to (i) demonstrate the mesh discretizations of the finite-element models used to simulate the effects of waste package drop events are sufficient to provide reasonably convergent results that can be used to assess potential failure, (ii) justify the constitutive models used to represent the response of the waste package materials to impact loads (e.g., the inclusion or exclusion of temperature and strain rate effects), (iii) provide documentation of all boundary conditions used for the numerical models and the technical basis or rationale for them, and (iv) provide evidence the criterion used to establish failure adequately bounds the uncertainties associated with effects not explicitly considered in the simulation. Specific uncertainties not presently considered in the waste package drop analyses are (i) residual stresses arising from the closure weld fabrication process, (ii) dimensional and material variabilities, (iii) ground motion effects caused by a seismic event (waste package drops and tipovers are more likely to occur during seismic events), and (iv) sliding and inertial effects of the spent nuclear fuel.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

4.1.7.2.3.10 Fuel Cladding Thermal Control

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

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

4.1.7.3 Summary and Status

4.1.7.3.1 Surface Facilities

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

4.1.7.3.2 Subsurface Facilities

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

4.1.7.3.3 Waste Package and Other Engineered Barriers

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

- The final waste package design bases and their relationships to the design criteria
- The final waste package design and specifications
- Design criteria for residual stress mitigation of the waste package fabrication welds by solution annealing
- Design criteria for residual stress mitigation of the waste package closure weld using either laser peening or low-plasticity burnishing
- The effects of microstructural and compositional variations of the plate and filler materials on the thermal stability and mechanical properties of the Alloy 22 waste package outer container

4.1.7.3.4 Status of Key Technical Issue Agreements

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

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

Preclosure Item	Status	Related Agreement*	Note
Relationship between the Design Criteria and Design Basis and the Regulatory Requirements	Staff review incomplete	†	—
Geologic Repository Operations Area Design Methodologies	Staff review incomplete	†	—
Assumptions, Codes, and Standards for Surface Facilities Design	Staff review incomplete	†	—
Materials for Surface Facilities Design	Staff review incomplete	†	—
Load Combinations for Surface Facilities Design	Staff review incomplete	†	—
Surface Facilities Design Analyses and Documentation	Staff review incomplete	†	—
Assumptions, Codes, and Standards for Subsurface Facilities Design	Staff review incomplete	†	—
Subsurface Operating Systems Design	Staff review incomplete	†	—
Material and Material Properties for Subsurface Facilities Design	Complete	RDTME.3.01‡	Section 4.1.7.2.2.2
Load Combinations for Subsurface Facilities Design	Staff review incomplete	RDTME.2.01‡ RDTME.2.02 RDTME.3.02 RDTME.3.03	Seismic load characterization and critical combination of thermal and seismic loadings
Models and Rock Properties for Subsurface Facilities Design	Staff review incomplete	RDTME.3.04 RDTME.3.05 RDTME.3.07 RDTME.3.08 RDTME.3.10 RDTME.3.13	Rock properties and data sufficiency, rock strength, and fracture pattern analyses

Table 4.1.7-2. Summary of Resolution Status for Design of Structures, Systems, and Components Important to Safety and Safety Controls Preclosure Topic (continued)			
Preclosure Item	**Status**	**Related Agreement***	**Note**
Subsurface Ground-Support Systems Design	Staff review incomplete	RDTME.3.06 RDTME.3.09	Drift invert stability and rock support system analyses
Subsurface Ventilation System Design	Staff review incomplete	RDTME.3.14	Ventilation modeling and validation
Subsurface Power and Power Distribution Systems Design	Staff review incomplete	†	—
Maintenance Plan for Subsurface Facilities	Staff review incomplete	†	—
Waste Package and Engineered Barrier System Design	Staff review incomplete	PRE.07.01 through PRE.07.05	Criticality analysis, finite element modeling, weld filler material compatibility, nondestructive evaluation methods, and mechanical properties after welding

*Related DOE and NRC agreements are associated with one or more review methods.
†Not discussed at the first DOE and NRC Technical Exchange and Management Meeting on Preclosure Safety [Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. ML021340719. Washington, DC: NRC. 2001. <www.nrc.gov/waste/hlw-disposal/public-involvement/mtg-archive.html#KTI>]
‡No further concerns at this time.

4.1.7.4 References

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

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

4.2 Plans for Retrieval and Alternate Storage of Radioactive Wastes

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

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

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