2.2 Plans for Retrieval and Alternate Storage of Radioactive Wastes

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

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

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

2.4 Status of Preclosure Issue Resolution and Path Forward

Based on 10 CFR Part 63 and its review of the DOE preliminary preclosure safety assessment report (CRWMS M&O, 2001), the repository safety strategy (CRWMS M&O, 2000), and other support documents, NRC staff preliminarily identified 10 preclosure topics that DOE should address in any future license application regarding the potential high-level waste repository at Yucca Mountain.

(1) Site Description As It Pertains to Preclosure Safety Analysis

(2) Description of Structures, Systems, Components, Equipment, and Operational Process Activities

(3) Identification of Hazards and Initiating Events

(4) Identification of Event Sequences

(5) Consequence Analyses

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

(7) Design of Structures, Systems, and Components Important to Safety and Safety Controls

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

(9) Plans for Retrieval and Alternate Storage of Radioactive Wastes

(10) Plans for Permanent Closure and Decontamination, or Decontamination and Dismantlement of Surface Facilities

Resolution of concerns related to these preclosure topics (8), (9), and (10) has not been initiated. Therefore, no progress toward these three areas is documented in this issue resolution status report. Identification and resolution of concerns in the remaining subject areas are at various stages of progress.

2.4.1 Progress on Preclosure Topics

Identification of technical concerns associated with preclosure topics (1) through (7) is at various stages of development. Subtopics for the various technical areas identified for these seven preclosure topics, as of the cutoff date for this issue resolution report, are discussed in this subsection (Table 2.4-1). The list is not complete at this time, and technical concerns will continue to be identified and clarified as the review of DOE documents proceeds. It should also be noted that not all the preclosure technical concerns identified were addressed in the

July 2001 Technical Exchange Meeting on Preclosure Safety.[1] Additional information about the status of seismic design and thermal-mechanical effects on underground facility design related to preclosure topic (7) is discussed in Section 2.4.2.

Detailed discussions and agreements reached regarding the technical concerns are provided in appropriate sections of this issue resolution report. Table 2.4-1 provides the status of preclosure technical concerns. The table also enumerates the related DOE and NRC agreements pertaining to the preclosure technical areas. Note that the status of all key technical issues are provided in Table 1.1-3. In addition, all agreements pertaining to the key technical issues and preclosure subtopics are provided in Appendix A.

2.4.2 Progress on Preclosure Concerns Addressed in the Repository Design and Thermal-Mechanical Effects Key Technical Issue

In the Repository Design and Thermal-Mechanical Effects Key Technical Issue, three subissues are relevant to preclosure topic (7): Subissue 1, Implementation of an Effective Design Control Process Within the Overall Quality Assurance Program; Subissue 2, Design of the Geological Repository Operations Area for the Effects of Seismic Events and Direct Fault Disruption; and Subissue 3, Thermal-Mechanical Effects on Underground Facility Design and Performance.

Table 2.4-1. Related Technical Concerns and Agreements

Preclosure Topics and Key Technical Issue	Concerns or Subissues	Status	Related Agreements
Site Description As It Pertains to Preclosure Safety Analysis	Geotechnical Investigation for Surface Facility	Not Addressed	None
	Design Basis Ash Fall	Not Addressed	None
Description of Structures, Systems, Components, Equipment, and Operational Process Activities	High-Level Waste Characterization	Not Addressed	None
Identification of Hazards and Initiating Events	Aircraft Hazards	Addressed	PRE.03.01
	Tornado Missile Hazards	Addressed	PRE.03.02
	Nearby Military Facilities Hazards	Not Addressed	None

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

Table 2.4-1. Related Technical Concerns and Agreements (continued)			
Preclosure Topics and Key Technical Issue	**Concerns or Subissues**	**Status**	**Related Agreements**
Identification of Hazards and Initiating Events	Operational Hazards Including Human Reliability	Not Addressed	None
	Earthquake as an Initiating Event	Addressed	RDTME.2.01 RDTME.2.02
	Fire Hazards	Not Addressed	None
Identification of Event Sequences	Events Screened Out by Design	Addressed	Agreement Summary*
	Justification of Probability Estimates	Addressed	Agreement Summary*
Consequence Analyses	Dose Calculation Methodology for Category 1 Event Sequences	Addressed	None†
	Dose Calculation Methodology for Category 2 Event Sequences	Not Addressed‡	None
Identification of Structures, Systems, and Components Important to Safety; Safety Controls; and Measures to Ensure Availability of the Safety Systems	Q-List Methodology	Addressed	PRE.06.01 PRE.06.02
	Quality Level Categorization	Addressed	PRE.06.01 PRE.06.02
Design of Structures, Systems, and Components Important to Safety and Safety Controls	Level of Design Details	Addressed	None§
	Engineered Barrier Subsystem and Fabrication	Addressed	PRE.07.02 through PRE.07.05
	Burnup Credit and Criticality	Addressed	PRE.07.01
	Soil-Structure Interaction	Not Addressed	None
	Ventilation Design	Not Addressed	None
	Fire Protection Design	Not Addressed	None

Table 2.4-1. Related Technical Concerns and Agreements (continued)			
Preclosure Topics and Key Technical Issue	**Concerns or Subissues**	**Status**	**Related Agreements**
Meeting the 10 CFR Part 20 As Low As Is Reasonably Achievable Requirements for Normal Operations and Category 1 Event Sequences	Not Yet Identified	Review Not Initiated	None
Plans for Retrieval and Alternate Storage of Radioactive Wastes	Not Yet Identified	Review Not Initiated	None
Plans for Permanent Closure and Decontamination, or Decontamination and Dismantlement of Surface Facilities	Not Yet Identified	Review Not Initiated	None
Repository Design and Thermal-Mechanical Effects	Subissue 1—Implementation of an Effective Design Control Process Within the Overall Quality Assurance Program	Closed	None
	Subissue 2—Design of the Geological Repository Operations Area for the Effects of Seismic Events and Direct Fault Disruption	Closed-Pending	RDTME.2.01 RDTME.2.02
	Subissue 3—Thermal-Mechanical Effects on Underground Facility Design and Performance	Closed-Pending	RDTME.3.01 through RDTME.3.14

* Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2001.
† Common understanding with DOE was reached.
‡ No significant uncertainties because well-established methods are available.
§ A draft position paper was provided to DOE.

Historically, DOE implementation of a design control process for design, construction, and operation of the geologic repository operations area has been one of the NRC major concerns. The staff conducted a series of interactions and reviews and an in-field verification to evaluate the effectiveness of the DOE design control process. Through these interactions, deficiencies covering a wide spectrum of the design control process, including data traceability, management, qualification, and software control, were identified [for a detailed discussion, refer to NRC (2000)]. In responding to the NRC concerns, DOE developed and implemented new administrative procedures to replace the existing quality assurance procedures. The new administrative procedures extend to the contractors. The staff believe these new administrative procedures simplify the document hierarchy that controls the design and analysis activities. As a result, transparency and traceability of the flowdown from the regulatory requirements to design bases and criteria are improved. The staff consider this simplified design control process acceptable, and Key Technical Issue Subissue 1, Implementation of an Effective Design Control Process Within the Overall Quality Assurance Program, is closed with respect to issue resolution. The implementation of the design control process, however, will continue to be monitored through observation of DOE audits or NRC independent audits and inspections of DOE activities.

DOE proposed three topical reports to address Key Technical Issue Subissue 2, Design of the Geological Repository Operations Area for the Effects of Seismic Events and Direct Fault Disruption. NRC staff reviewed and accepted the first and second topical reports (DOE, 1994, 1996). NRC will review the third topical report, Design of the Geological Repository Operations Area for the Effects of Seismic Events and Direct Fault Disruption, once it is submitted.

Key Technical Issue Subissue 3, Thermal-Mechanical Effects on Underground Facility Design and Performance, was discussed during the technical exchange meeting with DOE about the Repository Design and Thermal-Mechanical Effects Key Technical Issue.[2] Agreements on various aspects of the subissues were reached during the meeting. Consequently, Subissues 2 and 3 are currently closed-pending. Detailed discussions about concerns are provided in Section 2.1.7 of this issue resolution report. Table 2.4-1 provides the status of Subissues 2 and 3 and related DOE and NRC agreements pertaining to the Repository Design and Thermal-Mechanical Effects Key Technical Issue. The status and detailed agreements pertaining to all key technical issues are provided in Table 1.1-3 and Appendix A.

2.4.3 Path Forward

The path forward for addressing the preclosure-related concerns includes four parts:

(1) Conducting Appendix 7 meetings with DOE to monitor the progress of addressing the agreements reached during the previous technical exchange meetings

(2) Continuing the review of DOE preclosure-related documents when they become available and the identification of technical concerns, if any

(3) Conducting a technical exchange meeting to discuss the remaining preclosure concerns listed in Section 2.4.1 and new concerns identified so far through reviewing DOE preclosure-related documents

(4) Conducting limited independent preclosure safety analyses to identify vulnerabilities in the DOE design and related safety case

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

2.4.4 References

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

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

DOE. "Methodology to Assess Fault Displacement and Vibratory Groundmotion Hazards at Yucca Mountain." YMP/TR–002–NP. Revision 0. Washington, DC: DOE. 1994.

———. "Seismic Design Methodology for a Geologic Repository at Yucca Mountain." YMP/TR–003–NP. Revision 1. Washington, DC: DOE. 1996.

NRC. "Issue Resolution Status Report—Key Technical Issues: Repository Design and Thermal-Mechanical Effects." Revision 3. Washington, DC: NRC, Division of Waste Management. 2000.

3 REPOSITORY SAFETY AFTER PERMANENT CLOSURE

3.1 System Description and Demonstration of Multiple Barriers

3.1.1 Description of Issue

Postclosure performance objectives specified in 10 CFR Part 63 require a system of multiple barriers (at least one engineered and one natural). As defined in the regulations, barriers are materials or structures that prevent or substantially delay movement of water or radionuclides. Thus, a key element of the safety case is the identification and description of the capabilities of the repository barriers. Examples of natural barriers at Yucca Mountain include the unsaturated and saturated volcanic and alluvial rock units that control movement of radionuclides by processes such as infiltration, matrix diffusion, and sorption. Engineered barriers DOE has considered in design options include a titanium drip shield, a double-walled container for waste packages, fuel cladding, and invert materials. Each barrier provides additional assurance that the postclosure performance objectives can be met. The description of each barrier capability provides an overall understanding of the DOE safety case and how the diversity of the barriers enhances the resiliency of the repository system.

As provided in 10 CFR Part 63, DOE is required to identify the barriers in the safety case, describe the capabilities of each of the barriers, and provide the technical basis for the capability of the barriers (the technical basis is to be consistent with the technical basis used to support the total system performance assessment). In general, staff will review the potential Total System Performance Assessment–License Application to ensure that DOE identifies all barriers in its safety case; describes the capability of the barriers consistent with the parameter, models, and assumptions in the total system performance assessment; and provides a technical basis consistent with that used for the total system performance assessment.

The following summaries are excerpted from 10 CFR Part 63.

10 CFR 63.113—Performance objectives for the geologic repository after permanent closure.

- The geologic repository must include multiple barriers, consisting of both natural barriers and an engineered barrier subsystem.

- The engineered barrier subsystem must be designed so that, working in combination with natural barriers, radiological exposures to the reasonably maximally exposed individual are within the limits specified in 10 CFR 63.311 of Subpart L. Compliance with this paragraph must be demonstrated through a total system performance assessment (that meets the requirements specified in 10 CFR 63.114 of this subpart, and 10 CFR 63.303, 63.305, 63.312, and 63.342 of Subpart L).

- The engineered barrier subsystem must be designed so that, working in combination with natural barriers, radionuclides released into the accessible environment are within the limits specified in 10 CFR 63.331 of Subpart L. Compliance with this paragraph must be demonstrated through a total system performance assessment (that meets the requirements specified in 10 CFR 63.114 of this subpart and 10 CFR 63.303, 63.332, and 63.342 of Subpart L).

10 CFR 63.115—Requirements for multiple barriers. Demonstration of compliance with 10 CFR 63.113 must

- Identify those design features of the engineered barrier subsystem, and natural features of the geologic setting, considered barriers important to waste isolation.

- Describe the capability of barriers identified as important to waste isolation to isolate waste, taking into account uncertainties in characterizing and modeling the behavior of the barriers.

- Provide technical basis for description of the capability of barriers identified as important to waste isolation to isolate waste. The technical basis for each barrier's capability shall be based on and consistent with the technical basis for the total system performance assessments used to demonstrate compliance with 10 CFR 63.113(b) and (c).

Consistent with 10 CFR Part 63, the Multiple Barriers Subissue in NRC (2002) focuses on the demonstration of multiple barriers and includes (i) identification of design features of the engineered barrier subsystem and natural features of the geologic setting considered barriers important to waste isolation, (ii) descriptions of the capability of barriers to isolate waste, and (iii) technical basis for each barrier capability. In addition, the review plan (NRC, 2002) addresses the staff expectation of the contents of the DOE total system performance assessment and supporting documents. Specifically, it focuses on those aspects of the total system performance assessment that will allow for an independent review of the results.

NRC staff will review the potential Total System Performance Assessment–License Application to ensure that multiple barrier considerations satisfy the requirements of 10 CFR 63.113(a). Staff will ensure that an engineered barrier subsystem has been designed that, working in combination with natural barriers, satisfies the requirement for a system of multiple barriers and complies with postclosure performance standards.

NRC staff will review the potential Total System Performance Assessment–License Application to ensure that multiple barrier considerations satisfy the requirements at 10 CFR 63.115(a)–(c). Staff will ensure that those design features of the engineered barrier subsystem and natural features of the geologic setting considered barriers important to waste isolation have been identified. A description has been provided of the capabilities of barriers identified as important to waste isolation, taking into account uncertainties in characterizing and modeling the barriers. The technical basis provided for this description is based on and consistent with the technical basis for the total system performance assessment.

This section provides a review of the multiple barrier analysis presented in the DOE total system performance assessment, a discussion of the NRC review, and agreements reached with the DOE. NRC review was limited to the methodology portion of multiple barriers. Compliance with the standards in 10 CFR Part 63 for individual and groundwater protection and human intrusion is not considered in prelicensing issue resolution. The comments describe the staff expectation of the contents of the DOE total system performance assessment, and the supporting documents define those aspects that will allow an independent review of the total system performance assessment results.

3.1.2 Relationship to Key Technical Issue Subissues

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

3.1.3 Importance to Postclosure Performance

If the repository system is made up of multiple barriers, it will be more tolerant of unanticipated failures and external challenges. Understanding the capability of the system component barriers provides an understanding of the repository system, which can increase confidence that the postclosure performance objectives will be met.

The description of barrier capability provides information that helps interpret the total system performance assessment results and provides information independent from the condition of the other barriers, so that insights can be gained into total system performance assessment results. Such information illustrates the resilience or lack of resilience of the repository to unanticipated failures or external challenges.

The evaluation of a first-of-a-kind repository for an extended time period (i.e., 10,000 years) results in uncertainty in characterizing the natural system being included in the total system performance assessment. Besides, those materials used in the engineered barrier subsystem that are relatively new (i.e., without a long history of use), have uncertainty in their life prediction. Consideration of multiple barriers as a part of total system performance assessment compensates for such residual uncertainties in estimating performance and increases confidence that postclosure performance objectives will be met.

The description of each barrier capability provides the reviewer flexibility to consider the nature and extent of conservatism in the evaluations used for compliance demonstration and to decide whether there is a need to require DOE to reduce uncertainties in the assessment (e.g., collecting more site data) or to include further mitigative measures.

3.1.4 Technical Basis

NRC has developed a review plan (NRC, 2002) consistent with acceptance criteria and review methods found in previous issue resolution status reports. This section briefly describes the DOE approach and the NRC staff review of that approach. Finally, this section presents agreements DOE and NRC reached to address the staff concerns.

Overall, the current information, along with agreements reached between DOE and NRC, is sufficient to conclude that the necessary information will be available on the identification of barriers, description of barrier capability, and technical basis for barrier capability either before or at the time of a potential license application.

The NRC comments on the DOE multiple barrier analysis and the resulting agreements that led to the closed-pending status for this subissue are based on the information provided in

CRWMS M&O (2000a,b). A presentation titled, Total System Performance Assessment and Integration Key Technical Issue Subissue 1—Multiple Barriers, made at the technical exchange held in Las Vegas, Nevada, during August 6–10, 2001,[1] provided additional understanding of the DOE multiple barriers approach and future plan to support the DOE total system performance assessment. The staff also used their experience from the past independent research, information in open literature, review of previous DOE total system performance assessments, information learned during meetings with DOE, the approach used in the NRC TPA Version 4.0 code (Mohanty, et al., 2002), acceptance criteria and review methods in NRC (2002), and technical bases for these acceptance criteria contained in the Revision 3 Issue Resolution Status Reports of other key technical issues. In addition, insight gained from sensitivity studies using the NRC TPA Version 3.2 code (Mohanty, et al., 1999) has been incorporated to the extent feasible.

The DOE Approach

DOE documented its approach to identifying natural and engineered barriers in CRWMS M&O (2000a,b). DOE identified four natural barriers and five engineered barriers. Natural barriers consisted of (i) surficial soils and topography, (ii) unsaturated zone rocks above the repository, (iii) unsaturated zone rocks below the repository horizon, and (iv) tuff and alluvial aquifers. Engineered barriers consisted of (i) the titanium drip shield, (ii) the C–22 waste canister, (iii) commercial spent nuclear fuel cladding, (iv) the waste form (e.g., high-level waste glass), and (v) a drift invert (e.g., crushed tuff).

In CRWMS M&O (2000a,b) and the DOE presentation,[2] DOE stated that barrier importance analysis is used in conjunction with sensitivity analysis to demonstrate barrier capability. Barrier importance analysis encompasses[3] (i) evaluation of significance of parameter and model uncertainty, (ii) evaluation of robustness of system performance using low probability scenarios within the framework of the total system performance assessment, and (iii) quantification of the capability of the barrier to isolate waste. Two types of analyses have been performed: degraded barrier importance analysis and neutralized barrier importance analysis. The degraded barrier importance analysis has been performed by fixing several parameters associated with a barrier at the 95th percentile (or 5th percentile if that leads to maximizing the dose rate) values in the total system performance assessment model and rerunning the probabilistic analyses. For the neutralized barrier importance analysis, the function of a barrier is eliminated by setting selected parameters in a way that correspond to omission

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[1]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[2]DOE. "Total System Performance Assessment and Integration." Presentation to DOE/NRC Technical Eaxchange on the *Total System Performance Assessment and Integration Key Technical Issue, August 6–9, 2001, Las Vegas, Nevada.* Las Vegas, Nevada: DOE, Yucca Mountain Site Characterization Office. 2001.

[3]Andrews, R.W. "Sensitivity and Barrier Importance Analyses for TSPA–SR." *Presentation to DOE/NRC Technical Exchange on Total System Performance Assessment (TSPA) for Yucca Mountain, June 6–7, 2000, San Antonio, Texas.* Washington, DC: DOE, Office of Civilian Radioactive Waste Management. 2000.

(i.e., neutralization) of a process model factor or equivalently (in most cases), a barrier. DOE points out that the neutralization of a barrier (compared to the degradation of a barrier, which is within the total system performance assessment parameter range) permits gaining insights into total system performance assessment and provides insights into barrier redundancy. In the degraded barrier importance analysis, DOE assumes that various natural and engineered barriers are degraded either individually or in combination. DOE recognizes that because the degraded barrier importance analysis necessarily stays within the basecase uncertainty ranges of individual analyses, it cannot elevate in importance any barrier having a restricted range of uncertainty.

DOE examined the relative contribution of each barrier by comparing the nominal performance results (i.e., dose curves) with the degraded performance results for radionuclides within and beyond the compliance period. The contribution of individual barriers has been compared to the overall performance objective.

The NRC review of the two DOE documents describing the demonstration of multiple barriers, in CRWMS M&O (2000a,b), resulted in several concerns, primarily in the areas of description of barrier capability and technical basis for barrier capability. The staff believe that barrier capability needs to be described consistent with the definitions in 10 CFR Part 63 (i.e., prevents or substantially reduces movement of water or radionuclides). The concerns that led to reaching an agreement with DOE are listed next. The concerns that did not require agreements because the DOE clarifications addressed the issue can be found in the handouts provided at the DOE and NRC Technical Exchange on Total Systerm Performance Assessment.[4]

- DOE states the capabilities of barriers include (i) limiting contact of water on waste packages by reducing infiltration, (ii) prolonging waste package lifetimes, (iii) limiting radionuclide mobility and release, and (iv) slowing transport away from the repository. The NRC staff found that DOE presented the capability of barriers primarily in terms of dose. For example, CRWMS M&O (2000a, pp. 2–5) describes barrier capability, but no diagrams are presented to support the discussion. Although CRWMS M&O (2000a) asserts the barriers limit water and radionuclide movements, the results from barrier neutralization importance analyses and degraded barrier importance analyses (see figures in Chapter 3 of CRWMS M&O, 2000a) are based only on dose, and not on barrier capability, to prevent or delay movement of water or radionuclides. To understand the barrier capability, the NRC staff should be able to understand how the total system performance assessment results can be explained through barrier capability (e.g., retardation of radionuclides in the saturated zone, waste package lifetime, and matrix diffusion in the unsaturated zone). Understanding the way natural

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[4]DOE. "Total System Performance Assessment and Integration." Presentation to DOE/NRC Technical Eaxchange on the *Total System Performance Assessment and Integration Key Technical Issue, August 6–9, 2001, Las Vegas, Nevada.* Las Vegas, Nevada: DOE, Yucca Mountain Site Characterization Office. 2001.

and engineered barriers isolate waste or delay radionuclide release will increase confidence in the total system performance assessment objectives specified at 10 CFR 63.11(b).

- The methods used to differentiate the contributions of barriers that perform similar functions need to be explained. Barriers that perform similar functions could include components of natural and engineered systems (e.g., the combination of the natural system above the repository and the drip shield) along important boundaries. The discussion of barrier capabilities needs to differentiate between the independent and the interdependent contributions of the individual barriers.

- The uncertainty associated with particular barriers has not been described. The description needs to include model uncertainty (such as the performance of the barrier, assuming alternative conceptual models) and uncertainty in the attributes of the barrier (e.g., parameter uncertainty). The performance needs to be discussed in light of barrier capability to prevent or delay movement of water or radionuclides and, consequently, to limit the expected annual dose.

- The DOE analyses do not describe the interdependence of barriers and also the treatment of combinations of barriers appears to be inconsistent. For example, the combination of barriers treated in CRWMS M&O (2000a) for the degraded barrier importance analyses is different from that used in the barrier neutralization importance analyses. Similarly, the combination of barriers presented in CRWMS M&O (2000b) is different from the combinations presented in CRWMS M&O (2000a) for degraded barrier importance analyses and barrier neutralization analyses. It is difficult to understand the results from the degraded barrier importance analyses and the barrier neutralization importance analyses for identifying barrier importance, without a discussion of the independent and interdependent contributions of the barriers.

  Example 1: The presence of the drip shield in the degraded waste package analyses (CRWMS M&O, 2000a) could mask the effect of the waste package on radionuclide transport during the early period or at least until the drip shield fails. Although such analyses (i.e., in the presence of the drip shield) shows the protection afforded by the drip shield even after the waste package fails, the actual protection provided by each individual barrier in 10,000 years is not clearly identified.

  Example 2: It is not clear why performance improved for the degraded radionuclide concentration limits case, which represents nonmechanistic juvenile failure scenario-sensitivity to radionuclide concentration limits, between 2,000 and 8,000 years (CRWMS M&O, 2000a, Figure 3-20, p. 3-18).

- The description of the capability for individual barriers to prevent or substantially delay movement of water or radionuclide materials needs to include a discussion of the changes in barrier capability during time (throughout the 10,000-year compliance period). The discussion should include the extent to which the conceptual models of the barriers consider cumulative degradation processes during time, processes that may significantly affect the performance of the barrier, and temporal changes within the repository system. As examples, time-dependent environmental or physical-chemical variabilities of the system (e.g., pressure, temperature, or spatial changes before, during, and after the thermal pulse); dynamic conditions (e.g., boiling zone/refluxation; calcite-opal mobilization and precipitation in fractures, lithophysae, and matrix pores; and drift collapse induced by thermal-mechanical stresses) may need to be discussed to appropriately describe the performance of particular barriers.

- The description of barrier capabilities needs to include a discussion of the effects of spatial variability on the ability of the barrier to prevent or substantially delay movement of water or radionuclide materials, including a discussion of the spatial resolution in the models and data used to evaluate the performance of the barriers. For example, assume 50 percent of the Calico Hills nonwelded vitric unit is strongly sorbing and 50 percent is not. As another example, in the what-if analysis of the nonmechanistic juvenile failure scenario in CRWMS M&O (2000a, pp. 3–15), one waste package was artificially set to fail after 100 years. The consequences associated with the failed waste package are influenced by the location of the failed waste package (e.g., the characteristics of radionuclide release, water flow, and radionuclide transport in the vicinity of the failed waste package, where these characteristics may be affected by spatial heterogeneity and its representation in the model used in the analysis).

NRC presented the previously mentioned concerns to DOE, and general agreements were reached at the DOE and NRC Total System Performance Assessment and Integration Issue Resolution Meeting, August 6–10, 2001.[5] DOE agreed to provide (i) enhanced descriptive treatment for presenting barrier capabilities in its final approach for demonstrating multiple barriers and (ii) a discussion of the capabilities of individual barriers, in light of existing parameter uncertainty (e.g., in barrier and system characteristics) and model uncertainty.

DOE also agreed to provide a discussion of the following when documenting barrier capabilities and the corresponding technical bases: (i) parameter uncertainty, (ii) model uncertainty (i.e., the effect of viable alternative conceptual models), (iii) spatial and temporal variabilities in the performance of the barriers, (iv) independent and interdependent capabilities of the barriers (e.g., including a differentiation of the capabilities of barriers performing similar functions), and (v) barrier effectiveness with regard to individual radionuclides. DOE will analyze and document barrier capabilities, in light of existing data and analyses of the performance of the repository system.

3.1.5 Status and Path Forward

The status of the System Description and Demonstration of Multiple Barriers Subissue of the Total System Performance Assessment and Integration Key Technical Issue is provided in Table 3.1-1. This subissue is considered closed-pending by the NRC staff as documented following the DOE and NRC Technical Exchange on Total System Performance Assessment

[5]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

and Integration.[6] The proposed DOE approach, together with the DOE agreements to provide NRC with additional information, acceptably addresses the NRC questions so that no information beyond that already provided, or agreed to be provided, will likely be required at the time of a potential license application.

It should be noted that the NRC review to date has been limited to the methodology portion of multiple barriers, and NRC is not addressing whether DOE has adequately identified multiple barriers or if DOE has demonstrated multiple barriers are present. The status and the detailed agreements (path forward) pertaining to all key technical issue subissues are provided in Table 1.1-3 and Appendix A.

Table 3.1-1. Status of Resolution of the System Description and Demonstration of Multiple Barriers Subissue

Key Technical Issue	Subissue	Status	Related Agreements*
Container Life and Source Term	Subissue 3—The Rate at Which Radionuclides in Spent Nuclear Fuels Are Released from the Engineered Barrier Subsystem through the Oxidation and Dissolution of Spent Nuclear Fuel	Closed-Pending	CLST.3.01
	Subissue 4—The Rate at Which the Radionuclides in High-level Waste Glass Are Leached and Released from the Engineered Barrier Subsystem	Closed-Pending	CLST.4.01
Total System Performance Assessment and Integration	Subissue 1—System Description and Demonstration of Multiple Barriers	Closed-Pending	TSPAI.1.01 TSPAI.1.02

*Related DOE and NRC agreements are associated with one or all acceptance criteria.

3.1.6 References

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

[6]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

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

3.2 Scenario Analysis and Event Probability

3.2.1 Scenario Analysis

3.2.1.1 Description of Issue

A complete safety evaluation of a geologic repository for high-level waste requires consideration of potential future conditions affecting its behavior during the period of regulatory concern. This safety evaluation may be accomplished through scenario analysis, which is the systematic enumeration of features, events, and processes that can reasonably occur in the repository system. Scenario analysis facilitates identifying the possible ways in which the geologic repository environment can evolve so a defensible representation of the system can be included in the total system performance assessment.

A scenario is defined as the plausible future evolution of the repository system during the period of regulatory concern. A scenario includes a postulated sequence (or absence) of events and assumptions about initial and boundary conditions. A scenario analysis is composed of four steps: (i) identification of features, events, and processes relevant to the proposed high-level waste geologic repository; (ii) selection or screening of features, events, and processes important to estimating dose risk to a reasonably maximally exposed individual during the period of regulatory concern; (iii) formation of scenario classes from a screened or reduced collection of features, events, and processes; and (iv) selection or screening of the scenario classes for actual implementation into a total system performance assessment.

This section provides a review of the scenario analysis methodology implemented by DOE. Technical bases for scenario analysis are documented in analysis and model reports, CRWMS M&O (2000a), and other technical reports. The scenario analysis review is documented in two parts, one referring to the identification of features, events, and processes that affect compliance with the overall performance objective and the other referring to the identification of events with probabilities greater than 10^{-8} per year.

3.2.1.2 Relationship to Key Technical Issue Subissues

The identification of features, events, and processes important to repository safety is pertinent to all the key technical issue subissues. The subsequent sections incorporate applicable portions of these technical issue subissues, however, no effort was made to explicitly identify each subissue in the text. Features, events, and processes incorporated into the performance assessment are reviewed under the appropriate integrated subissues under model abstraction.

3.2.1.3 Importance to Postclosure Performance

A scenario analysis attempts to identify all features, events, and processes that could influence, directly or indirectly, dose risk from the proposed high-level waste repository to a reasonably maximally exposed individual. A well-implemented process for identification of these features, events, and processes helps to ensure relevant aspects of the proposed high-level waste repository, and associated implications to the dose risk, are studied. Appropriate identification and screening of scenario classes are intended to guarantee that all relevant sequences of events and processes are accounted for in the dose risk assessment. A well-documented compendium of features, events, and processes facilitates identification of the aspects analyzed in the evaluation of the repository safety and serves as a road map to the location of the analyses and their conclusions. Therefore, the goal of scenario analysis is to ensure that no aspect of the proposed high-level waste repository is overlooked in the evaluation of its safety.

3.2.1.4 Technical Basis

NRC developed a plan (2002) consistent with the acceptance criteria and review methods found in previous issue resolution status reports. A review of DOE approaches for development of a scenario analysis to support the total system performance assessment is provided in the following subsections. The review is organized according to the four acceptance criteria: (i) The Identification of an Initial List of Features, Events, and Processes Is Adequate; (ii) Screening of the Initial List of Features, Events, and Processes Is Appropriate; (iii) Formation of Scenario Classes Using the Reduced Set of Events Is Adequate; and (iv) Screening of Scenario Classes Is Appropriate.

3.2.1.4.1 The Identification of an Initial List of Features, Events, and Processes Is Adequate

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.2.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the adequacy of the identification of an initial list of features, events, and processes.

The process used to construct the initial list of features, events, and processes is detailed in CRWMS M&O (2000a, 2001a). DOE compiled a database of features, events, and processes potentially relevant to the proposed high-level waste repository (the Yucca Mountain Project Database of Features, Events, and Processes, hereon referred to as the database). This database is a collection of features, events, and processes from other radioactive waste disposal programs cataloged by the Nuclear Energy Agency of the Organization for Economic Co-operation and Development. This list was supplemented with entries from Yucca Mountain project literature; brainstorming and iterative reviews from experts; and feedback from DOE and NRC technical exchanges, Appendix 7 meetings, and NRC issue resolution status reports (CRWMS M&O 20001a). DOE acknowledges that construction of the list of features, events, and processes is an iterative process subject to refinement (CRWMS M&O, 2000a). DOE stated this list is open and may continue to expand if additional features, events, and processes are identified during the site recommendation process or the development of a potential license application (CRWMS M&O, 2000a).

A total of 1,808 entries, identified as primary, secondary, or classification, has been cataloged in the CRWMS M&O (2001b). Only primary and secondary entries correspond to actual features, events, and processes. Classification entries are intended to enhance the organization of the database. Primary entries have been given broad definitions so they encompass multiple secondary entries. It is expected that, by developing screening arguments for primary features, events, and processes, screening rationales for secondary features, events, and processes would follow. A total of 328 primary features, events, and processes has been identified in the database (CRWMS M&O, 2001a).

DOE argues that the list of features, events, and processes is comprehensive because these (i) have been identified from diverse backgrounds (from several international waste disposal programs) using a variety of methods (expert judgment, informal elicitation, event tree analysis, and stakeholder review) and (ii) have been subjected to iterative discussions and systematic classification (CRWMS M&O, 2000a). Also, DOE stated this list of features, events, and processes is indeed comprehensive (CRWMS M&O, 2001a) because few new elements have been identified in recent iterative reviews.

According to CRWMS M&O (2001a), the database may be updated by DOE through a systematic review of NRC issue resolution status reports, a review of a newer version (Version 1.2) of the Nuclear Energy Agency database, and the resolution of any outstanding NRC near-field environment audit issues identified in Pickett and Leslie (1999) and outstanding issues in NRC (2000).

NRC staff evaluated the list of features, events, and processes reported in several analysis and model reports and in the CRWMS M&O (2001b) and concluded that some aspects of the proposed high-level waste repository are not described in this list. For example, no item is listed in the database addressing response of the drip shield to static loads and seismic excitation. The database should contain elements to account for degradation of the drip shield caused by the interaction of seismic excitation with dead loads (e.g., rockfall or drift collapse), either for the screening argument of an existing feature, event, and process in the database or for a new entry. Entry 1.2.03.02.00 (Seismic Vibration Causes Container Failure)[1] assesses the effect of ground motion on the waste package and drip shield, without consideration of possible preexisting static loads (CRWMS M&O, 2000b, 2001c). Part of the screening argument for 2.1.06.06.00 (Effects and Degradation of Drip Shield) in CRWMS M&O (2001c) is based on an assumption that does not account for the possibility of static loads affecting the drip shield and, possibly, the waste package.

The database does not address the effect of trace metal cations on Alloy 22 and titanium corrosion and stress corrosion cracking, which is a possibility according to results recently reported by Barkatt and Gorman.[2]

At issue is the comprehensiveness of the list of features, events, and processes. For the issues identified in the previous two paragraphs, DOE and NRC have agreements on technical aspects that address outstanding concerns (e.g., Subissue 1 of Container Life and Source

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[1]In this chapter, features, events, and processes listed in the Yucca Mountain Project Database are referred to by the database entry number and title enclosed by parentheses [e.g., 2.1.07.02.00 (Mechanical Degradation or Collapse of Drift)]. The meaning of the database entry number in the form X.X.YY.ZZ.WW is described in CRWMS M&O (2001a).

[2]Barkatt, A. and J.A. Gorman. "Tests to Explore Specific Aspects of the Corrosion Resistance of C–22." *Nuclear Waste Technical Review Board Meeting, August 1, 2000.* Carson City, Nevada. 2000.

Term Key Technical Issue Agreement 14[3] and Subissue 3 of Evolution of the Near-Field Environment Key Technical Issue Agreement 4[4]). DOE agreed to revise descriptions and screening arguments of adequate features, events, and processes to enclose the two items listed previously.[5]

The definition of some primary features, events, and processes is too broad and nondescript to permit easy identification of those aspects included. For example, detailed processes related to the interaction of the ascending dike with the repository drift are not identified as features, events, and processes in the database. Instead, the database includes only general categories such as 1.2.04.04.00 (Magma Interacts with Waste) and 1.2.04.01.00 (Igneous Activity). This high-level definition of features, events, and processes may cause elements relevant to repository and dike interactions and interactions between magma and waste packages and spent nuclear fuel to be overlooked. Features, events, and processes related to magma/repository interactions that do not appear to be explicitly listed in the database include solid and fluid dynamics at the dike tip, vesiculation, plume dynamics, effect of drip shield on magma/repository interactions, geologic factors, threshold flow characteristics, gas segregation, alternate models of vent formation, effects of air shafts and drifts, consideration of flow segregation, localization of magma, recirculation of magma, and evolution of flow conditions. Canister/magma interactions that appear to have been missed include hoop stresses caused by differential expansion of the inner and outer waste packages, melting of materials, thermal shock, and phase changes in Alloy 22 because of the long-term exposure to elevated temperatures. Spent nuclear fuel/magma interactions that may have been missed include cladding response to high temperatures, cladding/fuel chemical reactions causing damage to the waste form (no credit is currently taken for the presence of cladding), mechanical shear, oxidation (during and posteruption), reworking of magma-borne spent nuclear fuel in tunnels and adits, and evolution of flow conditions.

In addition to the difficulty in outlining detailed items addressed by features, events, and processes with broad definitions, the broad definitions produce overlap among database entries, adding complexity to the identification of those aspects addressed by the list of features, events, and processes. Examples of features, events, and processes with broad definitions include (without being exhaustive)

- 1.1.12.01.00 (Accidents and Unplanned Events During Operation)—The entry 1.1.02.01.00 (Site Flooding During Construction and Operation) is explicitly identified in its definition as a particular instance of the former.

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

[4]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9-12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

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

- 1.2.03.01.00 (Seismic Activity)—The entry 1.2.03.02.00 (Seismic Vibration Causes Container Failure) seems a particular instance of the former.

- 2.2.12.00.00 [Undetected Features (in Geosphere)]—This item is too broad for a clear screening argument to be developed. Undetected features relevant to repository performance may be considered in uncertainty and hazard estimates as suggested in the screening argument (CRWMS M&O, 2001c). Multiple features, events, and processes are related to features in the geosphere. For example, features at the repository horizon are also addressed in 1.1.07.00.00 (Repository Design). Thus, the precise scope of this database entry is not clear.

- 2.3.13.01.00 (Biosphere Characteristics)—The broad span of this item causes the scope to be unclear. For example, 2.3.13.02.00 (Biosphere Transport), 2.3.11.01.00 (Precipitation), and 2.4.09.02.00 (Animal Farms and Fisheries) seem to be instances of this entry.

Questions about the scope of several primary features, events, and processes and the differing levels of detail encompassed by them were presented to DOE at the May 15–17[6] and August 6–10,[7] 2001, DOE and NRC Technical Exchanges and Management Meetings on Total System Performance Assessment and Integration. At the May 15–17 meeting, NRC observed that 10 CFR Part 63 requires a systematic analysis of features, events, and processes that might affect the performance of a potential geologic repository at Yucca Mountain. Although it does not specify the manner by which features, events, and processes should be investigated, 10 CFR Part 63 requires that DOE "... provide the technical basis for either inclusion or exclusion of specific features, events, and processes... ." NRC is interested in a transparent, traceable, and technically defensible investigative process leading to a clear understanding of the DOE basis for consideration of features, events, and processes in a total system performance assessment. The varying levels of information used to describe the scope of primary features, events, and processes make it difficult to judge the comprehensiveness of the database.[8] Based on the documentation available, it was not possible for NRC to determine what aspects that might affect the performance of a potential geologic repository at Yucca Mountain were considered by DOE, and where particular features, events, and processes were addressed. Also, it was not evident that the list of features, events, and processes was consistent with transparency and traceability requirements (i.e., it was not evident that the list could be audited).

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

[7]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

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

DOE stated that the list of secondary features, events, and processes is not intended to specify details of primary entries. The definitions of primaries enclose the secondary entries but, in general, have broader scopes. Secondary features, events, and processes are listed in the database to enable traceability and to identify the origin of the primary entry, not to enumerate all aspects addressed by the collection of primary features, events, and processes. DOE stated that the set of primary features, events, and processes should be judged for completeness and comprehensiveness.[9] If DOE adopts aspects of the Nuclear Energy Agency database, then DOE should justify the appropriateness and applicability to the proposed geologic repository at Yucca Mountain. Such information is not available in current DOE documentation.

At the August 6–10, 2001, meeting, DOE stated that it would revise the descriptions of all of the features, events, and processes to (i) better identify all components included in a feature, event, and process; (ii) ensure full incorporation of relevant aspects of a feature, event, and process; (iii) eliminate use of secondary entry terminology, yet retain traceability to the Nuclear Energy Agency database or other source documents; and (iv) make the level-of-detail more consistent, where possible, with a clear differentiation between features, events, and processes and modeling aspects. DOE stated that it would be developing level of detail criteria and refining entries in the database consistent with these criteria. Finally, DOE stated that, besides revising screening arguments for excluded features, events, and processes to improve technical basis descriptions, it will clarify how features, events, and processes screened for inclusion are addressed in the total system performance assessment.[10]

Various agreements addressing the issues highlighted in Section 3.2.1.4.1 were reached at the May 15–17 and August 6–10, 2001, DOE and NRC Technical Exchanges and Management Meetings on Total System Performance Assessment and Integration, and are listed in Section 3.2.1.5.

3.2.1.4.2 Screening of the Initial List of Features, Events, and Processes Is Appropriate

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.2.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the appropriateness of the screening of the initial list of features, events, and processes.

DOE classified the 328 primary features, events, and processes in CRWMS M&O (2001b) into process model subject areas. Eleven analysis and model reports discuss developing screening arguments for features, events, and processes, which are listed in Table 3.2.1-1. Database entries were assigned to more than one analysis and model report because, in general, the

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

[10]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

Table 3.2.1-1. Set of Features, Events, and Processes Analysis and Model Reports for Developing Screening Arguments

Analysis and Model Report Title	Control Identification	Revision/ICN	Year
Features, Events, and Processes in Unsaturated Zone Flow and Transport	ANL–NBS–MD–000001	01/00	2001
Features, Events, and Processes in Saturated Zone Flow and Transport	ANL–NBS–MD–000002	01/00	2000
Evaluation of the Applicability of Biosphere-Related Features, Events, and Processes	ANL–MGR–MD–000011	01/00	2001
Features, Events, and Processes: Screening for Disruptive Events	ANL–WIS–MD–000005	00/01	2000
Features, Events, and Processes: Screening of Processes and Issues in Drip Shield and Waste Package Degradation	ANL–EBS–PA–000002	01/00	2001
Miscellaneous Waste-Form Features, Events, and Processes	ANL–WIS–MD–000009	00/01	2000
Clad Degradation—Features, Events, and Processes Screening Arguments	ANL–WIS–MD–000008	00/01	2000
Colloid-Associated Concentration Limits: Abstraction and Summary	ANL–WIS–MD–000012	00/01	2000
Features, Events, and Processes in Thermal Hydrology and Coupled Processes	ANL–NBS–MD–000004	01/00	2001
Engineered Barrier Subsystem Features, Events, and Processes/Degradation Models Abstraction	ANL–WIS–PA–000002	01/00	2001
Features, Events, and Processes: System Level and Criticality	ANL–WIS–MD–000019	00/00	2000

entries are relevant to more that one process model subject area. Entries addressed by more than one analysis and model report are denoted as shared features, events, and processes. Within an analysis and model report, the terms included and excluded are used to conclude if a feature-event process is relevant or irrelevant (with respect to the dose risk of the proposed high-level waste repository) to a given process-level model. Thus, shared features, events, and processes were given several screening assignments (e.g., included/excluded) by the various analysis and model reports. These screening decisions have not yet been integrated into a single screening decision, but DOE is intending to do so (CRWMS M&O, 2000a).

Each primary database entry was screened as included or excluded on the basis of three criteria developed in the DOE Interim Guidance.[11] These criteria are regulatory, probability, and consequence (CRWMS M&O, 2000a). The Regulatory Criterion refers to the exclusion of primary features, events, and processes from the performance assessment because they are not in accordance with the regulatory guidance[12] or are not applicable by regulation. The Probability Criterion states that features, events, and processes with a probability of occurrence of less than 10^{-4} in 10,000 years can be excluded from consideration in the total system performance assessment. Finally, the Consequence Criterion states that features, events, and processes whose exclusion would not significantly change the expected annual dose may be excluded from the total system performance assessment (CRWMS M&O, 2000a). A summary of the screening decisions (e.g., included/excluded) and the basis (regulatory, probability, or consequence) for the 328 primary features, events, and processes is available in CRWMS M&O (2000a), and the electronic version (in Microsoft® Access) is available in CRWMS M&O (2001b).

DOE plans to update screening arguments and screening decisions in analysis and model reports in accordance with a lower thermal load design [current screening discussions are based on a reference repository design described in CRWMS M&O (2000a)]. Additional effort will focus on integration of screening information and primary descriptions for shared features, events, and processes, and explicit identification of the scenario class (nominal, disruptive, or human intrusion) for each of the elements in the list of features, events, and processes screened as included. Screening arguments will be revised to be entirely consistent with the Interim Guidance[13] (CRWMS M&O, 2001a). As mentioned in Section 3.2.1.4.1, it is also expected that DOE will refine the feature, event, and process descriptions to address NRC concerns per the agreements reached during the May 15–17 and August 6–10, 2001, DOE and NRC Technical Exchanges and Management Meetings on Total System Performance Assessment and Integration.

Staff evaluated screening arguments in analysis and model reports listed in Table 3.2.1-1. Screening arguments in some analysis and model reports depend on assumptions yet to

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[11]Dyer, J.R. "Revised Interim Guidance Pending Issuance of New U.S. Nuclear Regulatory Commission (NRC) Regulations (Revision 01, July 22, 1999), for Yucca Mountain Nevada." Letter (September 3) to D.R. Wilkins, CRWMS M&O. Washington, DC: DOE. 1999.

[12]Ibid.

[13]Ibid.

be verified (CRWMS M&O, 2000c, 2001d,e). Some screening arguments are indicated to be preliminary {e.g., 2.1.07.01.00 [Rockfall (Large Block)]; 1.2.02.01.00 (Fractures); 1.2.02.02.00 (Faulting); 1.2.03.01.00 (Seismic Activity) in CRWMS M&O (2000b); 2.1.14.14.00 (Out-of-Package Criticality, Fuel/Magma Mixture) in CRWMS M&O (2000d); and items listed in Attachment I in CRWMS M&O (2001f)}. It is acknowledged that to-be-verified assumptions are properly tracked by DOE, that work reported in the cited analysis and model reports constitutes work in progress, and that these documents will be revised to disclose more definite screening arguments, as discussed at the May 2001 technical exchange.[14]

A summary of the detailed evaluation of the screening arguments is contained in Table 3.2.1-2, which lists the 328 primary features, events, and processes of CRWMS M&O (2001a), in ascending order of database tracking numbers. In Table 3.2.1-2, features, events, and processes have been classified in accordance with the integrated subissue structure. Elements not pertinent to a given integrated subissue are indicated by a long dash (–). Features, events, and processes not clearly belonging to any of the integrated subissues are listed in the Orphan column. The DOE screening decision is symbolized by I and E (included and excluded), and the initial staff evaluation is labeled as S or U (satisfactory or unsatisfactory). Those items classified with U were discussed at the May 15–17,[15] August 6–10,[16] and September 5,[17] 2001, DOE and NRC Technical Exchanges and Management Meetings, and agreements are available. The column labeled Technical Exchange in Table 3.2.1-2 contains tracking numbers used at these technical exchanges and management meetings to identify the NRC comments. The same tracking numbers are used in Appendix B. A notation of I/U has been used in Table 3.2.1-2 to denote screening arguments where inconsistencies have been identified. The symbol I/U is not intended as a criticism to the way the features, events, and processes have been included in the model abstraction. An isolated U (i.e., not accompanied by I or E) in Table 3.2.1-2 indicates a feature, event, and process not evaluated in a suggested integrated subissue scope. Additional details on the evaluation of screening arguments are available in Appendix B. The symbol RF identifies those features, events, and processes with screening arguments that appeal to requirements in 10 CFR Part 63 and appearing adequate. The symbol QA highlights those features, events, and processes with screening arguments invoking the implementation of quality assurance procedures. These screening arguments appear adequate pending the development of quality assurance procedures with

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

[15]Ibid.

[16]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[17]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

Table 3.2.1-2. Summary of Features, Events, and Processes Screening Argument Evaluation

Database Tracking Number	Feature, Event, and Process Name	ENG1*	ENG2	ENG3	ENG4	UZ1	UZ2	UZ3	SZ1	SZ2	Direct1	Direct2	Dose1	Dose2	Dose3	Orphan	Technical Exchange
0.1.02.00.00	Timescales of concern	–	–	–	–	–	–	–	–	–	–	–	–	–	–	I	–
0.1.03.00.00	Spatial domain of concern	–	–	–	–	–	–	–	–	–	–	–	–	–	–	I	–
0.1.09.00.00	Regulatory requirements and exclusions	I	I	I	I	I	I	I	I	I	I	I	I	I	I	–	–
0.1.10.00.00	Model and data issues	–	–	–	–	–	–	–	–	–	–	–	–	–	–	I E/S	–
1.1.01.01.00	Open-site investigation boreholes	–	–	–	–	E/QA	E/QA	–	–	–	–	–	–	–	–	–	–
1.1.01.02.00	Loss of integrity of borehole seals	–	–	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–
1.1.02.00.00	Excavation/construction	–	I E/S	–	–	–	I E/S	–	–	–	U †	–	–	–	–	–	75
1.1.02.01.00	Site flooding (during construction and operation)	–	–	–	–	E/QA	–	–	–	–	–	–	–	–	–	–	–
1.1.02.02.00	Effects of preclosure ventilation	–	I	–	–	–	I	–	–	–	–	–	–	–	–	–	–
1.1.02.03.00	Undesirable materials left	E/A	–	–	E/U	–	E/S	E/U	–	–	–	–	–	–	–	–	–
1.1.03.01.00	Error in waste or backfill emplacement	–	E/QA	–	–	–	–	–	–	–	–	–	–	–	–	–	57
1.1.04.01.00	Incomplete closure	–	–	–	–	–	E/S	–	–	–	U	–	–	–	–	–	75
1.1.05.00.00	Records and markers, repository	–	–	–	–	–	–	–	–	–	–	–	–	–	S	–	–
1.1.07.00.00	Repository design	–	I E/QA	I	I E/QA	–	I	–	–	–	I	–	–	–	–	–	–
1.1.08.00.00	Quality control	I E/QA	I E/QA	I	I E/QA	–	–	–	–	–	–	–	–	–	–	–	–
1.1.09.00.00	Schedule and planning	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–
1.1.10.00.00	Administrative control, repository site	–	–	–	–	–	–	–	–	–	–	–	–	–	–	E/S	–
1.1.11.00.00	Monitoring of repository	–	–	–	–	–	–	–	–	–	–	–	–	–	–	I	–
1.1.12.01.00	Accidents and unplanned events during operation	E/QA	E/QA	–	E/QA	–	–	–	–	–	–	–	–	–	–	E/S	–
1.1.13.00.00	Retrievability	–	–	I	I	–	–	–	–	–	–	–	–	–	–	–	–
1.2.01.01.00	Tectonic activity, large scale	–	E/S	–	–	–	–	–	–	–	E/S	–	–	–	–	–	–
1.2.02.01.00	Fractures	–	I E/S	–	–	E/A	I E/A	–	E/S	–	–	–	–	–	–	–	68
1.2.02.02.00	Faulting	–	I E/A	–	–	–	I E/A	–	I E/A	–	–	–	–	–	–	–	J-25
1.2.02.03.00	Fault movement shears waste container	–	E/A	–	–	–	–	–	–	–	–	–	–	–	–	–	J-25, J-26
1.2.03.01.00	Seismic activity	–	E/A	–	I	–	E/A	–	E/A	–	–	–	–	–	–	–	J-27
1.2.03.02.00	Seismic vibration causes container failure	–	I E/A	–	–	–	–	–	–	–	–	–	–	–	–	–	78, J-25
1.2.03.03.00	Seismicity associated with igneous activity	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–	–	–
1.2.04.01.00	Igneous activity	–	I	–	–	–	–	–	–	–	I E/S	I	I	I	I	–	–
1.2.04.02.00	Igneous activity causes changes to rock properties	–	E/S	–	–	–	E/S	E/U	E/S	E/S	E/U	–	–	–	–	–	J-22
1.2.04.03.00	Igneous intrusion into repository	–	I	–	–	–	–	–	–	–	I	–	–	–	–	–	–
1.2.04.04.00	Magma interacts with waste	I	I	–	I	–	–	–	–	–	I	–	–	–	–	–	–
1.2.04.05.00	Magmatic transport of waste	–	I E/S	–	–	–	–	–	–	–	I E/RF	I	–	–	–	–	–
1.2.04.06.00	Basaltic cinder cone erupts through the repository	–	I	–	–	–	–	–	–	–	I	I	–	–	–	–	–

Table 3.2.1-2. Summary of Features, Events, and Processes Screening Argument Evaluation (continued)

Database Tracking Number	Feature, Event, and Process Name	ENG1	ENG2	ENG3	ENG4	UZ1	UZ2	UZ3	SZ1	SZ2	Direct1	Direct2	Dose1	Dose2	Dose3	Orphan	Technical Exchange
1.2.04.07.00	Ashfall	–	–	–	–	–	–	–	E/U	–	–	–	I E/U	I E/U	U	–	8, 19
1.2.05.00.00	Metamorphism	–	–	–	–	–	–	–	–	–	–	–	–	–	–	E/S	–
1.2.06.00.00	Hydrothermal activity	–	–	E/A	–	–	E/U	–	E/S	E/A	–	–	–	–	–	–	4, J-23
1.2.07.01.00	Erosion/denudation	–	–	–	–	E/U	–	–	–	–	–	–	–	I	I E/S	–	J-16
1.2.07.02.00	Deposition	–	–	–	–	E/S	–	–	–	–	–	–	–	I	I	–	–
1.2.08.00.00	Diagenesis	–	–	–	–	–	–	E/S	–	–	–	–	–	–	–	–	–
1.2.09.00.00	Salt diapirism and dissolution	–	–	–	–	–	–	–	–	–	–	–	–	–	–	E/S	–
1.2.09.01.00	Diapirism	–	–	–	–	–	–	–	–	–	–	–	–	–	–	E/S	–
1.2.09.02.00	Large-scale dissolution	–	–	–	–	–	E/S	E/S	E/S	E/S	–	–	–	–	–	–	–
1.2.10.01.00	Hydrological response to seismic activity	–	–	–	–	E/S	E/S	–	E/S	–	–	–	–	–	–	–	J-17
1.2.10.02.00	Hydrologic response to igneous activity	–	–	–	–	E/U	E/S	–	E/S	–	–	–	–	–	–	–	–
1.3.01.00.00	Climate change, global	–	–	–	–	I	–	–	–	–	–	–	I	I	I	–	–
1.3.04.00.00	Periglacial effects	–	–	–	–	E/U	–	–	–	–	–	–	–	E/S	E/S	–	J-18
1.3.05.00.00	Glacial and ice sheet effects, local	–	–	–	–	E/S	–	–	–	–	–	–	E/S	E/S	E/S	–	–
1.3.07.01.00	Drought/water table decline	–	–	–	–	–	–	–	E/A	E/A	–	–	E/A	–	E/A	–	11
1.3.07.02.00	Water table rise	–	–	–	–	–	–	–	I	I	–	–	U	U	U	–	19
1.4.01.00.00	Human influences on climate	–	–	–	–	E/S	–	–	–	–	–	–	E/RF	–	E/RF	–	–
1.4.01.01.00	Climate modification increases recharge	–	–	–	–	I	I	–	–	–	–	–	I	–	–	–	–
1.4.01.02.00	Greenhouse gas effects	–	–	–	–	E/S	–	–	–	–	–	–	–	–	E/RF	–	–
1.4.01.03.00	Acid rain	–	–	–	–	E/RF	–	–	–	–	–	–	–	–	E/RF	–	–
1.4.01.04.00	Ozone layer failure	–	–	–	–	E/S	–	–	–	–	–	–	–	–	E/RF	–	–
1.4.02.01.00	Deliberate human intrusion	–	–	–	–	–	–	–	–	–	–	–	–	–	–	E/S	–
1.4.02.02.00	Inadvertent human intrusion	–	–	–	–	–	–	–	–	–	–	–	–	–	–	I	–
1.4.03.00.00	Unintrusive site investigation	–	–	–	–	–	–	–	–	–	–	–	–	–	–	E/S	–
1.4.04.00.00	Drilling activities (human intrusion)	–	–	–	–	–	–	–	–	–	–	–	–	–	–	I E/RF	–
1.4.04.01.00	Effects of drilling intrusion	–	–	–	–	–	–	–	–	–	–	–	–	–	–	I	–
1.4.04.02.00	Abandoned and undetected boreholes	–	–	–	–	E/S	E/S	–	–	–	–	–	–	–	–	–	–
1.4.05.00.00	Mining and other underground activities (human intrusion)	–	–	–	–	–	–	–	–	–	–	–	–	–	–	E/S	–
1.4.06.01.00	Altered soil or surface water chemistry	–	–	–	–	–	–	–	–	E/U	–	–	–	–	E/RF	–	7
1.4.07.01.00	Water management activities	–	–	–	–	–	–	–	I E/S	I E/S	–	–	I E/U	I E/U	I E/U	–	18
1.4.07.02.00	Wells	–	–	–	–	–	–	–	I	–	–	–	I E/RF	–	I E/RF	–	–
1.4.08.00.00	Social and institutional developments	–	–	–	–	–	–	–	–	–	–	–	E/RF	E/RF	E/RF	–	–
1.4.09.00.00	Technological developments	–	–	–	–	–	–	–	–	–	–	–	E/RF	E/RF	E/RF	–	–
1.4.11.00.00	Explosions and crashes (human activities)	–	–	–	–	–	–	–	–	–	–	–	–	E/RF	–	–	–
1.5.01.01.00	Meteorite impact	–	–	–	–	–	–	–	–	–	–	–	–	E/S	E/S	–	–
1.5.01.02.00	Extraterrestrial events	–	–	–	–	–	–	–	–	–	–	–	–	–	E/S	–	–
1.5.02.00.00	Species evolution	–	–	–	–	–	–	–	–	–	–	–	–	–	E/RF	–	–

Table 3.2.1-2. Summary of Features, Events, and Processes Screening Argument Evaluation (continued)

Database Tracking Number	Feature, Event, and Process Name	ENG1	ENG2	ENG3	ENG4	UZ1	UZ2	UZ3	SZ1	SZ2	Direct1	Direct2	Dose1	Dose2	Dose3	Orphan	Technical Exchange
1.5.03.01.00	Changes in the Earth's magnetic field	–	–	–	–	–	–	–	–	–	–	–	–	–	–	E/S	–
1.5.03.02.00	Earth tides	–	–	–	–	–	–	–	–	–	–	–	–	–	–	E/S	–
2.1.01.01.00	Waste inventory	–	–	–	I	–	–	–	–	–	I	–	I	I	–	–	–
2.1.01.02.00	Codisposal/co-location of waste	I	–	I	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.01.03.00	Heterogeneity of waste forms	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.01.04.00	Spatial heterogeneity of emplaced waste	E/A	–	–	E/A	–	–	–	–	–	–	–	–	–	–	–	48
2.1.02.01.00	Defense spent nuclear fuel degradation, alteration, and dissolution	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.02.00	Commercial spent nuclear fuel alteration, dissolution, and radionuclide release	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.03.00	Glass degradation, alteration, and dissolution	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.04.00	Alpha recoil enhances dissolution	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.05.00	Glass cracking and surface area	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.06.00	Glass recrystallization	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.07.00	Gap and grain release of Cs, I	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.08.00	Pyrophoricity	–	–	E/S	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.09.00	Void space (in glass container)	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.10.00	Cellulosic degradation	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.11.00	Waterlogged rods	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.12.00	Cladding degradation before YMP receives it	–	–	–	I E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.13.00	General corrosion of cladding	–	–	–	E/U	–	–	–	–	–	–	–	–	–	–	–	50
2.1.02.14.00	Microbial corrosion (MIC) of cladding	–	–	–	I/U	–	–	–	–	–	–	–	–	–	–	–	51
2.1.02.15.00	Acid corrosion of cladding from radiolysis	–	–	–	I/U	–	–	–	–	–	–	–	–	–	–	–	49,51
2.1.02.16.00	Localized corrosion (pitting) of cladding	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.17.00	Localized corrosion (crevice corrosion) of cladding	–	–	–	E/A	–	–	–	–	–	–	–	–	–	–	–	47
2.1.02.18.00	High dissolved silica content of waters enhances corrosion of cladding	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.19.00	Creep rupture of cladding	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.20.00	Pressurization from He production causes cladding failure	–	–	–	I/U	–	–	–	–	–	–	–	–	–	–	–	41
2.1.02.21.00	Stress corrosion cracking (SCC) of cladding	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.22.00	Hydride embrittlement of cladding	–	–	–	E/U	–	–	–	–	–	–	–	–	–	–	–	53
2.1.02.23.00	Cladding unzipping	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.24.00	Mechanical failure of cladding	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.25.00	Defense spent nuclear fuel cladding degradation	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.26.00	Diffusion controlled cavity growth	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.27.00	Localized corrosion perforation from fluoride	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.28.00	Various features of the approximately 250 Defense spent nuclear fuel types and grouping into waste categories	–	–	–	I E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.02.29.00	Flammable gas generation from Defense spent nuclear fuel	–	–	–	I E/S	–	–	–	–	–	–	–	–	–	–	–	–

Table 3.2.1-2. Summary of Features, Events, and Processes Screening Argument Evaluation (continued)																	
Database Tracking Number	Feature, Event, and Process Name	ENG1	ENG2	ENG3	ENG4	UZ1	UZ2	UZ3	SZ1	SZ2	Direct1	Direct2	Dose1	Dose2	Dose3	Orphan	Technical Exchange
2.1.03.01.00	Corrosion of waste containers	I	–	I	–	–	–	–	–	–	–	–	–	–	–	–	–
2.1.03.02.00	Stress corrosion cracking of waste containers	I E/A	I E/A	I	–	–	–	–	–	–	–	–	–	–	–	–	34
2.1.03.03.00	Pitting of waste containers	I	–	I	–	–	–	–	–	–	–	–	–	–	–	–	–
2.1.03.04.00	Hydride cracking of waste containers	E/S	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–	–
2.1.03.05.00	Microbially mediated corrosion of waste container	I E/A	–	I	–	–	–	–	–	–	–	–	–	–	–	–	30
2.1.03.06.00	Internal corrosion of waste container	E/S	–	I	–	–	–	–	–	–	–	–	–	–	–	–	–
2.1.03.07.00	Mechanical impact on waste container	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–	–	–
2.1.03.08.00	Juvenile and early failure of waste containers	I E/A	I E/A	–	–	–	–	–	–	–	–	–	–	–	–	–	35
2.1.03.09.00	Copper corrosion	E/S	–	–	–	–	–	–	–	–	–	–	–	–	–	–	–
2.1.03.10.00	Container healing	E/S	–	I	–	–	–	–	–	–	–	–	–	–	–	–	–
2.1.03.11.00	Container form	E/S	E/U	E/S	–	–	–	–	–	–	E/S	–	–	–	–	–	J-1
2.1.03.12.00	Container failure (long-term)	I	I	I	–	–	–	–	–	–	U	–	–	–	–	–	75
2.1.04.01.00	Preferential pathways in backfill	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–	–
2.1.04.02.00	Physical and chemical properties of backfill	–	E/S	I	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.04.03.00	Erosion or dissolution of backfill	–	E/S	E/S	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.04.04.00	Mechanical effects of backfill	–	E/S	–	–	–	–	–	–	–	E/S	–	–	–	–	–	–
2.1.04.05.00	Backfill evolution	–	E/S	I	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.04.06.00	Properties of bentonite	–	–	–	–	–	–	–	–	–	–	–	–	–	–	E/S	–
2.1.04.07.00	Buffer characteristics	–	–	–	–	–	–	–	–	–	–	–	–	–	–	E/S	–
2.1.04.08.00	Diffusion in backfill	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.04.09.00	Radionuclide transport through backfill	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.05.01.00	Seal physical properties	–	–	–	–	E/U	–	–	–	–	–	–	–	–	–	–	J-19
2.1.05.02.00	Groundwater flow and radionuclide transport in seals	–	–	–	–	E/U	E/U	–	–	–	–	–	–	–	–	–	J-19
2.1.05.03.00	Seal degradation	–	–	–	–	E/U	–	–	–	–	–	–	–	–	–	–	J-19
2.1.06.01.00	Degradation of cementitious materials in drift	–	I	I E/A	–	–	–	U	–	–	–	–	–	–	–	–	J-3
2.1.06.02.00	Effects of rock reinforcement materials	–	I	I	–	–	I	–	–	–	–	–	–	–	–	–	–
2.1.06.03.00	Degradation of the liner	–	E/S	I	E/S	–	–	E/S	–	–	–	–	–	–	–	–	–
2.1.06.04.00	Flow through the liner	–	–	E/S	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.06.05.00	Degradation of invert and pedestal	–	E/U	E/S	–	–	–	E/U	–	–	–	–	–	–	–	–	J-2, J-4
2.1.06.06.00	Effects and degradation of drip shield	I E/U	I E/A	I E/A	–	–	–	–	–	–	–	–	–	–	–	–	39
2.1.06.07.00	Effects at material interfaces	E/A	–	I	E/S	–	–	–	–	–	–	–	–	–	–	–	29
2.1.07.01.00	Rockfall (large block)	–	E/A	–	E/A	–	–	–	–	–	–	–	–	–	–	–	79
2.1.07.02.00	Mechanical degradation or collapse of drift	E/A	E/A	E/A	–	–	–	–	–	–	U	–	–	–	–	–	75, 77
2.1.07.03.00	Movement of containers	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–	–	–
2.1.07.04.00	Hydrostatic pressure on container	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–	–	–
2.1.07.05.00	Creeping of metallic materials in the engineered barrier system	–	E/A	–	–	–	–	–	–	–	–	–	–	–	–	–	37
2.1.07.06.00	Floor buckling	E/A	E/A	–	E/A	–	–	–	–	–	–	–	–	–	–	–	56

Database Tracking Number	Feature, Event, and Process Name	ENG1	ENG2	ENG3	ENG4	UZ1	UZ2	UZ3	SZ1	SZ2	Direct1	Direct2	Dose1	Dose2	Dose3	Orphan	Technical Exchange
	Table 3.2.1-2. Summary of Features, Events, and Processes Screening Argument Evaluation (continued)																
2.1.08.01.00	Increased unsaturated water flux at the repository	–	–	–	–	–	I	–	–	–	–	–	–	–	–	–	–
2.1.08.02.00	Enhanced influx (Philip's drip)	–	–	–	–	–	I	–	–	–	–	–	–	–	–	–	–
2.1.08.03.00	Repository dryout due to waste heat	–	–	–	–	–	I	–	–	–	–	–	–	–	–	–	–
2.1.08.04.00	Cold traps	–	–	E/A	–	–	E/A	–	–	–	–	–	–	–	–	–	–
2.1.08.05.00	Flow through invert	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	59
2.1.08.06.00	Wicking in waste and engineered barrier system	–	–	I	–	–	I	–	–	–	–	–	–	–	–	–	–
2.1.08.07.00	Pathways for unsaturated flow and transport in the waste and engineered barrier system	–	–	I E/A	I	–	–	–	–	–	–	–	–	–	–	–	42
2.1.08.08.00	Induced hydrological changes in the waste and engineered barrier system	–	–	I	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.08.09.00	Saturated groundwater flow in waste and engineered barrier system	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.08.10.00	Desaturation/dewatering of the repository	–	–	–	I	–	I	–	–	–	–	–	–	–	–	–	–
2.1.08.11.00	Resaturation of repository	–	–	I	–	–	I	–	–	–	–	–	–	–	–	–	–
2.1.08.12.00	Drainage with transport, sealing and plugging	–	–	E/S	–	–	E/S	–	–	–	–	–	–	–	–	–	–
2.1.08.13.00	Drains	–	–	E/S	–	–	E/S	–	–	–	–	–	–	–	–	–	–
2.1.08.14.00	Condensation on underside of drip shield	–	–	E/S	–	–	E/S	–	–	–	–	–	–	–	–	–	–
2.1.08.15.00	Waste-form and backfill consolidation	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.01.00	Properties of the potential carrier plume in the waste and engineered barrier system	–	–	I	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.02.00	Interaction with corrosion products	–	–	E/A	I	–	–	–	–	–	–	–	–	–	–	–	54
2.1.09.03.00	Volume increase of corrosion products	E/A	E/S	–	–	–	–	–	–	–	–	–	–	–	–	–	36
2.1.09.04.00	Radionuclide solubility, solubility limits, and speciation in the waste form and engineered barrier system	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.05.00	In-drift sorption	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.06.00	Reduction-oxidation potential in waste and engineered barrier system	E/S	–	I	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.07.00	Reaction kinetics in waste and engineered barrier system	I E/A	–	I E/A	I E/S	–	–	–	–	–	–	–	–	–	–	–	55
2.1.09.08.00	Chemical gradients/enhanced diffusion in waste and engineered barrier system	–	–	I	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.09.00	Electrochemical effects (electrophoresis, galvanic coupling) in waste and engineered barrier system	E/S	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.10.00	Secondary phase effects on dissolved radionuclide concentrations at the waste form	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.11.00	Waste-rock contact	–	–	E/S	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.12.00	Rind (altered zone) formation in waste, engineered barrier system, and adjacent rock	–	I E/A	I	I	E/A	I E/A	E/S	–	–	–	–	–	–	–	–	63
2.1.09.13.00	Complexation by organics in waste and engineered barrier system	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.14.00	Colloid formation in waste and engineered barrier system	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–

Table 3.2.1-2. Summary of Features, Events, and Processes Screening Argument Evaluation (continued)

Database Tracking Number	Feature, Event, and Process Name	ENG1	ENG2	ENG3	ENG4	UZ1	UZ2	UZ3	SZ1	SZ2	Direct1	Direct2	Dose1	Dose2	Dose3	Orphan	Technical Exchange
2.1.09.15.00	Formation of true colloids in waste and engineered barrier system	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.16.00	Formation of pseudo-colloids (natural) in waste and engineered barrier system	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.17.00	Formation of pseudo-colloids (corrosion products) in waste and engineered barrier system	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.18.00	Microbial colloid transport in the waste and engineered barrier system	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.19.00	Colloid transport and sorption in the waste and engineered barrier system	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.20.00	Colloid filtration in the waste and engineered barrier system	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.21.00	Suspensions of particles larger than colloids	–	–	–	E/U	–	–	E/U	I	U	–	–	I	–	–	–	J-5, 5
2.1.09.22.00	Colloid sorption at the air-water interface	–	–	–	–	–	–	E/S	–	E/S	–	–	–	–	–	–	–
2.1.09.23.00	Colloidal stability and concentration dependence on aqueous chemistry	–	–	–	–	–	–	I	–	I	–	–	–	–	–	–	–
2.1.09.24.00	Colloidal diffusion	–	–	–	–	–	–	I	–	I	–	–	–	–	–	–	–
2.1.09.25.00	Colloidal phases are produced by coprecipitation (in waste and engineered barrier system)	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.09.26.00	Colloid gravitational settling	–	–	–	E/S	–	–	E/S	–	–	–	–	–	–	–	–	–
2.1.10.01.00	Biological activity in waste and engineered barrier system	I E/S	–	I	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.11.01.00	Heat output/temperature in waste and engineered barrier system	I	I	I	I	–	I	–	–	–	–	–	–	–	–	–	–
2.1.11.02.00	Nonuniform heat distribution/edge effects in repository	–	–	I	I	–	I E/A	–	–	–	–	–	–	–	–	–	65
2.1.11.03.00	Exothermic reactions in waste and engineered barrier system	–	–	E/S	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.11.04.00	Temperature effects/coupled processes in waste and engineered barrier system	I	I	I	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.11.05.00	Differing thermal expansion of repository components	I E/U	I E/U	–	–	–	–	–	–	–	–	–	–	–	–	–	38
2.1.11.06.00	Thermal sensitization of waste containers increases fragility	I	I	–	–	–	–	–	–	–	–	–	–	–	–	–	–
2.1.11.07.00	Thermally induced stress changes in waste and engineered barrier system	–	I	–	I E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.11.08.00	Thermal effects: chemical and microbiological changes in the waste and engineered barrier system	–	–	I	I E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.11.09.00	Thermal effects on liquid or two-phase fluid flow in the waste and engineered barrier system	–	–	I	I	–	–	–	–	–	–	–	–	–	–	–	–
2.1.11.10.00	Thermal effects on diffusion (Soret effect) in waste and engineered barrier system	E/S	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.12.01.00	Gas generation	E/S	–	E/U	–	–	–	–	–	–	–	–	–	–	–	–	60
2.1.12.02.00	Gas generation (He) from fuel decay	E/S	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.12.03.00	Gas generation (H_2) from metal corrosion	E/S	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–	–

Table 3.2.1-2. Summary of Features, Events, and Processes Screening Argument Evaluation (continued)

Database Tracking Number	Feature, Event, and Process Name	ENG1	ENG2	ENG3	ENG4	UZ1	UZ2	UZ3	SZ1	SZ2	Direct1	Direct2	Dose1	Dose2	Dose3	Orphan	Technical Exchange
2.1.12.04.00	Gas generation (CO_2, CH_4, H_2S) from microbial degradation	E/S	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–	–
2.1.12.05.00	Gas generation from concrete	–	–	E/U	–	–	–	–	–	–	–	–	–	–	–	–	60
2.1.12.06.00	Gas transport in waste and engineered barrier system	E/S	–	E/S	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.12.07.00	Radioactive gases in waste and engineered barrier system	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.12.08.00	Gas explosions	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.13.01.00	Radiolysis	E/A	–	E/U	E/U	–	–	–	–	–	–	–	–	–	–	–	32
2.1.13.02.00	Radiation damage in waste and engineered barrier system	E/S	E/S	E/S	E/S	–	–	–	–	–	–	–	–	–	–	–	–
2.1.13.03.00	Mutation	–	–	–	–	–	–	–	–	–	–	–	–	–	–	E/S	–
2.1.14.01.00	Criticality in waste and engineered barrier system	–	–	–	E/A	–	–	–	–	–	–	–	–	–	–	–	74
2.1.14.02.00	Criticality *in situ*, nominal configuration, top breach	–	–	–	E/A	–	–	–	–	–	–	–	–	–	–	–	74
2.1.14.03.00	Criticality *in situ*, waste package internal structures degrade faster than waste form, top breach	–	–	–	E/A	–	–	–	–	–	–	–	–	–	–	–	74
2.1.14.04.00	Criticality *in situ*, waste package internal structures degrade at same rate as waste form, top breach	–	–	–	E/A	–	–	–	–	–	–	–	–	–	–	–	74
2.1.14.05.00	Criticality *in situ*, waste package internal structures degrade slower than waste form, top breach	–	–	–	E/A	–	–	–	–	–	–	–	–	–	–	–	74
2.1.14.06.00	Criticality *in situ*, waste form degrades in place and swells, top breach	–	–	–	E/A	–	–	–	–	–	–	–	–	–	–	–	74
2.1.14.07.00	Criticality *in situ*, bottom breach allows flow through waste package, fissile material collects at bottom of waste package	–	–	–	E/A	–	–	–	–	–	–	–	–	–	–	–	74
2.1.14.08.00	Criticality *in situ*, bottom breach allows flow through waste package, waste form degrades in place	–	–	–	E/A	–	–	–	–	–	–	–	–	–	–	–	74
2.1.14.09.00	Near-field criticality, fissile material deposited in near-field pond	–	–	–	E/A	–	–	–	–	–	–	–	–	–	–	–	74
2.1.14.10.00	Near-field criticality, fissile solution flows into drift lowpoint	–	–	–	E/A	–	–	–	–	–	–	–	–	–	–	–	74
2.1.14.11.00	Near-field criticality, fissile solution is adsorbed or reduced in invert	–	–	–	E/A	–	–	–	–	–	–	–	–	–	–	–	74
2.1.14.12.00	Near-field criticality, filtered slurry or colloidal stream collects on invert surface	–	–	–	E/A	–	–	–	–	–	–	–	–	–	–	–	74
2.1.14.13.00	Near-field criticality associated with colloidal deposits	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	74
2.1.14.14.00	Out-of-package criticality, fuel/magma mixture	–	–	–	–	–	–	–	–	–	–	–	–	–	–	E/S	–
2.2.01.01.00	Excavation and construction-related changes in the adjacent host rock	–	I E/S	–	–	–	I E/A	–	–	–	–	–	–	–	–	–	69
2.2.01.02.00	Thermal and other waste and engineered barrier system-related changes in the adjacent host rock	–	E/A	–	–	–	E/A	E/S	–	–	–	–	–	–	–	–	62
2.2.01.03.00	Changes in fluid saturations in the excavation disturbed zone	–	–	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–
2.2.01.04.00	Elemental solubility in excavation disturbed zone	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–	–

Table 3.2.1-2. Summary of Features, Events, and Processes Screening Argument Evaluation (continued)

Database Tracking Number	Feature, Event, and Process Name	ENG1	ENG2	ENG3	ENG4	UZ1	UZ2	UZ3	SZ1	SZ2	Direct1	Direct2	Dose1	Dose2	Dose3	Orphan	Technical Exchange
2.2.01.05.00	Radionuclide transport in excavation disturbed zone	–	–	–	–	–	–	E/S	–	–	–	–	–	–	–	–	–
2.2.03.01.00	Stratigraphy	–	–	–	–	I	I	I	I	I	I	–	–	–	–	–	–
2.2.03.02.00	Rock properties of host rock and other units	–	I	–	–	I	I	I	I	I	–	–	–	–	–	–	–
2.2.06.01.00	Changes in stress (due to thermal, seismic, or tectonic effects) change porosity and permeability of rock	–	E/A	E/A	–	–	E/A	–	E/S	–	–	–	–	–	–	–	66
2.2.06.02.00	Changes in stress (due to thermal, seismic, or tectonic effects) produce change in permeability of faults	–	–	–	–	–	E/S	–	E/S	–	–	–	–	–	–	–	–
2.2.06.03.00	Changes in stress (due to seismic or tectonic effects) alter perched water zones	–	–	–	–	–	I	–	I	–	–	–	–	–	–	–	–
2.2.06.04.00	Effects of subsidence	–	–	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–
2.2.06.05.00	Salt creep	–	–	–	–	–	–	–	–	–	–	–	–	–	–	E/S	–
2.2.07.01.00	Locally saturated flow at bedrock/alluvium contact	–	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–
2.2.07.02.00	Unsaturated groundwater flow in geosphere	–	–	–	–	I	I	–	–	–	–	–	–	–	–	–	–
2.2.07.03.00	Capillary rise	–	–	–	–	–	–	–	–	–	–	–	–	–	E/RF	–	–
2.2.07.04.00	Focusing of unsaturated flow (fingers, weeps)	–	–	–	–	I	I	–	–	–	–	–	–	–	–	–	–
2.2.07.05.00	Flow and transport in the unsaturated zone from episodic infiltration	–	–	–	–	I	I E/A	–	–	–	–	–	–	–	–	–	20
2.2.07.06.00	Episodic/pulse release from repository	–	–	–	I	–	–	–	–	–	–	–	–	–	–	–	–
2.2.07.07.00	Perched water develops	–	–	–	–	–	I	I	–	–	–	–	–	–	–	–	–
2.2.07.08.00	Fracture flow in the unsaturated zone	–	–	–	–	–	I	–	–	–	–	–	–	–	–	–	–
2.2.07.09.00	Matrix imbibition in the unsaturated zone	–	–	–	–	–	I	–	–	–	–	–	–	–	–	–	–
2.2.07.10.00	Condensation zone forms around drifts	–	–	I	–	–	I	–	–	–	–	–	–	–	–	–	–
2.2.07.11.00	Return flow from condensation cap/resaturation of dryout zone	–	–	I	–	–	I	–	–	–	–	–	–	–	–	–	–
2.2.07.12.00	Saturated groundwater flow	–	–	–	–	–	–	–	I	–	–	–	I	–	–	–	–
2.2.07.13.00	Water-conducting features in the saturated zone	–	–	–	–	–	–	–	I	–	–	–	I	–	–	–	–
2.2.07.14.00	Density effects on groundwater flow	–	–	–	–	–	–	–	E/S	–	–	–	E/S	–	–	–	–
2.2.07.15.00	Advection and dispersion	–	–	–	–	–	–	U	I	I	–	–	–	–	–	–	J-8
2.2.07.16.00	Dilution of radionuclides in groundwater	–	–	–	–	–	–	–	I	I	–	–	I	–	–	–	–
2.2.07.17.00	Diffusion in the saturated zone	–	–	–	–	–	–	–	–	I	–	–	I	–	–	–	–
2.2.07.18.00	Film flow into drifts	–	–	–	–	–	I E/A	–	–	–	–	–	–	–	–	–	USFIC-1
2.2.07.19.00	Lateral flow from Solitario Canyon fault enters potential waste emplacement drifts	–	–	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–
2.2.08.01.00	Groundwater chemistry/composition in unsaturated zone and saturated zone	–	–	–	–	–	–	I E/U	–	I	–	–	–	–	U	–	19
2.2.08.02.00	Radionuclide transport occurs in a carrier plume in geosphere	–	–	–	–	–	–	E/U	I	I	–	–	U	U	U	–	J-8
2.2.08.03.00	Geochemical interactions in geosphere (dissolution, precipitation, weathering) and effects on radionuclide transport	–	–	–	–	–	E/U	E/U	–	I	–	–	–	–	–	–	J-9

Table 3.2.1-2. Summary of Features, Events, and Processes Screening Argument Evaluation (continued)

Database Tracking Number	Feature, Event, and Process Name	ENG1	ENG2	ENG3	ENG4	UZ1	UZ2	UZ3	SZ1	SZ2	Direct1	Direct2	Dose1	Dose2	Dose3	Orphan	Technical Exchange
2.2.08.04.00	Redissolution of precipitates directs more corrosive fluids to containers	–	–	I	–	–	I	–	–	–	–	–	–	–	–	–	–
2.2.08.05.00	Osmotic processes	–	–	–	–	–	–	E/S	–	–	–	–	–	–	–	–	–
2.2.08.06.00	Complexation in geosphere	–	–	–	–	–	–	I E/U	–	I	–	–	I	–	–	–	J-10
2.2.08.07.00	Radionuclide solubility limits in the geosphere	–	–	–	–	–	–	E/U	–	I	–	–	I	U	–	–	20, J-11
2.2.08.08.00	Matrix diffusion in geosphere	–	–	–	–	–	I	I	–	I	–	–	–	–	–	–	–
2.2.08.09.00	Sorption in unsaturated zone and saturated zone	–	–	–	–	–	–	I	–	I	–	–	–	–	–	–	–
2.2.08.10.00	Colloidal transport in geosphere	–	–	–	–	–	–	I	–	I	–	–	I	–	–	–	–
2.2.08.11.00	Distribution and release of nuclides from the geosphere	–	–	–	–	–	–	–	–	–	–	–	U	I	U	–	19
2.2.08.14.00	Condensation on underside of drip shield	E/S	–	E/S	–	–	E/S	–	–	–	–	–	–	–	–	–	–
2.2.09.01.00	Microbial activity in geosphere	–	–	–	–	–	–	E/S	–	I	–	–	–	–	–	–	–
2.2.10.01.00	Repository-induced thermal effects in geosphere	–	–	I	–	I E/S	I	I E/U	E/S	E/U	–	–	–	–	–	–	J-12
2.2.10.02.00	Thermal convection cell develops in saturated zone	–	–	–	–	–	–	–	E/U	–	–	–	–	–	–	–	13
2.2.10.03.00	Natural geothermal effects	–	–	–	–	–	I/A	–	I/A	I/A	–	–	–	–	–	–	3
2.2.10.04.00	Thermo-mechanical alteration of fractures near repository	–	E/A	E/A	–	–	E/A	–	–	–	–	–	–	–	–	–	70
2.2.10.05.00	Thermo-mechanical alteration of rocks above and below the repository	–	–	–	–	–	E/U	–	–	–	–	–	–	–	–	–	67
2.2.10.06.00	Thermo-chemical alteration (solubility, speciation, phase changes, precipitation/dissolution)	–	–	I E/A	–	–	–	E/A E/U	E/U	E/U	–	–	–	–	–	–	J-13, 9, 64
2.2.10.07.00	Thermo-chemical alteration of the Calico Hills unit	–	–	–	–	–	E/U	E/U	–	–	–	–	–	–	–	–	J-14
2.2.10.08.00	Thermo-chemical alteration of the saturated zone	–	–	–	–	–	–	–	E/U	E/U	–	–	–	–	–	–	9
2.2.10.09.00	Thermo-chemical alteration of the Topopah Spring basal vitrophyre	–	–	–	–	–	E/U	E/U	–	–	–	–	–	–	–	–	J-15
2.2.10.10.00	Two-phase bouyant flow/heat pipes	–	–	–	–	–	I	–	–	–	–	–	–	–	–	–	–
2.2.10.11.00	Natural air flow in unsaturated zone	–	–	–	–	E/S	E/S	–	–	–	–	–	–	–	–	–	–
2.2.10.12.00	Geosphere dryout due to waste heat	–	–	–	–	U	I	–	–	–	–	–	–	–	–	–	61
2.2.10.13.00	Density-driven groundwater flow (thermal)	–	–	–	–	–	I	–	E/S	I E/A	–	–	–	–	–	–	12
2.2.10.14.00	Mineralogic dehydration reactions	–	–	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–
2.2.11.01.00	Naturally occurring gases in geosphere	–	–	–	–	–	–	E/S	E/S	E/S	–	–	–	–	–	–	–
2.2.11.02.00	Gas pressure effects	–	–	–	–	–	E/U	–	–	–	–	–	–	–	–	–	J-21
2.2.11.03.00	Gas transport in geosphere	–	–	–	–	–	–	I E/S	–	–	–	–	–	–	–	–	–
2.2.12.00.00	Undetected features (in geosphere)	–	–	–	–	E/S	E/S	–	I	I	–	–	I	–	–	–	–
2.2.14.01.00	Critical assembly forms away from repository	–	–	–	–	–	–	E/A	–	E/A	–	–	–	–	–	–	74
2.2.14.02.00	Far-field criticality, precipitation in organic reducing zone in or near water table	–	–	–	–	–	–	E/A	–	E/A	–	–	–	–	–	–	74
2.2.14.03.00	Far-field criticality, sorption on clay/zeolite in Topopah Springs basal vitrophyre	–	–	–	–	–	–	E/A	–	E/A	–	–	–	–	–	–	74

Table 3.2.1-2. Summary of Features, Events, and Processes Screening Argument Evaluation (continued)

Database Tracking Number	Feature, Event, and Process Name	ENG1	ENG2	ENG3	ENG4	UZ1	UZ2	UZ3	SZ1	SZ2	Direct1	Direct2	Dose1	Dose2	Dose3	Orphan	Technical Exchange
2.2.14.04.00	Far-field criticality, precipitation caused by hydrothermal upwell or redox front in the saturated zone	–	–	–	–	–	–	E/A	–	E/A	–	–	–	–	–	–	74
2.2.14.05.00	Far-field criticality, precipitation in perched water above Topopah Springs basal vitrophyre	–	–	–	–	–	–	E/A	–	E/A	–	–	–	–	–	–	74
2.2.14.06.00	Far-field criticality, precipitation in fractures of Topopah Springs welded rock	–	–	–	–	–	–	E/A	–	E/A	–	–	–	–	–	–	74
2.2.14.07.00	Far-field criticality, dryout produces fissile salt in a perched water basin	–	–	–	–	–	–	E/A	–	E/A	–	–	–	–	–	–	74
2.2.14.08.00	Far-field criticality associated with colloidal deposits	–	–	–	–	–	–	E/A	–	E/A	–	–	–	–	–	–	74
2.3.01.00.00	Topography and morphology	–	–	–	–	I	–	–	–	–	U	–	–	U	–	–	75, IA-1
2.3.02.01.00	Soil type	–	–	–	–	–	–	–	–	–	–	–	–	I	I	–	–
2.3.02.02.00	Radionuclide accumulation in soils	–	–	–	–	–	–	–	I	–	I E/U	–	–	I E/U	I E/RF	–	IA-1
2.3.02.03.00	Soil and sediment transport	–	–	–	–	–	–	–	–	–	–	–	–	I E/U	I E/RF	–	IA-1
2.3.04.01.00	Surface water transport and mixing	–	–	–	–	–	–	–	–	–	–	–	–	E/S	E/S	–	–
2.3.06.00.00	Marine features	–	–	–	–	–	–	–	–	–	–	–	–	E/S	E/S	–	–
2.3.09.01.00	Animal burrowing/intrusion	–	–	–	–	–	–	–	–	–	–	–	–	E/S	E/S	–	–
2.3.11.01.00	Precipitation	–	–	–	–	I	–	–	–	–	–	–	–	I	I	–	–
2.3.11.02.00	Surface runoff and flooding	–	–	–	–	I	–	–	–	–	–	–	–	U	I	–	IA-1
2.3.11.03.00	Infiltration and recharge (hydrologic and chemical effects)	–	–	I	–	I	–	–	–	–	–	–	–	I	–	–	–
2.3.11.04.00	Groundwater discharge to surface	–	–	–	–	–	–	–	E/S	E/U	–	–	E/S	E/S	U	–	10, 19
2.3.13.01.00	Biosphere characteristics	–	–	–	–	I E/S	–	–	–	–	–	–	I E/S	I E/U	I E/U	–	21
2.3.13.02.00	Biosphere transport	–	–	–	–	–	–	–	–	–	–	–	–	I E/U	I E/U	–	24, IA-1
2.3.13.03.00	Effects of repository heat on biosphere	–	–	–	–	E/S	–	–	–	–	–	–	–	–	–	–	–
2.4.01.00.00	Human characteristics (physiology, metabolism)	–	–	–	–	–	–	–	–	–	–	–	–	–	I E/RF	–	–
2.4.03.00.00	Diet and fluid intake	–	–	–	–	–	–	–	–	–	–	–	–	–	I E/RF	–	–
2.4.04.01.00	Human lifestyle	–	–	–	–	–	–	–	–	–	–	–	I E/RF	I E/RF	I E/RF	–	–
2.4.07.00.00	Dwellings	–	–	–	–	–	–	–	–	–	–	–	–	–	I E/U	–	25
2.4.08.00.00	Wild and natural land and water use	–	–	–	–	–	–	–	–	–	–	–	E/RF	E/RF	E/RF	–	–
2.4.09.01.00	Agricultural land use and irrigation	–	–	–	–	–	–	–	–	–	–	–	I	I	I E/RF	–	–
2.4.09.02.00	Animal farms and fisheries	–	–	–	–	–	–	–	–	–	–	–	I	–	I	–	–
2.4.10.00.00	Urban and industrial land and water use	–	–	–	–	–	–	–	–	–	–	–	E/RF	E/RF	E/RF	–	–
3.1.01.01.00	Radioactive decay and ingrowth	–	–	–	I	–	–	I	–	I	–	–	U	I	U	–	19
3.2.07.01.00	Isotopic dilution	–	–	–	–	–	–	I E/S	I	–	–	–	I	I	–	–	–

Table 3.2.1-2. Summary of Features, Events, and Processes Screening Argument Evaluation (continued)

Database Tracking Number	Feature, Event, and Process Name	ENG1	ENG2	ENG3	ENG4	UZ1	UZ2	UZ3	SZ1	SZ2	Direct1	Direct2	Dose1	Dose2	Dose3	Orphan	Technical Exchange
3.2.10.00.00	Atmospheric transport of contaminants	–	–	–	–	–	–	–	–	–	–	I	I E/S	I E/S	I E/S	–	–
3.3.01.00.00	Drinking water, foodstuffs and drugs, contaminant concentrations in	–	–	–	–	–	–	–	–	–	–	–	–	–	I	–	–
3.3.02.01.00	Plant uptake	–	–	–	–	–	–	–	–	–	–	–	–	–	I E/RF	–	–
3.3.02.02.00	Animal uptake	–	–	–	–	–	–	–	–	–	–	–	–	–	I E/RF	–	–
3.3.02.03.00	Bioaccumulation	–	–	–	–	–	–	–	–	–	–	–	–	–	I	–	–
3.3.03.01.00	Contaminated nonfood products and exposure	–	–	–	–	–	–	–	–	–	–	–	–	–	E/RF	–	–
3.3.04.01.00	Ingestion	–	–	–	–	–	–	–	–	–	–	–	–	–	I	–	–
3.3.04.02.00	Inhalation	–	–	–	–	–	–	–	–	–	–	–	–	–	I	–	–
3.3.04.03.00	External exposure	–	–	–	–	–	–	–	–	–	–	–	–	–	I E/S	–	–
3.3.05.01.00	Radiation doses	–	–	–	–	–	–	–	–	–	–	–	–	–	I	–	–
3.3.06.00.00	Radiological toxicity/effects	–	–	–	–	–	–	–	–	–	–	–	–	–	E/RF	–	–
3.3.06.01.00	Toxicity of mined rock	–	–	–	–	–	–	–	–	–	–	–	–	–	–	E/S	–
3.3.06.02.00	Sensitization to radiation	–	–	–	–	–	–	–	–	–	–	–	–	–	E/RF	–	–
3.3.07.00.00	Nonradiological toxicity/effects	–	–	–	–	–	–	–	–	–	–	–	–	–	E/RF	–	–
3.3.08.00.00	Radon and radon daughter exposure	–	–	–	–	–	–	–	–	–	–	–	–	–	E/U	–	26

*See Table 1.1-2 for definitions of integrated subissues.

†See Appendix B for path forward to progress from unsatisfactory (U) to satisfactory (S)

Notations That Refer to Integrated Subissues

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

Symbols

S	Satisfactory
U	Initially evaluated as Unsatisfactory (items already discussed with DOE, and agreements have been produced to address concern)
I	Included
E	Excluded
A	Existing DOE/NRC Technical Exchange Agreements are related to screening argument
RF	Screening argument based on 10 CFR Part 63
QA	Screening based on not yet implemented quality assurance procedures; acceptance is pending elaboration of such procedures

objectives consistent with those cited in the screening arguments. Finally, the symbol A identifies those entries for which screening arguments related to or dependent on work needed to satisfy agreements reached at DOE and NRC key technical issue technical exchanges. Appendix B contains details on why some screening arguments were initially classified as unsatisfactory. The comments are listed in ascending order according to database tracking numbers with the exception of the first entries, which address general comments applicable to multiple features, events, and processes. All comments in Appendix B have been discussed with DOE at the May 15–17[18] and August 6–10,[19] 2001, DOE and NRC Technical Exchanges and Management Meetings on Total System Performance Assessment and Integration, and at the September 5,[20] 2001, Technical Exchange and Management Meeting on Igneous Activity. Tracking numbers assigned to the NRC comments at these technical exchanges and the agreed-on paths forward are also included in Appendix B.

In general, DOE agreed to clarify screening arguments or provide technical bases supporting screening decisions. For those features, events, and processes related to existing DOE and NRC agreements, DOE agreed to revise the screening arguments in pertinent analysis and model reports after completion of the work needed to satisfy the agreements. DOE also agreed to expand the scope of analyses and model reports addressing features, events, and processes, to contain relevant items not currently in their scope, and clarify the definition of some features, events, and processes. Details of the concerns and agreed-on paths forward are contained in Appendix B. The agreements reached between DOE and NRC are listed in Section 3.2.1.5.

3.2.1.4.3 Formation of Scenario Classes Using the Reduced Set of Events Is Adequate

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.2.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the adequacy of the formation of scenario classes using the reduced set of events.

DOE indicated that included features, events, and processes are combined in two possible scenario classes (disruptive and nominal), and both classes would be represented in the total

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

[19]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[20]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

system performance assessment[21] (CRWMS M&O, 2000a). The nominal scenario class includes all features, events, and processes assumed to occur during 10,000 years, and the disruptive scenario class encompasses features, events, and processes related to igneous activity (CRWMS M&O, 2000a). This approach to scenario class formation is appropriate. Adequate formation of scenario classes depends in part on a complete identification of features, events, and processes, development of appropriate screening rationale, and screening decisions for features, events, and processes (i.e., either to be included or not into the performance assessment). For example, features, events, and processes exist for which a screening decision could impact the identification of scenario classes such as 2.1.07.02.00 (Mechanical Degradation or Collapse of Drift), given potential implications of drift collapse on temperature, chemistry, seepage rates, and drip shield performance. Nonetheless, the information provided by DOE on its current approach to form scenario classes is sufficient for NRC to make a regulatory decision at the time of future license application.

3.2.1.4.4 Screening of Scenario Classes Is Appropriate

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.2.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the appropriateness of the screening of scenario classes.

DOE indicated that both the disruptive and nominal scenario classes are represented in the total system performance assessment[22] (CRWMS M&O, 2000a,b). Thus, none of the scenario classes identified so far will be screened out from the performance assessment.

3.2.1.5 Status and Path Forward

Table 3.2.1-3 provides related DOE and NRC agreements pertaining to the Scenario Analysis, as well as the status of the associated key technical issue subissues. Note that the status as well as the detailed agreements pertaining to all the key technical issue subissues are provided in Table 1.1-3 and Appendix A. Details on the agreed-on paths forward to address NRC questions on the screening of features, events, and processes discussed at the May 15–17[23] and 6–10,[24] 2001, DOE and NRC Technical Exchanges and Management Meetings, are presented in Appendix B.

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[21]Swift, P. "TSPA-SR Features, Events, and Processes Approach: Process and Methodology." *Presentation at the DOE and NRC Technical Exchange on Total System Performance Assessment (TSPA) for Yucca Mountain, San Antonio, TX. June 6–7, 2000.* San Antonio, Texas. 2000.

[22]Ibid.

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

[24]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

The NRC staff have confidence the DOE proposed approach, together with DOE agreements to provide NRC with additional information (through specified testing, analyses, and the like), acceptably addresses NRC questions so that no information beyond that provided, or agreed to, will likely be required at the time of an initial license application.

Table 3.2.1-3. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreements*
Container Life and Source Term	Subissue 3—Rate at Which Radionuclides in Spent Nuclear Fuel Are Released from the Engineered Barrier Subsystem through the Oxidation and Dissolution of Spent Fuel	Closed-Pending	CLST.3.01 CLST.3.04
	Subissue 4—Rate at Which Radionuclides in High-Level Waste Glass are Leached and Released from the Engineered Barrier Subsystem	Closed-Pending	CLST.4.01 CLST.4.04
	Subissue 5—Effect of In-Package Criticality on Waste Package and Engineered Barrier Subsystem Performance	Closed-Pending	CLST.5.01 CLST.5.02 CLST.5.03 CLST.5.06 CLST.5.07
Evolution of the Near-Field Environment	Subissue 1—Effects of Coupled Thermal-Hydrologic-Chemical Processes on Seepage and Flow	Closed-Pending	ENFE.1.01 ENFE.1.02 ENFE.1.06
	Subissue 2—Effects of Coupled Thermal-Hydrologic-Chemical Processes on Waste Package Chemical Environment	Closed-Pending	ENFE.2.01 ENFE.2.02 ENFE.2.03
	Subissue 4—Effects of Coupled Thermal-Hydrologic-Chemical Processes on Radionuclide Transport through Engineered and Natural Barriers	Closed-Pending	ENFE.4.03 through ENFE.4.08
	Subissue 5—Effects of Coupled Thermal-Hydrologic-Chemical Processes on Potential Nuclear Criticality in the Near Field	Closed-Pending	ENFE.5.01 ENFE.5.02
Igneous Activity	Subissue 1—Probability of Future Igneous Activity	Closed-Pending	IA.1.01 IA.1.02

Table 3.2.1-3. Related Key Technical Issue Subissues and Agreements (continued)

Key Technical Issue	Subissue	Status	Related Agreements*
Repository Design and Thermal-Mechanical Effects	Subissue 3—Thermal-Mechanical Effects on Underground Facility Design and Performance	Closed-Pending	RDTME.3.19
Radionuclide Transport	Subissue 1—Radionuclide Transport through Porous Rock	Closed-Pending	RT.1.03
	Subissue 2—Radionuclide Transport through Alluvium	Closed-Pending	RT.2.02 RT.2.10 RT.2.11
	Subissue 4—Nuclear Criticality in the Far Field	Closed-Pending	RT.4.01 RT.4.02
Structural Deformation and Seismicity	Subissue 1—Faulting	Closed-Pending	SDS.1.01
	Subissue 2—Seismicity	Closed-Pending	SDS.2.02
Thermal Effects on Flow	Subissue 1—Features, Events, and Processes Related to Thermal Effects on Flow	Closed-Pending	TEF.1.01 TEF.1.02
Unsaturated and Saturated Flow Under Isothermal Conditions	Subissue 5—Saturated Zone Ambient Flow Conditions and Dilution Processes	Closed-Pending	USFIC.5.14
Total System Performance Assessment and Integration	Subissue 1—System Description and Demonstration of Multiple Barriers	Close-Pending	TSPAI.1.01 TSPAI.1.02
	Subissue 2—Scenario Analysis and Event Probability	Closed-Pending	TSPAI.2.01 through TSPAI2.07
	Subissue 3—Model Abstraction	Closed-Pending	TSPAI.3.01 through TSPAI 3.42
	Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards	Closed-Pending	TSPAI.4.01 through TSPAI.4.07

*Related DOE and NRC agreements are associated with one or all four generic acceptance criteria.

3.2.1.6 References

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

———. "Features, Events and Processes: Screening for Disruptive Events." ANL–WIS–MD–000005. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000b.

———. "Features, Events, and Processes in SZ Flow and Transport." ANL–NBS–MD–000002. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2000c.

———. "Features, Events, and Processes: System-Level and Criticality." ANL–WIS–MD–000019. Revision 00. Las Vegas, Nevada: CRWMS M&O. 2000d.

———. "The Development of Information Catalogued in Rev 00 of the YMP FEP Database." TDR–WIS–MD–000003. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2001a.

———. "Yucca Mountain FEP Database." TDR–WIS–MD–000003. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2001b.

———. "FEPs Screening of Processes and Issues in Drip Shield and Waste Package Degradation." ANL–EBS–PA–000002. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2001c.

———. "Features, Events, and Processes in UZ Flow and Transport." ANL–NBS–MD–000001. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2001d.

———. "Features, Events, and Processes in Thermal Hydrology and Coupled Processes." ANL–NBS–MD–000004. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2001e.

———. "EBS FEPs/Degradation Modes Abstraction." ANL–WIS–PA–000002. Revision 01. Las Vegas, Nevada: CRWMS M&O. 2001f.

NRC. "Issue Resolution Status Report. Key Technical Issue: Total System Performance Assessment and Integration." Revision 3. Washington, DC: NRC. 2000.

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

Pickett, D.A. and B.W. Leslie. "An Audit of the DOE Treatment of Features, Events, and Processes at Yucca Mountain, Nevada, with Emphasis on the Evolution of the Near-Field Environment." San Antonio, Texas: CNWRA. 1999.

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

3.2.2.1 Description of Issue

The Identification of Events with Probabilities Greater Than 10^{-8} Per Year is necessary to ensure that all significant events have been included in demonstrating compliance with the postclosure performance objective at 10 CFR 60.113. (See requirements for performance assessment at 10 CFR 60.114.) The identification of events with probabilities greater than 10^{-8} per year includes the following parts: (i) appropriate definition of events and event sequences, (ii) appropriate determination of the annual probability of each event with sufficient technical basis, (iii) appropriate use of conceptual models to determine the probability of events, (iv) use of appropriate parameters to define the probability of events, and (v) appropriate consideration of uncertainty in models and parameters used to calculate the probability of events.

This section provides a review of the methodologies used by DOE to identify the events that have a probability of occurrence at the Yucca Mountain repository greater than 10^{-8} per year in its Total System Performance Assessment. The DOE description and technical basis for the Identification of Events with Probabilities Greater Than 10^{-8} Per Year are documented in CRWMS M&O (2000a), five supporting analysis and model reports, and a calculational package (CRWMS M&O, 2000b). Portions of additional analysis and model reports are reviewed because they contain data or analyses that support the proposed Total System Performance Assessment abstractions.

3.2.2.2 Relationship to Key Technical Issue Subissues

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

- Igneous Activity
- Faulting
- Seismicity
- Nuclear Criticality

According to 10 CFR Part 63, the disruption of the repository because of human intrusion will be analyzed using a stylized scenario, and the probability of this event class does not have to be determined. The technical basis for the assignment of probability values to these event classes has been previously captured within the framework of the following key technical issue subissues:

- Igneous Activity: Subissue 1—Probability of Igneous Activity (NRC, 1999a)

- Structural Deformation and Seismicity: Subissue 1—Faulting (NRC, 1999b)

- Structural Deformation and Seismicity: Subissue 2—Seismicity (NRC, 1999b)

Repository Safety After Permanent Closure

- Container Life and Source Term: Subissue 5—The Effect of In-Package Criticality on Waste Package and Engineered Barrier Subsystem Performance (NRC, 2001)
- Evolution of the Near-Field Environment: Subissue 4—Effects of Coupled Thermal-Hydrologic-Chemical Processes on Radionuclide Transport Through Engineered and Natural Barriers (NRC, 2000a)
- Radionuclide Transport: Subissue 4—Nuclear Criticality in the Far Field (NRC, 2000b)
- Total System Performance Assessment and Integration: Subissue 2—Scenario Analysis and Event Probability (NRC, 2000c)
- Total System Performance Assessment and Integration: Subissue 3—Model Abstraction (NRC, 2000c)

The key technical issue subissues formed the bases for the previous version of the issue resolution status reports and also were the bases for technical exchanges with DOE where agreements were reached about what additional information DOE needed to provide to resolve the subissue. The resolution status of the Scenario Analysis and Event Probability Subissue is based on the resolution status of the contributing key technical issue subissues. The subsequent sections incorporate applicable portions of these key technical issue subissues. No effort was made, however, to explicitly identify each subissue.

3.2.2.3 Importance to Postclosure Performance

One aspect of risk informing the NRC review was to determine how the Identification of Events with Probabilities Greater Than 10^{-8} Per Year is related to the DOE repository safety strategy. The probability of igneous activity must be known to accurately estimate the long-term risk, as recognized in CRWMS M&O (2000c) for the proposed Yucca Mountain site. CRWMS M&O (2000c) identifies the probability of igneous intrusion as one of the eight principal factors for the Yucca Mountain repository system. The occurrence of seismic activity or faulting could result in failure of the waste package or drip shield. Performance of the waste package and performance of the drip shield/drift invert system are also identified as principal factors for the Yucca Mountain repository system (CRWMS M&O, 2000c).

The Identification of Events with Probabilities Greater Than 10^{-8} Per Year is important because this identification determines which events are needed to be considered further in the performance assessment. 10 CFR 63.114(d) requires that the performance assessment for Yucca Mountain must consider all events with at least 1 chance in 10,000 of occurring during the 10,000-year compliance period for the repository, which corresponds to an annual probability of 10^{-8} per year for events that have probabilities of occurrence that are independent of time. Events that are less likely than this do not need to be considered in the performance assessment. Events that are at least this likely must either be modeled within the performance assessment or be shown to not significantly affect the magnitude and time of the resulting radiological exposures to the reasonably maximally exposed individual or radionuclide releases to the accessible environment.

Additionally, Identification of Events with Probabilities Greater Than 10^{-8} Per Year is important for appropriately comparing the consequences of disruptive events against the 0.15-mSv/yr [15-mrem/yr] all-pathways dose standard in 10 CFR Part 63. 10 CFR 63.2 indicates in the definition of performance assessment that estimates of dose from all significant events and processes should be weighted by their probability of occurrence when included in the calculation of dose to the reasonably maximally exposed individual. Therefore, the probability of occurrence of a disruptive event is an important factor in the determination of whether the repository system will meet the limits specified in 10 CFR Part 63.

3.2.2.4 Technical Basis

NRC developed a plan (2002) consistent with the acceptance criteria and review methods found in previous issue resolution status reports. A review of DOE approaches for Identification of Events with Probabilities Greater Than 10^{-8} Per Year is provided in the following subsections. The review will be divided into four subsections: Igneous Activity, Seismicity, Faulting, and Criticality. Each subsection is organized according to the acceptance criteria in the Yucca Mountain Review Plan: (i) Events Are Adequately Defined, (ii) Probability Estimates for Future Events Are Supported by Appropriate Technical Basis , (iii) Probability Model Support Is Adequate, (iv) Probability Model Parameters Have Been Adequately Established, and (v) Uncertainty in Event Probability Is Adequately Evaluated.

3.2.2.4.1 Igneous Activity

The probability of igneous activity affecting the repository system was discussed and reached closed-pending status at a technical exchange held in August 2000.[1] NRC expects to receive all information required to complete the agreements by fiscal year 2003.

3.2.2.4.1.1 Events Are Adequately Defined

Overall, the current information is sufficient to conclude that the necessary information will be available to assess the probability of igneous activity affecting the repository system at the time of a potential license application.

Repository performance considerations require that the probability of volcanic disruption is calculated discretely from the probability of intrusive disruption because the effects on repository performance are significantly different for extrusive and intrusive processes. A volcanic igneous event that penetrates the repository has the potential to entrain, fragment, and transport radioactive material into the accessible environment. In contrast, an intrusive igneous event that penetrates the repository would produce thermal, mechanical, and chemical loads on engineered systems, which could affect waste-package degradation. Radioactive release associated with intrusive igneous events is through hydrologic flow and transport rather than through direct transport by volcanic processes. Therefore, probability calculations need to

[1]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (August 29–31, 2000)." Letter (October 23) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

distinguish between volcanic and intrusive igneous events to appropriately determine the contribution of each event to the probability weighted dose.

DOE documented the approach and technical basis for the definition of an igneous event in CRWMS M&O (2000a) and supporting analysis and model reports. CRWMS M&O (2000f) summarizes the technical basis for the definition of an igneous event. DOE estimate of the probability of an igneous event affecting the repository is based on the results of an expert elicitation to determine the probability of igneous activity at Yucca Mountain (CRWMS M&O, 1996). DOE defined a volcanic event as a point in space representing a volcano and an associated intrusive dike having length, azimuth, and location extending from the point event (CRWMS M&O, 2000f). Although the probabilistic volcanic hazard assessment assumed volcanic events to have both an extrusive (eruptive volcano) and intrusive component (dike), the output of the probabilistic volcanic hazard assessment was the annual frequency of intersection of the repository by only an intrusive basaltic dike. The probability of a volcanic eruption, conditional on dike intersection through the repository, likely would be lower using the probabilistic volcanic hazard assessment methodology. The DOE probabilistic volcanic hazard assessment did not calculate the conditional probability that a dike intersecting the repository footprint would result in an extrusive volcanic eruption through the repository. Models for the distribution of vents along a dike (based on the DOE probabilistic volcanic hazard assessment expert output and some observed vent spacings in the Yucca Mountain region) indicate that the probabilistic volcanic hazard assessment-derived eruption probability is always less than the dike intersection probability by a factor of approximately two (CRWMS M&O, 2000f).

The distinction between intrusive and extrusive igneous events is sufficiently clear in the DOE documentation to allow NRC to have enough information at the time of licensing to make a regulatory decision in this area.

3.2.2.4.1.2 Probability Estimates for Future Events Are Supported by Appropriate Technical Basis

Overall, the current information, along with agreements reached between DOE and NRC, is sufficient to conclude that the necessary information will be available to assess the probability of igneous activity affecting the repository system at the time of a potential license application.

Previous studies of volcanism in the Yucca Mountain region and elsewhere cumulatively indicate that models describing the recurrence rate or probability of basaltic volcanism should reflect the clustered nature of basaltic volcanism and shifts in the locus of basaltic volcanism through time. Models also should be amenable to comparison with basic geological data, such as fault patterns and neotectonic stress information, that affect vent distributions on a comparatively more detailed scale. The models used to estimate future igneous activity in the Yucca Mountain region should either explicitly account for the following or obtain bounding estimates:

- Shifts in the locus of volcanic activity through time
- Vent clusters
- Vent alignments and correlation of vents and faults

Data from other basaltic volcanic fields may be used to test the models. The nature of these spatial patterns in the Yucca Mountain region and how these compare with spatial patterns in cinder cone volcanism observed in other basaltic volcanic fields are reviewed in this section.

DOE documented the approach and technical basis for calculating the probability of an igneous event affecting the repository system in CRWMS M&O (2000a) and supporting analysis and model reports. The analysis and model report (CRWMS M&O, 2000f) summarizes the technical basis for the estimate of the probability of igneous activity affecting the Yucca Mountain repository. The results of the probabilistic volcanic hazard assessment (CRWMS M&O, 1996) form the basis of the DOE estimate of the probability of igneous activity affecting the repository system. For the probabilistic volcanic hazard assessment, an expert panel was convened in 1995 to review pertinent data relating to volcanism at Yucca Mountain and, based on these data, to quantify both the annual probability and associated uncertainty of an intrusive volcanic event intersecting a potential repository at Yucca Mountain. The experts reviewed two decades of data collected by volcanologists who conducted studies to quantify the probability that a future volcanic eruption would disrupt the potential repository. The mean intersection probability based on the results of the probabilistic volcanic hazard assessment was slightly greater than 10^{-8} per year (CRWMS M&O 2000f).

Agreement exists between the models and observed data on the basic patterns of basaltic volcanism in the Yucca Mountain region. These patterns include changes in the locus of volcanism with time, recurring volcanic activity within vent clusters, formation of vent alignments, and structural controls on the locations of volcanoes. Each of these patterns in vent distribution has an important impact on volcanic probability models and is considered in many probability models.

All current probability estimates for future igneous activity at the proposed repository site are based on past patterns of igneous activity in the Yucca Mountain region. Some parameter values or ranges used in these probability models, however, are dependent on definitions of the spatial or temporal extent of the Yucca Mountain region igneous system. Ongoing work suggests Crater Flat Basin basalts since about 12 million years may have a common petrogenesis, whereas 7–12-million years Yucca Mountain region basalt petrogenesis may be strongly influenced by silicic caldera-forming processes. Thus, Miocene basalt in the Crater Flat basin provides relevant information for risk assessments not included in current DOE models. Additionally, there are concerns about how the probabilistic volcanic hazard assessment was conducted. DOE selected only a limited range of experts for the probabilistic volcanic hazard assessment, using an internal nomination rather than a self-selection process. Potential biases or conflicts of interest among the experts are not documented. Modifications to initial elicitation reports also are not documented. These items do not follow the guidance in NUREG–1563 (NRC, 1996) for conducting an expert elicitation, and, therefore, make it difficult to evaluate the conclusions of the probabilistic volcanic hazard assessment elicitation (CRWMS M&O, 1996). Therefore, there is concern that the DOE probability model could result in an inaccurate estimate of the probability of igneous activity affecting the repository system. NRC staff independent assessments of the probability of igneous activity affecting the Yucca Mountain repository estimate it to be approximately 10^{-7} per year for both extrusive and intrusive volcanism (Hill and Connor, 2000). Therefore, DOE agreed[2] to include, in the Total System Performance Assessment–Site Recommendation and any license application, the results of a single-point sensitivity analysis for extrusive and intrusive igneous processes affecting the repository system at a probability of 10^{-7} per year. The NRC staff will consider this sensitivity analysis in its review.

3.2.2.4.1.3 Probability Model Support Is Adequate

Overall, the current information, along with agreements between DOE and NRC, is sufficient to conclude that the necessary information will be available to assess the probability of igneous activity affecting the repository system at the time of a potential license application.

DOE documented the support for the models predicting the probability of an igneous event affecting the repository system in CRWMS M&O (2000a) and supporting analysis and model reports. The CRWMS M&O (2000f) analysis and model report summarizes the technical basis for the estimate of the probability of igneous activity affecting the Yucca Mountain repository. The results of the probabilistic volcanic hazard assessment (CRWMS M&O, 1996) form the basis of the DOE estimate of the probability of igneous activity affecting the repository system. The conceptual model of volcanism, including how and where magmas form and what processes control the timing and location of magma ascent through the crust to form volcanoes, has a fundamental impact on how probability models are formulated and the consequent results of probability models. The probabilistic volcanic hazard assessment experts distinguished between deep (mantle source) and shallow (upper crustal structure and stress field) processes when considering different scales (regional and local) of spatial control on volcanism. Many probabilistic volcanic hazard assessment models restricted the areas of greatest likelihood for future volcanic activity to the areas where previous volcanism has occurred. DOE also justifies the probabilistic volcanic hazard assessment volcanic source-zone definitions by relating these zones to areas within the crater flat basin that have undergone the greatest amount of shallow crustal extension (e.g., Fridrich, et al., 1999, Figure 5; CRWMS M&O, 2000f, Figures 9a and 9b).

Although some volcanic source zones in CRWMS M&O (1996, 2000f) are supported by tectonic models, many other zones and other tectonic models are not supported. Few tectonic models or data are cited in CRWMS M&O (1996) for zone definitions. Currently available geophysical data (gravity, aeromagnetic, and seismic) do not support zone definitions used in the probabilistic volcanic hazard assessment (CRWMS M&O, 1996, 2000f). DOE does not seem to have established the validity of the probabilistic volcanic hazard assessment source-zone modeling approach. Additionally, there is an inconsistency between the probabilistic volcanic hazard assessment and the current DOE probability models. Probabilistic volcanic hazard assessment volcanic source zones clearly were defined on timing and location of past volcanism within the source zone. A new event center (i.e., volcano) forms only in the source zone, with only a subsurface intrusion potentially extending out of the zone and intersecting the

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[2]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

repository. The model in CRWMS M&O (2000f), however, has new volcanoes forming randomly along the intrusion, often outside the predefined volcanic source zone. By probabilistic volcanic hazard assessment definition, new volcanoes should occur only within the source zone at recurrences defined by past patterns of volcanic activity within that zone. If volcanoes can form outside the source zone as indicated in CRWMS M&O (2000f), the source zones must be expanded to encompass the location of future volcanism. The frequency of dike intersections would then increase using the expanded zones, as shorter, more abundant dikes would intersect the proposed repository location. DOE needs to demonstrate that its preferred approach can reasonably forecast the timing and location of future igneous events (cf., Condit and Connor, 1996). Therefore, there is concern that the probability model used could result in an inaccurate estimate of the probability of igneous activity affecting the repository system. NRC staff independent assessments of the probability of igneous activity affecting the Yucca Mountain repository estimate it to be approximately 10^{-7} per year for both extrusive and intrusive volcanism (Hill and Connor, 2000). Therefore, DOE agreed[3] to include, in the Total System Performance Assessment–Site Recommendation and any license application, the results of a single-point sensitivity analysis for extrusive and intrusive igneous processes affecting the repository system at a probability of 10^{-7} per year. The NRC staff will consider this sensitivity analysis in its review.

3.2.2.4.1.4 Probability Model Parameters Have Been Adequately Established

Overall, the current information, along with agreements reached between DOE and NRC, is sufficient to conclude that the necessary information will be available to assess the probability of igneous activity affecting the repository system at the time of a potential license application.

DOE documented the technical basis for the parameters supporting the models that predict the probability of an igneous event affecting the repository system in CRWMS M&O (2000a) and supporting analysis and model reports. The analysis and model report in CRWMS M&O (2000f) summarizes the technical basis for the probability model parameters. The results of the probabilistic volcanic hazard assessment (CRWMS M&O, 1996) form the basis of the DOE estimate of the probability of igneous activity affecting the repository system.

NRC staff have concerns about the selective use of data from the probabilistic volcanic hazard assessment (CRWMS M&O, 1996) that occurs in CRWMS M&O (2000f). For example, vent spacing (CRWMS M&O, 2000f, Section 6.5.2.2) only uses data from the 1-million years Crater Flat and 0.3-million years Sleeping Butte volcanoes, but ignores relevant information from the 3.7-million years Crater Flat, buried anomalies in Amargosa Desert, Paiute Ridge Intrusive Complex, and other features used by DOE to support igneous process models for the Yucca Mountain region. There also is an assumption that a relationship exists in the probabilistic volcanic hazard assessment (CRWMS M&O, 1996) between the number of events and the number of dikes. The probabilistic volcanic hazard assessment (CRWMS M&O, 1996) considered these as independent parameters. Thus, there is concern that the parameters used

[3]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

in the probability model could result in an inaccurate estimate of the probability of igneous activity affecting the repository system. NRC staff independent assessments of the probability of igneous activity affecting the Yucca Mountain repository estimate it to be approximately 10^{-7} per year for both extrusive and intrusive volcanism (Hill and Connor, 2000). Therefore, DOE agreed[4] to include, in the Total System Performance Assessment–Site Recommendation and any license application, the results of a single-point sensitivity analysis for extrusive and intrusive igneous processes affecting the repository system at a probability of 10^{-7} per year. The NRC staff will consider this sensitivity analysis in its review.

3.2.2.4.1.5 Uncertainty in Event Probability Is Adequately Evaluated

Overall, the current information, along with agreements reached between DOE and NRC, is sufficient to conclude that the necessary information will be available to assess the probability of igneous activity affecting the repository system at the time of a potential license application.

DOE documented the technical basis for the uncertainty in the probability of an igneous event affecting the repository system in CRWMS M&O (2000a) and supporting analysis and model reports. CRWMS M&O (2000f) summarizes the technical basis for the uncertainty in the estimate of the probability of igneous activity affecting the Yucca Mountain repository. The results of the probabilistic volcanic hazard assessment (CRWMS M&O, 1996) form the basis of the DOE estimate of the probability of igneous activity affecting the repository system. There are no generally accepted methodologies for calculating the probabilities of future igneous activity in distributed volcanic fields for periods of 10,000 years. In addition, more than one conceptual model can be applied to this problem, resulting in a wide range of probability values. DOE is using expert elicitation (CRWMS M&O, 1996) to construct a range of probability models, estimate uncertainties in model results caused by reasonable variations in model parameters, and calculate a probability distribution for use in performance assessment models.

The use of an expert elicitation conducted following NRC guidance in NUREG–1563 (NRC, 1996) is an acceptable methodology to determine the uncertainty in the probability of an igneous event. NRC staff have some concerns about how the DOE expert elicitation was conducted and documented, as discussed in Section 3.2.2.4.1.2. Additionally, NRC has concerns that uncertainty in the probability of igneous activity caused by undetected igneous events in the Yucca Mountain region could significantly affect the DOE calculation of the probability of igneous activity affecting the repository system. Therefore, DOE agreed[5] to evaluate new aeromagnetic data for potential buried igneous features and the effect on the probability estimate.

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[4]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[5]Ibid.

3.2.2.4.2 Faulting

The probability of a faulting event affecting the repository system was discussed at a Technical Exchange held in October 2000.[6] The Structural Deformation and Seismicity Subissue 1, Faulting, reached closed-pending status at this technical exchange. NRC expects to receive all information required to complete the agreements by fiscal year 2003.

3.2.2.4.2.1 Events Are Adequately Defined

Overall, the current information is sufficient to conclude that the necessary information will be available to assess the probability of faulting affecting the repository system at the time of a potential license application.

The approach and technical basis for defining faulting events are contained in CRWMS M&O (2000a). DOE divides faulting events into separate features, events, and processes based on their potential consequence. DOE considers that faulting events could potentially alter groundwater flow around and below the drift or could potentially disrupt engineered barriers in the repository system. When considering the effects of faulting on groundwater flow, DOE defined an event as a fault displacement event that could either change fracture properties throughout the unsaturated zone flow model domain or change the fracture properties specifically within fault zones. These two end-member cases relate to the mechanical strain either distributed throughout the strata bounded by the faults or localized to the individual fault zones. When considering the effects of faulting on engineered barriers, DOE defined an event as the failure of a structure, system, or component to perform its functional goal because of fault displacement loading. DOE analyses consider the reactivation of existing faults and the formation of new faults as separate types of events with different probabilities and consequences.

The definition of events is sufficiently clear in the DOE documentation to allow NRC to have enough information at the time of licensing to make a regulatory decision in this area.

3.2.2.4.2.2 Probability Estimates for Future Events Are Supported by Appropriate Technical Basis

Overall, the current information, along with agreements reached between DOE and NRC, is sufficient to conclude that the necessary information will be available to assess the probability of faulting affecting the repository system at the time of a potential license application.

The approach and technical basis for defining the probability of faulting affecting the repository system are contained in CRWMS M&O (2000a) and the analysis and model reports in CRWMS M&O (2000e,g,h,i). The basis for the estimates of the probability of faulting events

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

affecting the repository system is the result of an expert elicitation documented in the U.S. Geological Survey (1998). The probabilistic seismic hazard assessment used data collected on faulting characteristics at Yucca Mountain and in the Basin and Range province during past earthquakes to develop a displacement hazard curve. Principal and secondary (or distributed) faulting were considered. Principal faulting refers to displacement along the main fault zone responsible for the release of seismic energy (i.e., an earthquake) (dePolo, et al., 1991). At Yucca Mountain, principal faulting is assumed to occur only along principal faults, mainly block-bounding faults like the Solitario Canyon and Paintbrush Canyon faults. In contrast, secondary or distributed faulting is defined as rupture of smaller faults, such as the Ghost Dance fault, that occurs in response to the rupture in the vicinity of the principal fault (dePolo, et al., 1991). These two subsets of faults are not mutually exclusive. Faults capable of principal rupture can also undergo secondary faulting in response to faulting on another principal fault. Because principal and secondary faults pose a potential risk to repository performance, DOE considered both types. NRC (1999) provides a review of the methodology used by the DOE expert elicitation to develop an appropriate probabilistic fault displacement hazard assessment. This curve plots the frequency of exceeding a fault displacement value. The probabilistic seismic hazard assessment concluded that mean displacements at all locations within the repository system, except for Bow Ridge and Solitario Canyon faults, are 0.1 cm [30.039 in.] or less at the 10^{-5} annual exceedance probability. The mean displacements for the Bow Ridge and Solitario Canyon faults are 8 and 32 cm [3.15 and 12.6 in], respectively, at the 10^{-5} exceedance probability. DOE extrapolated these results and used the median value predicted by the experts to provide estimates of the displacement at the 10^{-8} annual exceedance probability.

DOE concluded faulting affecting groundwater flow is credible because the fault displacement could change the properties of the fractures in the unsaturated zone rock. DOE has developed criteria for fault setback distances for the design of the repository, which will be applied to existing faults with known or suspected Quaternary-age displacements. This setback distance is designed to mitigate the shear stresses induced on the waste packages and drip shields. The probabilistic seismic hazard assessment concluded that the mean displacement at a 10^{-8} annual exceedance probability for small faults and shear fractures in the repository system is less than 1 m [39.4 in.]. This displacement roughly corresponds to the maximum measured Quaternary per-event displacement on the Solitario Canyon fault. Based on the gap between the drip shields and the drift walls, DOE concluded this displacement could not cause the failure of the waste package nor the drip shield. The probabilistic seismic hazard assessment also concluded that the mean annual probability of a shear fracture developing in intact rock is less than 10^{-8}. Therefore, DOE concluded that all aspects of faulting could be screened based on low probability except for the effects of faulting on groundwater flow.

Staff reviewed the data, conceptual models, and assumptions developed by DOE in the probabilistic seismic hazard assessment (U.S. Geological Survey, 1998) and found that DOE adequately evaluated the nature and amount of faulting and the appropriate range of both principal and secondary faulting hazard sources within the repository block. In addition, DOE adequately determined fault geometry applicable to development of the probabilistic fault displacement hazard assessment. Given present knowledge, the DOE interpretations of faulting from surficial and underground mapping, as presented in the DOE probabilistic seismic hazard assessment (U.S. Geological Survey, 1998), are geologically consistent and reasonable. The experts adequately noted faults as primary or secondary, because these classifications pertain to the probabilistic fault displacement hazard assessment. Faulting characteristics identified subsequently or for which new data are developed should be evaluated or reevaluated, respectively. Variation of fault orientation data is within acceptable limits for normal geologic work. Staff disagree, however, with the statistic used to combine the fault displacement hazard curves from the different experts in the probabilistic seismic hazard assessment. DOE uses the median value of the curves of the experts as the statistic of interest, whereas NRC staff believe that the mean is the more appropriate measure. Using the mean value of the curves would lead to a larger displacement being predicted at the 10^{-8} annual probability level. DOE agreed[7] to provide technical justification for use of median values or another statistical measure, such as the mean, or evaluate and implement an alternative approach.

3.2.2.4.2.3 Probability Model Support Is Adequate

Overall, the current information is sufficient to conclude that the necessary information will be available to assess the probability of faulting affecting the repository system at the time of a potential license application.

The support for the probability model is contained in the CRWMS M&O (2000a) and the analysis and model reports (CRWMS M&O, 2000e,h,i). The basis for the probability of faulting affecting the repository system is the result of probabilistic seismic hazard assessment. The experts in the probabilistic seismic hazard assessment appropriately considered primary and secondary faulting when defining fault displacement hazard curves. The level of ground motion predicted by the probabilistic seismic hazard assessment has been compared to tectonically and seismically active sites elsewhere in the Basin and Range Province (Wong and Olig, 1998) and found to be lower than other more seismically active areas in the Basin and Range province, such as along the Wasatch fault in north central Utah.

Staff review indicates that DOE adequately evaluated the nature and amount of faulting and the appropriate range of both principal and secondary faulting hazard sources within the repository block. In addition, DOE adequately determined fault geometry applicable to development of the probabilistic fault displacement hazard assessment. Given present knowledge, the DOE interpretations of faulting from surficial and underground mapping, as presented in U.S. Geological Survey (1998), are geologically consistent and reasonable.

3.2.2.4.2.4 Probability Model Parameters Have Been Adequately Established

Overall, the current information is sufficient to conclude that the necessary information will be available to assess the probability of faulting affecting the repository system at the time of a potential license application.

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

The technical basis for the parameters used in the probability model is contained in CRWMS M&O (2000a) and the (CRWMS M&O, 2000i) analysis and model report. The basis for the probability model is the result of the probabilistic seismic hazard assessment. The assessment of seismic hazards at Yucca Mountain in the probabilistic seismic hazard assessment relied on the results of scientific studies that characterized the tectonic activity in the region. These studies provided data and information on (i) the presence of faults within approximately 100 km [62 mi] of Yucca Mountain and if these faults had sustained Quaternary activity; (ii) the history and characteristics of past earthquakes, which were obtained from the results of detailed paleoseismic fault-trenching studies of active faults near Yucca Mountain; (iii) contemporary seismicity; (iv) historical and instrumentally recorded earthquakes in the Yucca Mountain region; (v) ground motion attenuation relationships for extensional tectonic regimes; (vi) local site attenuation characteristics; (vii) the tectonic stresses from hydrofracture measurements and earthquake focal mechanisms; (viii) geophysical data to assess tectonic models and identify subsurface faults; and (ix) geodetic data to measure ongoing crustal deformation.

Staff review indicates DOE adequately evaluated the nature and amount of faulting and the appropriate range of both principal and secondary faulting hazard sources within the repository block. In addition, DOE adequately determined fault geometry applicable to development of the probabilistic fault displacement hazard assessment. Given present knowledge, the DOE interpretations of faulting from surficial and underground mapping, as presented in U.S. Geological Survey (1998), are geologically consistent and reasonable. The experts adequately noted faults as primary or secondary for the purpose of the probabilistic fault displacement hazard assessment. The fault displacement hazard assessment must be reevaluated, however, if new faulting characteristics or data are identified. Some fault data taken by DOE from surface outcrops and from the exploratory studies facilities have been confirmed by independent checks by the NRC staff (NRC, 1999b). The variation of fault orientation data is within acceptable limits for normal geologic work. Field checks of fault locations, orientations, displacements, and other selected geometric features are generally in close agreement with the DOE observations and interpretations.

3.2.2.4.2.5 Uncertainty in Event Probability Is Adequately Evaluated

Overall, the current information, along with agreements reached between DOE and NRC, is sufficient to conclude that the necessary information will be available to assess the probability of faulting affecting the repository system at the time of a potential license application.

The technical basis for the estimate of uncertainty in the probability model is contained in CRWMS M&O (2000a) and the CRWMS M&O (2000i) analysis and model report. The uncertainty in the event probability is obtained from the results of the probabilistic seismic hazard assessment. Uncertainty in the estimate of the probability of a faulting event is based on the range of results in the probabilistic fault displacement hazard assessment from the different experts. DOE incorporates the uncertainty in the probability of the event by using the median value from the range of expert predictions for low probability ($<10^{-6}$ per year) fault displacements.

Staff disagree with the statistic used to combine the fault displacement hazard curves from the different experts in the probabilistic seismic hazard assessment. DOE uses the median value of the curves of the experts as the statistic of interest, whereas NRC staff believe that the mean is the more appropriate measure. Using the mean value of the curves would lead to a larger displacement being predicted at the 10^{-8} annual probability level. DOE agreed[8] to provide technical justification for use of median values or another statistical measure, such as the mean, or will evaluate and implement an alternative approach.

3.2.2.4.3 Seismicity

The probability of a seismic event affecting the repository system was discussed and reached closed-pending status at a technical exchange held in October 2000.[9] All information required to complete the agreements is expected to be received by the NRC by fiscal year 2003.

3.2.2.4.3.1 Events Are Adequately Defined

Overall, the current information is sufficient to conclude that the necessary information will be available to assess the probability of seismicity affecting the repository system at the time of a potential license application.

The approach and technical basis for defining seismic events are contained in CRWMS M&O (2000a). DOE indicates that small magnitude seismic events will be common at the Yucca Mountain repository whereas larger, more damaging seismic events will be less likely. Seismic events have the potential to affect performance through any of three effects: (i) rockfall causing direct damage to engineered barriers, (ii) failure of cladding, or (iii) changes to the groundwater flow system. These effects depend on the magnitude of the seismic event, so DOE defined a hazard curve in the probabilistic seismic hazard assessment (U.S. Geological Survey, 1998) that describes the probability of exceeding an earthquake of a given magnitude. A detailed review of the seismic aspects of the probabilistic seismic hazard assessment, including staff concerns and related agreements, is contained in Section 3.3.2 of this issue resolution status report.

The definition of events is sufficiently clear in the DOE documentation to allow NRC to have enough information at the time of licensing to make a regulatory decision in this area.

3.2.2.4.3.2 Probability Estimates for Future Events Are Supported by Appropriate Technical Basis

Overall, the current information, along with agreements reached between DOE and NRC, is sufficient to conclude that the necessary information will be available to assess the probability of seismic activity affecting the repository system at the time of a potential license application.

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

[9]Ibid.

Repository Safety After Permanent Closure

The approach and technical basis for defining the probability of seismicity affecting the repository system are contained in CRWMS M&O (2000a,e,i). DOE concluded that seismicity at Yucca Mountain is likely but that the magnitude of the event is an inverse function of the probability. The basis for the estimate of the probability of seismic events exceeding a given magnitude is the result of an expert elicitation documented in the U.S. Geological Survey (1998). A detailed review of the seismic aspects of the probabilistic seismic hazard assessment related to damage to cladding, including staff concerns and related agreements, is contained in Section 3.3.1 of this issue resolution status report. A detailed review of the seismic aspects of the probabilistic seismic hazard assessment related to rockfall and drift collapse, including staff concerns and related agreements, is contained in Section 3.3.2 of this issue resolution status report.

NRC staff have not identified any additional concerns beyond those identified in Sections 3.3.1 and 3.3.2 of this issue resolution status report.

3.2.2.4.3.3 Probability Model Support Is Adequate

Overall, the current information, along with agreements reached between DOE and NRC, is sufficient to conclude that the necessary information will be available to assess the probability of seismic activity affecting the repository system at the time of a potential license application.

The support for the probability model for seismicity affecting the repository system is contained in the CRWMS M&O (2000a,e,i). DOE concluded that seismicity at Yucca Mountain is likely but that the magnitude of the event is an inverse function of the probability. The basis for the estimate of the probability of seismic events exceeding a given magnitude is the result of an expert elicitation documented in the U.S. Geological Survey (1998). A detailed review of the seismic aspects of the probabilistic seismic hazard assessment related to damage to cladding, including staff concerns and related agreements, is contained in Section 3.3.1 of this issue resolution status report. A detailed review of the seismic aspects of the probabilistic seismic hazard assessment related to rockfall and drift collapse, including staff concerns and related agreements, is contained in Section 3.3.2 of this issue resolution status report.

NRC staff have not identified any additional concerns beyond those identified in Sections 3.3.1 and 3.3.2 of this issue resolution status report.

3.2.2.4.3.4 Probability Model Parameters Have Been Adequately Established

Overall, the current information, along with agreements reached between DOE and NRC, is sufficient to conclude that the necessary information will be available to assess the probability of seismic activity affecting the repository system at the time of a potential license application.

The approach and technical basis for defining the parameters for the probability model for seismicity affecting the repository system are contained in CRWMS M&O (2000a,e,i). DOE concluded that seismicity at Yucca Mountain is likely but that the magnitude of the event is an inverse function of the probability. The basis for the estimate of the probability of seismic events exceeding a given magnitude is the result of an expert elicitation documented in the U.S. Geological Survey (1998). A detailed review of the seismic aspects of the probabilistic seismic hazard assessment related to damage to cladding, including staff concerns and related agreements, is contained in Section 3.3.1 of this issue resolution status report. A detailed review of the seismic aspects of the probabilistic seismic hazard assessment related to rockfall and drift collapse, including staff concerns and related agreements, is contained in Section 3.3.2 of this issue resolution status report.

NRC staff have not identified any additional concerns beyond those identified in Sections 3.3.1 and 3.3.2 of this issue resolution status report.

3.2.2.4.3.5 Uncertainty in Event Probability Is Adequately Evaluated

Overall, the current information, along with agreements reached between DOE and NRC, is sufficient to conclude that the necessary information will be available to assess the probability of seismic activity affecting the repository system at the time of a potential license application.

The approach and technical basis for determining the uncertainty in the probability of seismicity affecting the repository system are contained in the CRWMS M&O (2000a,e,i). DOE concluded that seismicity at Yucca Mountain is likely but that the magnitude of the event is an inverse function of the probability. The basis for the estimate of the probability of seismic events exceeding a given magnitude is the result of an expert elicitation documented in the U.S. Geological Survey (1998). A detailed review of the seismic aspects of the probabilistic seismic hazard assessment related to damage to cladding, including staff concerns and related agreements, is contained in Section 3.3.1 of this issue resolution status report. A detailed review of the seismic aspects of the probabilistic seismic hazard assessment related to rockfall and drift collapse, including staff concerns and related agreements, is contained in Section 3.3.2 of this issue resolution status report.

NRC staff have not identified any additional concerns beyond those identified in Sections 3.3.1 and 3.3.2 of this issue resolution status report.

3.2.2.4.4 Nuclear Criticality

The probability of a criticality event affecting the repository system was discussed and reached closed-pending status at a technical exchange held in October 2000.[10] NRC expects to receive all information required to complete the agreements by fiscal year 2003 or before the submission of any license application for a repository at Yucca Mountain.

3.2.2.4.4.1 Events Are Adequately Defined

Overall, the current information is sufficient to conclude that the necessary information will be available to assess the probability of criticality in the repository system at the time of a potential license application.

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[10]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Criticality (October 23–24, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

The approach and technical basis for defining criticality events are contained in DOE (2000), and the calculation is in CRWMS M&O (2000b). DOE considers three major categories of criticality events: events, near-field events, and far-field events. The fuel can be in either intact or degraded condition for in-package events that occur within the waste package or near-field events that occur within the drift. Far-field events occur in the unsaturated zone or saturated zone below the repository and can only occur after the fuel degrades and releases fissile material.

NRC considers acceptable the division of criticality events based on the location of the event (e.g., in-package, near-field, and far-field).

3.2.2.4.4.2 Probability Estimates for Future Events Are Supported by Appropriate Technical Basis

Overall, the current information, along with agreements reached between DOE and NRC, is sufficient to conclude that the necessary information will be available to assess the probability of criticality in the repository system at the time of a potential license application.

The approach and technical basis for estimating the probability of criticality events are contained in DOE (2000), and the calculation is in CRWMS M&O (2000b). The probability of criticality in 10,000-year calculations does not follow the methodology outlined in the Topical Report on Disposal Criticality. Instead, it attempts to perform a simplified analysis to demonstrate that criticality events can be screened from the Total System Performance Assessment. The screening argument in this document for criticality is based on the low probability of a waste package failing within the first 10,000 years except through igneous events. Criticality in the waste package or the near field after an igneous event can be screened on the basis of low probability of forming a critical configuration after the event (CRWMS M&O, 2000b). The probability of a waste package failing before 10,000 years is stated to be 2.7×10^{-11}/waste package (CRWMS M&O, 2000b) based on results in the analysis and model report (CRWMS M&O, 2000j). This value, however, is based only on the probability of early waste package failure because of welding flaws. Other mechanisms for waste package failure are analyzed in this analysis and model report, including failures caused by flaws in the base metal, use of improper weld material, improper heat treatment of the welds, and damage incurred during handling operations. The occurrence of these failure mechanisms is much more likely than failures caused by flaws in the welds [a total of about 5.5×10^{-5} waste package (CRWMS M&O, 2000j)]. Additionally, this value of 2.7×10^{-11} was based on a value of 11.5 mm [0.45 in.] for the depth at which the stress in the waste package goes from compressive to tensile. However, this value is identified as being used only for an example to demonstrate the models rather than defensible data. Therefore, this value should not be used to screen events from the Total System Performance Assessment. NRC staff review of the analysis and model report (CRWMS M&O, 2000j) also identified several concerns. First, failure rates used in the calculations averaged failure data throughout a long history that allowed for improvements in fabrication techniques. These data may not be appropriate for the waste package, which will be manufactured using a new fabrication process and may not be able to benefit from the identification of improvements in the fabrication process as failures are identified. Second, the welding and heat treatment of the outer lids are remote operations (Bechtel SAIC Company, LLC, 2001), so the sequence of operations may not include a final laboratory check. This laboratory check was relied on when developing the probability of failure because of an improper heat treatment, and the probability of failure of a waste package would increase substantially without it. Third, the probability of handling damage did not include the possibility that an uninspected, damaged disposal container arriving from the fabricator remains undetected during arrival inspections at the repository. Additionally, a screening argument for criticality after igneous-induced waste package failure has only been provided for commercial spent nuclear fuel, not for DOE spent nuclear fuel or defense high-level waste. Therefore, the probability estimates that are used as the basis of the screening argument are not sufficient to support the screening of criticality from the performance assessment.

DOE submitted a topical report (DOE, 2000) that describes the methodology that will be used to determine the probability and consequences of a criticality event at the Yucca Mountain repository. This methodology provides a detailed analysis of possible locations within the repository system where a criticality event may occur. Using a probabilistic methodology, the criticality analysis will perform a detailed tracking of the fissile and neutron poison materials during the degradation of the waste form and waste package structural materials to determine the probability of a critical configuration being generated. NRC reviewed the initial revision of DOE (1998) and issued a safety evaluation report documenting the results of the staff review of the document (NRC, 2000d). This safety evaluation report contained 28 Open Items, which are areas of concern that NRC staff have about the methodology. DOE indicated that Revision 1 of the topical report has addressed 27 of the Open Items, and the resolution of the other Open Item, related to the verification of burnup of the spent nuclear fuel, is the subject of Agreement PRE.07.01. Additionally, a recent document DOE released attempts to screen criticality using a simple fault tree to determine the probability of criticality in the repository system. Both documents are currently being reviewed by NRC staff. Therefore, although DOE has not provided adequate justification for the screening of criticality from the repository system at this point, the information provided, along with the information required to be provided in the agreements,[11] will allow NRC staff to have sufficient information at the time of the license application to evaluate the DOE safety case.

3.2.2.4.4.3 Probability Model Support Is Adequate

Overall, the current information, along with agreements reached between DOE and NRC, is sufficient to conclude that the necessary information will be available to assess the probability of criticality in the repository system at the time of a potential license application.

The description of the support for the probability model is contained in DOE (2000) CRWMS M&O (2000d). The models that will be used to calculate the probability of a criticality event occurring within the repository system will be controlled under the DOE Configuration Management system. The primary codes in DOE (2000) that will need to be validated include geochemistry codes, neutron transport codes, and the configuration generator code. Where possible, DOE will use the same geochemistry codes as those in other areas of the repository

[11]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Criticality (October 23–24, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

program, within their range of validation. For example, the in-package chemistry code used in the criticality analysis will be validated to support the spent nuclear fuel dissolution model in the repository program, and the validation will not be repeated for the criticality analysis. However, the criticality analysis may need to perform geochemistry calculations for materials and areas of the repository outside the range of validation performed for the repository system. DOE will have to perform additional software validation to support the use of these models in these situations. The validation of the geochemical codes will be performed by comparing the results from the code against analytical solutions and against results obtained from other geochemistry-transport codes.

The neutron transport code will be validated by comparing the results of the code to data obtained from Commercial Reactor Critical experiments, radiochemical analyses, and Laboratory Critical Experiments. Any bias associated with the neutron transport code will be identified using these experiments and will be accounted for before comparing the calculated neutron multiplication factor to the critical limit. The configuration generator code will be validated by comparing the results of the code with appropriate hand calculations to demonstrate that it is implementing the model correctly.

Additionally, natural analog information will be used to gain insight in the behavior of radionuclides in the natural environment. For example, information from the natural reactors at Oklo, Gabon, Equatorial Africa, will provide insight on mechanisms of accumulation of fissile materials and transport of the resulting actinides and fission products away from the fissioning material. Additionally, information from the natural uranium deposit in Peña Blanca, Mexico, provides insights into the processes that lead to the accumulation and mobilization of uranium in unsaturated tuff.

The NRC staff review indicates that the proposed methodology of providing support for the probability calculation is appropriate. DOE agreed[12] to submit validation reports documenting the validation of the computer codes that will be used to calculate the probability of criticality within the repository system before the license application.

3.2.2.4.4.4 Probability Model Parameters Have Been Adequately Established

Overall, the current information, along with agreements reached between DOE and NRC, is sufficient to conclude that the necessary information will be available to assess the probability of criticality in the repository system at the time of a potential license application.

The approach for developing the technical bases for parameters used in the probability models is contained in DOE (1998, 2000), and the calculation is in CRWMS M&O (2000b). The parameters that will be used in calculating the probability of a criticality event occurring in the repository system will be derived from information developed and reviewed from other areas of the repository system. Important parameters in calculating the probability of criticality in the

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[12]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Criticality (October 23–24, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

repository system that will be justified in other areas of the repository program include the number of waste packages failed, parameters affecting the quantity of water entering the waste package (including the percolation rate and the seepage flow rate), water chemistry, the degradation rate of the fuel, and transport properties of the fissile materials (DOE, 1998). To support the use of these parameters, DOE will need only to demonstrate the parameters are consistent with the repository program and that there are no assumptions made in the selection of these parameter values that would be conservative with respect to nominal repository performance but nonconservative for the criticality calculation. Other parameters may be important in the calculation of the probability of criticality but not in other areas of the repository program, such as the degradation rate of basket support materials (DOE, 1998). DOE has agreed to providing proper justification for any parameter values for which sufficient justification has not been developed in other areas of the repository program. In general, DOE agreed[13] to provide an updated technical basis for screening criticality from the postclosure performance assessment.

The proposed methodology of using appropriate parameter values from other areas of the repository program in the criticality modeling is acceptable. Review of the justification of parameter values not defended in other areas of the repository program will be conducted when DOE provides the detailed calculations to determine the probability of criticality for all fuel types.

3.2.2.4.4.5 Uncertainty in Event Probability Is Adequately Evaluated

Overall, the current information is sufficient to conclude that the necessary information will be available to assess the probability of criticality in the repository system at the time of a potential license application.

The approach for calculating the uncertainty in the probability of criticality events is contained in DOE (2000), and the calculation is in CRWMS M&O (2000b). Using the topical report methodology, DOE will determine the probability of criticality by performing a Monte Carlo simulation that tracks the failure of the waste package, degradation of internal components of the waste package, and transport of fissile and poison materials through the repository system.

Parameters used in this model will be sampled from an uncertainty distribution to determine whether the system could go critical for a given parameter set. The estimate of the probability of criticality will be controlled by the uncertainty distributions used in the models. In the Monte Carlo process, an additional source of uncertainty is statistical uncertainty based on the number of realizations run. DOE indicated it will conduct sufficient realizations to ensure that this component of uncertainty is very small.

The methodology to estimate the probability of criticality in CRWMS M&O (2000b) is a deterministic calculation. These deterministic calculations rely on conclusions in other documents that the waste package will not fail within 10,000 years because of corrosion

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[13]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Criticality (October 23–24, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

processes (CRWMS M&O, 2000j) or seismic events (CRWMS M&O, 2000a). After an igneous event, these calculations use the mean values of distributions for water transport parameters and the fraction of waste packages capable of supporting a criticality event to demonstrate that the probability of a criticality event is a low-probability event.

The NRC staff review indicates that the proposed methodology in the Topical Report to include uncertainty in the estimate of the probability of a criticality event is appropriate.

3.2.2.5 Status and Path Forward

Table 3.2.2-1 provides related DOE and NRC agreements pertaining to the Identification of Events with Probability Greater Than 10^{-8} Per Year. The status and the detailed agreements (or path forward) pertaining to all the key technical issue subissues are provided in Table 1.1-3 and Appendix A. Additional agreements from the DOE and NRC Technical Exchange on August 6–10, 2001, are summarized in Appendix B.

The Total System Performance Assessment and Integration Key Technical Issue Subissue pertaining to the scenario analysis is considered closed-pending. Following is a summary of issues that DOE needs to resolve before this subissue can be closed.

Table 3.2.2-1. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreements*
Igneous Activity	Subissue 1—Probability of Igneous Activity	Closed-Pending	IA.1.01 IA.1.02
Structural Deformation and Seismicity	Subissue 1—Faulting	Closed-Pending	SDS.1.02
	Subissue 2—Seismicity	Closed-Pending	SDS.2.01 SDS.2.03
Container Life and Source Term	Subissue 5—The Effect of In-Package Criticality on Waste Package and Engineered Barrier Subsystem Performance	Closed-Pending	CLST.5.01 CLST.5.03 CLST.5.04
Evolution of the Near-Field Environment	Subissue 4—Coupled Thermal-Hydrologic-Chemical Processes on Radionuclide Transport Through Engineered and Natural Barriers	Closed-Pending	ENFE.5.01 ENFE.5.03
Radionuclide Transport	Subissue 4—Nuclear Criticality in the Far Field	Closed-Pending	RT.4.01 RT.4.03

Table 3.2.2-1. Related Key Technical Issue Subissues and Agreements (continued)			
Key Technical Issue	**Subissue**	**Status**	**Related Agreements***
Total System Performance Assessment and Integration	Subissue 2—Scenario Analysis and Event Probability	Closed-Pending	TSPAI.2.01 TSPAI.2.02 TSPAI.2.05 TSPAI.2.06 TSPAI.2.07
	Subissue 3—Model Abstraction	Closed-Pending	TSPAI.3.06
*Related DOE and NRC agreements are associated with one or all five generic acceptance criteria.			

3.2.2.6 References

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3.3 Model Abstraction

3.3.0 Model Abstraction—Generic Discussion

3.3.0.1 Description of Issue

When reviewing the DOE total system performance assessment, the NRC staff will evaluate elements (or model abstractions) of the repository system to determine how effective the overall system is at protecting the public health and safety. As discussed in Chapter 1, Introduction, there are 14 model abstraction sections the staff will use to determine compliance with 10 CFR 63.114 (see Figure 1.1-2 for a description of the model abstractions). These abstractions consider the aspects of the engineered, geosphere, and biosphere subsystems that may be important to performance. Important to performance means important to meeting the postclosure performance objectives specified at 10 CFR 63.113 and 63.311. The staff will use risk insights to focus their review on the important assumptions, models, and data in the total system performance assessment. The staff will also focus their review to ensure the degree of technical support for models and data abstractions is commensurate with its contribution to risk, which means the staff will review in greater detail those model abstractions and their important components on which DOE relies more heavily to prove its safety case.

The staff will also review the DOE total system performance assessment to decide if DOE properly characterized the features, events, and processes and properly incorporated them into the total system performance assessment. This review is necessary to decide if the DOE total system performance assessment is acceptable and complies with 10 CFR 63.114 and 63.115. The review methods and acceptance criteria the staff will use to evaluate compliance with the performance objectives (numerical standards) are in Section 4.2.1.4 of NRC (2002).

3.3.0.2 Relationship to Key Technical Issue Subissues

The following sections (3.3.1–3.3.14) discuss the 14 model abstractions. In each section, staff describes the relationship between the key technical issue subissues and the specific model abstraction being addressed.

The remainder of Section 3.3.0 discusses general issues and concerns associated with multiple model abstractions. These issues were identified as part of the staff review of the DOE site recommendation documents (CRWMS M&O, 2000a,b; DOE, 2001; Bechtel SAIC Company, LLC, 2001a,b) and various analysis and model reports (received through October 2001). The general issues the staff identified include

- Improvement needed in transparency and traceability of the model abstraction documentation

- Appropriately rigorous methodology not used for model abstraction simplifications and selections of parameter distributions, conceptual models, or modeling approaches

- Inadequate basis provided for the amount of information retained by the model abstractions

- Inadequate support for the process model results abstracted in the total system performance assessment and for the total system performance assessment

3.3.0.3 Importance to Postclosure Performance

A full and clear understanding of model abstractions is important to gain reasonable assurance in the estimated postclosure performance of the repository. The generic items discussed in this section (i.e., transparency and traceability of analyses, consistency of assumptions across various abstractions, and the verification of abstracted models through comparison with results from detailed process models) are applicable to all 14 abstractions discussed in Sections 3.3.1–3.3.14.

3.3.0.4 Technical Basis

Overall, the current information, along with the DOE and NRC agreements (Section 3.3.0.5), is sufficient to conclude the necessary information will be available, at the time of a potential license application, to allow NRC to conduct a detailed review.

A number of positive examples in the documentation are related to transparency and traceability. A positive example of transparency and traceability is seen in the DOE consideration and comparison of advective versus diffusive releases from the waste package. There are some areas, however, that need improvement. In particular, numerous examples exist where the discussion in a summary section or an individual abstraction section is inconsistent with other sections, the actual total system performance assessment model, or with the related analysis and model reports.[1] In particular, there are contradictory statements about the role of environmental variables in the corrosion models. In aggregate, the inconsistencies make it difficult for the reviewers to understand clearly some parts of the total system performance assessment model.

DOE agreed that transparency and traceability of documents will be improved and outlined its planned activities to improve the transparency and traceability:

- Update review procedures, with an emphasis on vertical slice reviews (e.g., by chapter and between documents to improve consistency)

- Improve or update the documents mentioned in the specific examples noted by NRC

- Complete a vertical slice review for consistency, which was under way at the time of the technical exchange

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[1]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

- Develop additional transparency tools, such as a flow chart of the total system performance assessment model, to further explain how data are passed between components and subcomponents of the overall Total System Performance Assessment–Site Recommendation model and the sources of these data and new graphics

- Allow time for additional reviews to include international peer review panels, internal review teams, and technical editors

To improve transparency and traceability, DOE also agreed to revisit the abstraction of colloid modeling and the use of the Waste Package Degradation Model in modeling the failure of the engineered barrier subsystem. NRC considered adequate the DOE general response addressing transparency and traceability, during the technical exchange of August 6–10, 2001.[2]

Based on a review of the Total System Performance Assessment–Site Recommendation and the supporting analysis and model reports, NRC staff consider the DOE methodology used for model abstraction simplifications and the selection of conservative parameter distributions, conceptual models, or modeling approaches needs additional rigor. In addition to integrating various abstractions into the total system performance assessment, DOE needs to use a consistent approach for conducting the total system performance assessment and making judgments regarding conservatism (i.e., leading to overestimating radiological consequences) and the treatment of uncertainty. For example, the system model or individual abstractions are sufficiently complex, which means human intuition cannot be relied on to make accurate decisions consistently. Specifically, it may be impossible to determine the effect of a parameter *a priori* for the complex, nonlinear models embedded in the total system performance assessment. Because of the interactions at the system level or among different parts of the system, intermediate parameter values may lead to larger doses to the reasonably maximally exposed individual than either bound of the distribution. For example, if ionic strength affected both colloid stability and cladding corrosion, it is possible that minimizing ionic strength to maximize colloid stability may not result in maximizing dose to the reasonably maximally exposed individual because it would also reduce the rate of cladding corrosion. A reduction in cladding corrosion corresponds to reduced releases of radionuclides and, consequently, a reduction in the transport of radionuclides in colloids and a reduction in the dose.

DOE agreed to improve this area and to develop written guidance in the model abstraction process for model developers so that: (i) the model abstraction process, (ii) the selection of conservatism in components, and (iii) the representation of uncertainty are systematic across the total system performance assessment model. These guidelines will address the evaluation of nonlinear models when conservatism is being used to address uncertainty and decisions are based on technical judgment in a complex system. DOE agreed the guidelines will be developed, implemented, and made available to NRC in fiscal year 2002. These proposed improvements represent an acceptable approach to address the NRC questions. In addition,

[2]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

the opportunity provided by availability of the guidance in fiscal year 2002 provides additional confidence that DOE will be able to implement these changes systematically in sufficient time to improve the total system performance assessment. Finally, if NRC has questions regarding the specific DOE approach, these questions can be communicated to DOE in a timely manner.

The abstraction process is typically a simplification of process model results into a form that represents an appropriate amount of uncertainty and variability, while allowing a computationally efficient solution. NRC recognizes that it is impossible to represent all of the spatial and temporal uncertainty and variability, as well as conceptual model uncertainty, in the overall total system performance assessment model. Staff have identified several instances, however, where DOE has not provided sufficient justification for the amount of information retained by the abstraction.[3] Specifically, DOE needs to justify the simplifications used with consideration of all affected subsystems or models. Two examples of inadequate technical bases for the simplification used in a model abstraction include (i) the DOE decision not to represent uncertainty in the infiltration map at each climate state and (ii) the DOE assumption that three seepage threshold levels adequately capture the contribution from the tails of the distribution.

DOE agreed to document the simplifications used for abstractions for all future total system performance assessments (TSPAI.3.39). DOE agreed to provide justification to show that the simplifications appropriately represent the necessary processes and appropriately propagate process model uncertainties. DOE also agreed to provide comparisons of output from process models to total system performance assessment abstractions. DOE indicated that the level of detail in the comparisons will be commensurate with any reduction in propagated uncertainty and the risk significance of the model. DOE stated that the documentation of the information will be provided in abstraction analysis and model reports in fiscal year 2003.

As part of the model development process, it is necessary to verify that the model is calculating properly, validate that an appropriate model has been developed for the problem being examined, and explain the detailed functioning of the model through complete analyses. DOE provided information on all three topics in CRWMS M&O (2000b). Several concerns were identified during the NRC staff review of the DOE Total System Performance Assessment–Site Recommendation model documentation. The following are examples of these concerns:

- Various errors were found in the DOE hand calculations.
- Abstracted models were used outside the ranges for which they were developed.
- It is not clear that DOE evaluated the significance of warnings and errors in the GoldSim (Golder Associates, 2000) error log file: neither the significance nor the evaluation of the warnings and errors were documented.

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[3]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

- DOE identified the elements of verification in CRWMS M&O (2000b) and supporting documents but has not rigorously verified the Total System Performance Assessment–Site Recommendation computer program.

- The limited set of random hand calculations did not represent a systematic approach to verification.

DOE issued Corrective Action Report No. BSC–01–C–001 dated May 3, 2001, that found "... the area of model validation is considered to be a significant condition adverse to quality." The corrective action report indicates that 18 of 24 analysis and model reports were inadequately validated, including 8 that were not validated at all. As the corrective action report indicates, the other methods deemed acceptable to develop support for process models were not satisfied.

DOE indicated that a root-cause analysis was being performed for Corrective Action Report No. BSC–01–C–001. DOE agreed to document the process used to develop confidence in the total system performance assessment models [e.g., steps similar to those described in NUREG–1636 (NRC, 1999)]. The detailed process is currently documented in the model development procedures being evaluated for process improvement in response to the model validation Corrective Action Report No. BSC–01–C–001. The upgraded model validation procedures will be available for NRC to review in fiscal year 2002. Additionally, DOE will document the implementation of the process for model confidence building and will demonstrate compliance with model confidence criteria in accordance with applicable procedures. This compliance will be documented in the respective analysis and model report revisions and made available to NRC in fiscal year 2003.

3.3.0.5 Status and Path Forward

Table 3.3.0-1 provides the DOE and NRC agreements pertaining to general issues and concerns associated with multiple model abstractions. Note that the status, and also the detailed agreements (or path forward) pertaining to all the key technical issue subissues, are provided in Table 1.1-3 and Appendix A. The DOE approach, together with the DOE agreements to provide NRC with additional information (through specified testing, analyses, and the like), acceptably addresses the NRC questions so that no information beyond that provided or agreed to will be required at the time of a potential license application. Sections 3.3.1 through 3.3.14 identify specific issues and concerns associated with each individual model abstraction.

Table 3.3.0-1. Related Key Technical Issue Subissues and Agreements			
Key Technical Issue	**Subissue**	**Status**	**Related Agreements***
Total System Performance Assessment and Integration	Subissue 3—Model Abstraction	Closed-Pending	TSPAI.3.38 TSPAI.3.39
	Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards	Closed-Pending	TSPAI.4.05 TSPAI.4.06
*Related DOE and NRC agreements are associated with one or all five generic acceptance criteria.			

3.3.0.6 References

Bechtel SAIC Company, LLC. "FY01 Supplemental Science and Performance Analyses." Vol. 1: Scientific Bases and Analyses. TDR–MGR–MD–000007. Revision 00 ICN 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2001a.

Bechtel SAIC Company, LLC. "FY01 Supplemental Science and Performance Analyses." Vol. 2: Performance Analyses. TDR–MGR–PA–000001. Revision 00 ICN 01. Las Vegas, Nevada: Bechtel SAIC Company, LLC. 2001b.

CRWMS M&O. "Total System Performance Assessment for the Site Recommendation." TRD–WIS–PA–000001. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000a.

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

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

3.3.1 Degradation of Engineered Barriers

3.3.1.1 Description of Issue

The Degradation of Engineered Barriers Integrated Subissue addresses the assessment of engineered barrier performance and waste package lifetimes. Engineered barriers include, in addition to the waste package, other components of the engineered barrier subsystem such as drip shield, drift invert, and backfill if any. In the proposed DOE site recommendation reference design for the various types of spent nuclear fuel and high-level waste glass, the waste package is composed (in addition to the various waste forms) of two concentric containers of different metallic materials emplaced horizontally in a drift. The outer container or barrier will be of a highly corrosion-resistant nickel-chromium-molybdenum alloy, Alloy 22, surrounding an inner container made of Type 316 nuclear grade stainless steel. Additionally, an inverted U-shaped drip shield, fabricated with a titanium-palladium alloy (Titanium Grade 7), will be extended over the length of the emplacement drifts, resting on the drift invert, to enclose the top and sides of the waste packages. Each waste package will rest on an emplacement pallet made of two Alloy 22 V-shaped supports connected by square stainless steel tubes, and emplaced on top of the drift invert. The current repository reference design does not include backfill. For undisturbed repository conditions, corrosion is expected to be the primary degradation process limiting the life of the principal engineered barriers, which are the waste package and the drip shield. Through-wall penetration of the drip shield by corrosion will facilitate contact of the water entering into the emplacement drifts with the waste package outer surface. The quantity and chemistry of water contacting the waste package, the relative humidity, the waste package temperature, and the metallurgical condition of the waste package materials will determine the mode and rate of corrosion of the waste package outer container. Loss of containment as a result of corrosion will allow release of radionuclides to the environment surrounding the waste package and their subsequent transport through the engineered barrier subsystem. The relationship between this integrated subissue and other integrated subissues is depicted in Figure 3.3.1-1 (NRC, 2000a). The overall organization and identification of all the integrated subissues are depicted in Figure 1.1-2.

This section provides a review of the abstractions of the engineered barrier degradation processes incorporated by DOE in its Total System Performance Assessment–Site Recommendation (CRWMS M&O, 2000a). Only degradation processes under undisturbed repository conditions are discussed. Mechanical disruption of the engineered barriers and volcanic disruption of waste packages (depicted in the left portion of Figure 3.3.1-1) are discussed in Sections 3.3.2. and 3.3.10. The DOE description and technical bases for the engineered barriers degradation abstractions focused on the waste package and drip shield are documented in the process model report CRWMS M&O (2000b) and in several related analysis and model reports. These analysis and model reports are reviewed to the extent that they contain models, data, and analyses that support the proposed Total System Performance Assessment–Site Recommendation abstractions. As appropriate, several system description documents are also reviewed to complete the evaluation of models and abstractions used by DOE in the performance assessment of the engineered barriers.

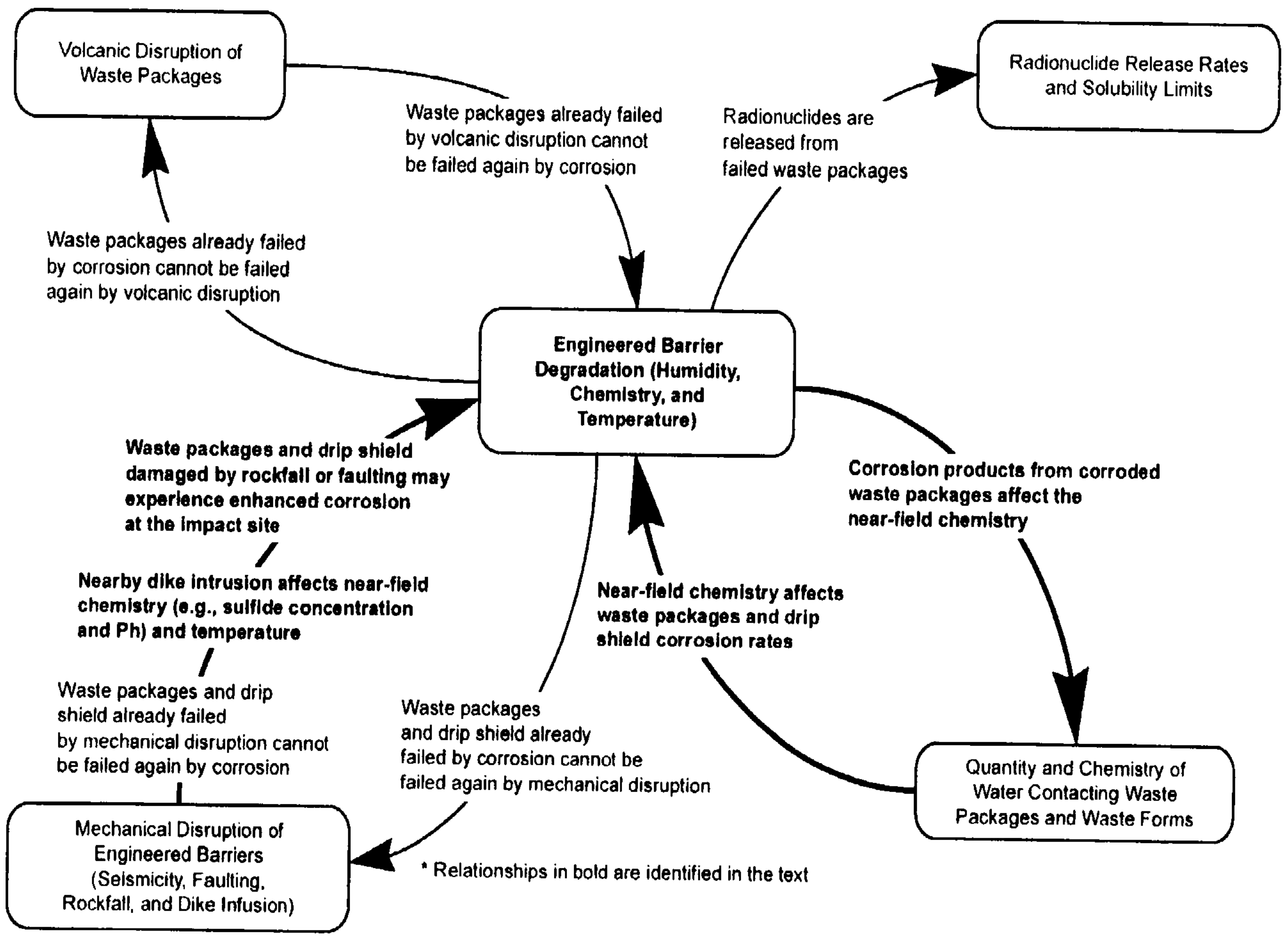

Figure 3.3.1-1. Diagram Illustrating the Relationship Between Engineered Barrier Degradation and Other Integrated Subissues

3.3.1.2 Relationship to Key Technical Issue Subissues

The Degradation of Engineered Barriers Integrated Subissue incorporates subject matter previously included in the following key technical issue subissues:

- Container Life and Source Term: Subissue 1—The Effects of Corrosion Processes on the Lifetime of the Containers (NRC, 2001)
- Container Life Source Term: Subissue 2—The Effects of Phase Instability of Materials and Initial Defects on the Mechanical Failure and Lifetime of the Containers (NRC, 2001)
- Container Life Source Term: Subissue 5—The Effect of In-package Criticality on Waste Package and Engineered Barrier Subsystem Performance (NRC, 2001)
- Container Life and Source Term: Subissue 6—The Effects of Alternate Engineered Barrier Subsystem Design Features on Container Lifetime and Radionuclide Release from the Engineered Barrier Subsystem (NRC, 2001)
- Total System Performance Assessment and Integration: Subissue 1—System Description and Demonstration of Multiple Barriers (NRC, 2000a)
- Total System Performance Assessment and Integration: Subissue 2—Scenario Analysis and Event Probability (NRC, 2000a)
- Total System Performance Assessment and Integration: Subissue 3—Model Abstraction (NRC, 2000a)
- Total System Performance Assessment and Integration: Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards (NRC, 2000a)
- Thermal Effects on Flow: Subissue 2—Thermal Effects on Temperature, Humidity, Saturation, and Flux (NRC, 2000b)
- Evolution of the Near-Field Environment: Subissue 2—The Effects of Coupled Thermal-hydrological-Chemical Processes on the Waste Package Chemical Environment (NRC, 2000c)
- Evolution of the Near-Field Environment: Subissue 3—The Effects of Coupled Thermal-hydrological-Chemical Processes on Chemical Environment for Radionuclide Release (NRC, 2000c)
- Evolution of the Near-Field Environment: Subissue 5—The Effects of Coupled Thermal-hydrological-Chemical Processes on Potential Nuclear Criticality in the Near Field (NRC, 2000c)

- Repository Design and Thermal-Mechanical Effects: Subissue 3—Thermal-Mechanical Effects on Underground Facility Design and Performance (NRC, 2000d)

The key technical issue subissues formed the bases for the previous versions of the issue resolution status reports and also were the bases for technical exchanges with DOE where agreements were reached on what additional information DOE needed to provide to resolve the subissue. The resolution status of this integrated subissue is based on the resolution status of each of the contributing key technical issue subissues. The subsequent sections incorporate applicable portions of these key technical issue subissues, however, no effort was made to explicitly identify each subissue.

3.3.1.3 Importance to Postclosure Performance

One aspect of risk-informing the NRC review was to determine how this integrated subissue is related to the DOE repository safety strategy. The performance of the engineered barriers after waste emplacement is extremely important to protect the public from any unreasonable long-term risk, as recognized in the DOE repository safety strategy for the proposed Yucca Mountain site (CRWMS M&O, 2000c). Both the performance of the waste package and that of the drip shield/drift invert system are listed among the eight principal factors for the postclosure safety case (CRWMS M&O, 2000c).

The waste package, composed of the containers and the waste forms, is the primary engineered barrier controlling the release of radionuclides from spent nuclear fuel and high-level waste glass. It should be noted, that contrary to the definitions of 10 CFR Part 63, DOE defines the waste package with the exclusion of the waste forms. Because corrosion processes, promoted by the presence of an aqueous environment contacting the surface of the containers, are the primary cause of container failure under undisturbed conditions, both the mode and rate of corrosion need to be evaluated to determine container lifetime. Corrosion processes potentially important in the degradation of the engineered barriers include humid-air and uniform aqueous corrosion, localized (pitting, crevice, and intergranular) corrosion, microbially influenced corrosion, stress corrosion cracking, and hydrogen embrittlement. In addition, dry-air oxidation occurs during the initial period after waste emplacement when the radioactive decay heat keeps moisture away from the gaseous environment surrounding the waste package. The ability of the waste package to contain radionuclides, and to limit their release after any initial penetration, is, therefore, determined by its long-term resistance to any of the modes of corrosion listed previously.

Performance of the drip shield needs to be considered as an important factor regarding safety because DOE incorporated it in the design of the engineered barrier subsystem to provide defense in depth by limiting the amount of water contacting the waste package as a result of dripping (CRWMS M&O, 2000c). Hence, the initiation of aqueous corrosion of waste packages can be delayed, resulting in a significantly longer container lifetime. In addition, once the containers are breached, the amount of water available for dissolution of both spent nuclear fuel and high-level waste glass and advective transport of the released radionuclides could be limited, even by the presence of a partially damaged drip shield.

The possibility of in-package criticality needs to be considered because steady-state criticality events could lead to increased radionuclide inventories. Depending on the power level and duration of critical conditions, significant amounts of radionuclides, including Tc-99, Np-237, and I-129, would be produced. The impact on repository performance would be an increase in radionuclide inventory available for release from the waste package and a potential increase in dose to the reasonably maximally exposed individual. Additionally, heat production from the additional fission reactions taking place during criticality conditions could indirectly impact repository performance by affecting the near-field environment and potentially increasing the waste package corrosion rate of the waste package and the dissolution rate of the waste form. Finally, a transient criticality event could result in mechanical failure of the already corroded waste package rupture of the spent nuclear fuel cladding, or both, increasing the exposed surface area and degradation rate of the spent nuclear fuel matrix.

3.3.1.4 Technical Basis

NRC developed a plan (2002) consistent with the acceptance criteria and review methods found in previous issue resolution status reports. A review of DOE approaches for including the degradation of engineered barriers in total system performance assessment abstractions is provided in the following subsections. The review of the technical basis for the degradation of engineered barriers abstraction is divided into three subsections: waste package, drip shield, and criticality within the waste package. Each subsection is organized according to the five acceptance criteria identified in Section 1.5: (i) System Description and Model Integration Are Adequate, (ii) Data Are Sufficient for Model Justification, (iii) Data Uncertainty Is Characterized and Propagated Through the Model Abstraction, (iv) Model Uncertainty Is Characterized and Propagated Through the Model Abstraction, and (v) Model Abstraction Output Is Supported by Objective Comparisons.

3.3.1.4.1 Degradation of the Waste Package

For undisturbed repository conditions, corrosion is considered the primary degradation process of the engineered barriers. In recent performance assessment studies, regardless of the specific waste package design, waste package degradation has been shown to be important to waste isolation at the proposed Yucca Mountain repository (Wilson, et al., 1994; CRWMS M&O, 1995, 1998a, 2000a; NRC, 1995, 1999; Kessler and McGuire, 1996; Shoesmith and Kolar,1998; DOE, 1998a; Mohanty and McCartin, 1998; Mohanty, et al., 1999). In addition, the NRC sensitivity studies have shown that the estimated average system performance during the 10,000-year period of regulatory interest is strongly influenced by the waste package lifetime (Mohanty, et al., 1999).

3.3.1.4.1.1 System Description and Model Integration Are Adequate

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the degradation of engineered barriers (degradation of the waste package) with respect to system description and model integration.

DOE documented the approach and technical basis for the abstraction of the degradation of the waste package in total system performance assessment in the process model report (CRWMS M&O, 2000b) and supporting analysis and model reports. The reference waste package design recommended for the proposed site recommendation (CRWMS M&O, 1999a) consists of an outer container of Alloy 22 surrounding an inner 5-cm [1.97-in] thick container made of Type 316 nuclear grade stainless steel. The main purpose of the inner container is to provide structural strength to the waste package. There are several design concepts for spent nuclear fuel and high-level waste glass containers (CRWMS M&O, 2000d), including five different designs for the commercial spent nuclear fuel with the same wall thickness {2 cm [0.79 in]} (CRWMS M&O, 2000e). The length, diameter, and interior of these five designs vary to accommodate fuel assembly variations. The commercial spent nuclear fuel disposal containers will be fabricated in two sizes (21 and 12 pressurized water reactor fuel assemblies) in which neutron absorber plates will be used. An additional waste package design for 21 pressurized water reactor fuel assemblies will contain control rods. The disposal containers for boiling water reactor spent nuclear fuel will be fabricated in two sizes for 44 and 24 fuel assemblies, both using neutron absorber plates. There are two designs that differ in length to hold the U.S. Navy spent nuclear fuel, both consisting of a single canister inside a disposal container with a wall thickness of 2.5 cm [0.98 in]. There are two designs of the codisposal container for DOE-owned spent nuclear fuel and high-level waste glass canisters, that only differ in length, having an outer container wall thickness of 2.5 cm [0.98 in]. These codisposal containers will hold five high-level waste glass canisters surrounding a DOE-owned spent nuclear fuel disposal canister inserted in the center of the container. The third waste package design for the DOE-owned spent nuclear fuel will accommodate two high-level waste glass canisters and two multicanister overpacks containing DOE-owned spent nuclear fuel canisters. A dual closure-lid design has been adopted for the waste package to mitigate against premature failure of the outer container as a result of stress corrosion cracking in the closure weld area. The closure end of the outer container, instead of one lid, has two lids. The inner lid is 1-cm [0.39-in] thick, and the outer lid is 2.5-cm [0.98-in] thick, with a physical gap between the two lids.

The corrosion processes potentially important in the degradation of the waste package outer container such as dry-air oxidation, humid-air and uniform aqueous corrosion, localized (pitting and crevice) corrosion, microbially influenced corrosion, stress corrosion cracking, and hydrogen embrittlement are considered in the process model report (CRWMS M&O, 2000b). The evaluation of features, events, and processes concerning waste package degradation that DOE has included or excluded (CRWMS M&O, 2000f) is described in Section 3.2.1 and incorporated into a features, events, and processes table. In general, there is agreement with DOE regarding the included features, events, and processes. The screening however, arguments and technical basis for several excluded features, events, and processes were not adequate, particularly those related to electrochemical processes and fabrication effects, including initial defects, welding processes, and postweld treatments. As described in Section 3.2.1, features, events, and processes were discussed during two Total System

Performance Assessment and Integration Technical Exchanges in May[1] and August 2001.[2] As a result of the meetings, DOE and NRC agreed on a path forward for each feature, event, and process (see Appendix B for specific details).

Dry-air oxidation is assumed to occur when the relative humidity of the repository environment is less then the critical relative humidity for the initiation of humid-air corrosion (CRWMS M&O, 2000b,g). The rate of dry-air oxidation is modeled assuming mass transport of reacting species limited by diffusion through the tightly adhering passive oxide film that results in a parabolic growth law where the film thickness is proportional to the square root of time. It is concluded that the oxidation rate is low at the waste package temperatures predicted after waste emplacement, and dry-air oxidation does not appear to limit waste package lifetime. For humid-air corrosion, DOE assumes that no water dripping occurs when relative humidity is greater than critical relative humidity. The corrosion rate and the distribution of corrosion rates are the same as for aqueous corrosion and are independent of time (CRWMS M&O, 2000b). The critical relative humidity is based on the deliquescence point (lowest relative humidity at which a saturated solution of the salt can be maintained at a given temperature) for sodium nitrate, which is conservatively assumed to be the salt that prevails on the container surface because it is the most hygroscopic salt that can be precipitated.

Aqueous corrosion is classified into two corrosion modes: general corrosion and localized corrosion. For corrosion-resistant nickel-chromium-molybdenum alloys such as Alloy 22, general corrosion in the expected waste package environments occurs in the form of passive corrosion, whereas localized corrosion is limited to pitting and crevice corrosion. Two conditions are considered to be simultaneously present for stabilization of an aqueous film on the waste package surface leading to aqueous corrosion—relative humidity in the emplacement drift greater than the deliquescence point of any salts deposited on the waste package surface and water dripping on the waste package. Below 100 °C [212 °F] the composition of water that contacts the waste package surface is assumed to be simulated J–13 concentrated water, whereas simulated saturated water is assumed to be present above 100 °C [212 °F]. Basic saturated water also has been identified as another plausible water chemistry that may develop on the waste package surface as a result of dripping and evaporation. The chemical composition of these waters is given in Table 3.3.1-1 (CRWMS M&O, 2000h). Two types of distinctive water chemistries were identified as produced by evaporation in laboratory experiments (CRWMS M&O, 2000i,j,k). Bicarbonate-type waters were generated by evaporation of synthetic J–13 water, whereas chloride-sulfate-type waters were formed by evaporation of pore water. In the bicarbonate-type waters, the ratio of the fluoride to chloride concentration is similar to that of the original J–13 water, however, the chloride concentration reaches values around 0.2 M. In the chloride-sulfate types, the concentration of fluoride is low

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

[2] Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

Table 3.3.1-1. Molar Concentration of Key Species in Simulated Concentrated Water, Simulated Saturated Water, and Basic Saturated Water*

Species	Simulated J–13 Concentrated Water (Molar)	Simulated Saturated Water (Molar)	Basic Saturated Water (Molar)
K^+	0.09	3.62	1.77
Na^+	1.78	2.12	4.74
F^-	0.07	0.00	0.07
Cl^-	0.19	3.62	3.82
NO_3^-	0.10	21.1	2.32
SO_4^{2-}	0.17	0.00	0.15
HCO_3^-	1.15	0.00	0.00
pH	—	—	11–13

*CRWMS M&O. "Environment on the Surfaces of the Drip Shield and Waste Package Outer Barrier." ANL–EBS–MD–000001. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000.

but the chloride concentration is in the molarity range. High chloride concentrations are also obtained in the modeling of the in-drift environment, taking into account seepage and thermal-hydrological-chemical coupled processes. These two types of water chemistries can lead to significant differences in the mode and rate of corrosion of waste package materials.

General corrosion is assumed to occur within the range of potentials leading to passive corrosion when the corrosion potential (E_{corr}) is less than the critical potential for the initiation of localized corrosion ($E_{critical}$). No mechanistic model is used to calculate corrosion rates within this regime. General corrosion rates are derived from weight-loss data obtained from the long-term corrosion test facility where numerous test specimens have been exposed to aqueous solutions based on modifications of J–13 water (CRWMS M&O, 2000g; McCright, 1998). Enhancement factors were used to consider the increases in corrosion rate associated with the effect of microstructural changes resulting from thermal treatments or modifications of the environment as a result of microbial activities. An enhancement factor is used to model the corrosion rate of thermally aged Alloy 22. Acceleration of the corrosion rates as a result of microbial activity is also treated using an enhancement factor, G_{MIC}. The condition for the occurrence of microbially influenced corrosion is a threshold relative humidity of 90 percent.

Localized corrosion of Alloy 22 is assumed to occur when the E_{corr} is greater than the $E_{critical}$. Mechanistic modeling of crevice corrosion to calculate spatial distributions of potential and current density, as well as transient calculations of dissolved species, was conducted. However, this deterministic modeling was not used in the model abstraction. Instead, initiation and repassivation potentials, as well as a potential defined by the occurrence of an anodic peak, defined as $E_{critical}$, were obtained in cyclic potentiodynamic polarization tests in a variety of electrolytes based on modifications of J–13 water. The potential for the anodic peak was conservatively selected to define the conditions for localized corrosion. The difference between $E_{critical}$ and E_{corr} for each solution tested was fitted to a function of the absolute temperature, the logarithm of the chloride concentration and the pH. It was found that the difference between $E_{critical}$ and E_{corr} depends on pH, but it does not exhibit any dependence on both absolute temperature and chloride concentration, over the range of conditions tested. Because of the lack of DOE experimental data, the rate of localized penetration of Alloy 22 was estimated from data available in the open literature using corrosion rates obtained in highly corrosive environments such as 10 percent $FeCl_3$ at 75 °C [167 °F]; dilute boiling HCl; and a solution containing 7 vol%, H_2SO_4, 3 vol% HCl, 1 wt% $FeCl_3$ and 1 wt% $CuCl_2$ at 102 °C [216 °F].

Stress corrosion cracking is one of the potential failure modes of the Alloy 22 outer container. DOE proposed two models for the evaluation of stress corrosion cracking susceptibility—the stress corrosion cracking stress intensity threshold model and the slip dissolution/film rupture model (CRWMS M&O, 2000b). The stress corrosion cracking threshold model is based on fracture mechanics concepts that suggest for stress corrosion cracking to occur, the stress intensity (K_I) at a flaw or defect must be equal to or greater than the threshold stress intensity factor for stress corrosion cracking (K_{Iscc}) in the presence of a corrosive environment. The slip dissolution/film rupture model relates crack advance to the metal anodic oxidation that occurs when the protective film at the crack tip is ruptured as a result of a tensile stress. In this model, a simple expression relates the crack propagation rate (V_t) with the crack tip strain rate ($\dot{\varepsilon}_{ct}$) and the crack tip strain rate with K_I, according to a power law relationship (CRWMS M&O, 2000l). For both the slip dissolution/film rupture model and the stress corrosion cracking threshold model, through-wall radial cracking is predicted as a result of the high values of the calculated stress intensity factor. Stress corrosion cracking, however, is limited to the surface area defined by the closure-lid welds. Therefore, the approach adopted by the DOE to mitigate or eliminate the possibility of crack growth is to reduce the residual stresses associated with welding. One method proposed involves the use of laser peening to introduce compressive stresses on the surface using multiple passes of a laser beam (CRWMS M&O, 2000b). This method will be used in the inner closure lid. The other method consists of localized annealing of the weld region using induction heating. This method will be applied to the weld in the outer closure lid.

All the corrosion process models discussed previously are abstracted and integrated in WAPDEG, the waste package degradation code, Version 4.0 (CRWMS M&O, 2000m). WAPDEG is a probabilistic code, incorporated in the Total System Performance Assessment–Site Recommendation, designed to run stochastic simulations in which random values are sampled to represent parameters in the corrosion models for calculating waste package lifetimes.

The description of the waste package, in terms of materials and fabrication processes that influence the consideration of corrosion processes affecting performance is adequate to the current level of design; however, a detailed description of the fabrication sequence and additional information on the effects of fabrication processes (e.g., welding and postweld thermal treatments) on the degradation of the containers will be needed as part of issue resolution. DOE studied the phase stability of Alloy 22, considering the precipitation of secondary topologically close-packed phases, such as μ, σ, and P-phase, which depend on time and temperature, (CRWMS M&O, 2000n). Alloy 22 specimens, exposed to temperatures in the range 427–800 °C [800–1,472 °F] for periods up to 40,000 hours, were analyzed for precipitation of topologically close-packed phases and long-range order. An activation energy for the precipitation of topologically close-packed phases has been determined to be near 280 kJ mol^{-1} [66.9 kcal mol^{-1}]. Based on the results of specimens analyzed thus far, bulk precipitation of topologically close-packed phases is not predicted in 10,000 years at 300 °C [572 °F] (CRWMS M&O, 2000b). The formation of grain boundary precipitates is deemed a worst-case scenario that would be equivalent to a 100-hour exposure at 700 °C [1,262 °F]. Using a similar Arrhenius-type relationship, it is predicted that the long-range order may occur after 1,000 years at 300 °C [572 °F]. No long-range order is predicted if the temperature remains below 260 °C [500 °F], however. Additional data and evaluations are necessary to properly model the effects of welding and thermal aging on the intergranular and crevice corrosion susceptibility of Alloy 22. The additional evaluations should include the effects of variations in base alloy composition, cold work, and water chemistry. In addition, the effects of welding parameters such as welding method, heat input, joint geometry, number of passes, and weld filler metal composition must be considered. DOE agreed[3] to provide updated information on aging, fabrication process, and welding. Detailed clarifications stated here need to be included in the agreed-on information.

In summary, the description of likely corrosion processes is sufficient for NRC to make regulatory decisions at the time of any future license application. Several aspects of modeling and model integration have limitations, however, because they are based on an empirical approach without sufficient mechanistic support. There is no clear integration between modeling of the environment in contact with the waste package, as discussed in detail in Section 3.3.3, and certain corrosion processes (e.g., localized corrosion), taking into account uncertainties in the calculated values of environmental variables such as chloride concentration and pH, among other factors. Additional information will be necessary to complete the evaluation of stress corrosion cracking modeling taking into account the proposed stress mitigation techniques resulting from postweld treatments and the detrimental effect of specific chemical species that may be present in the waste package environment. Most of these comments have been presented in more detail in NRC (2001). The technical bases for these comments are supported by the experimental work conducted at CNWRA, together with an extensive review of the open literature referenced in NRC (2001). Agreements reached with DOE regarding these comments are also documented in that report and summarized in Section 3.3.1.5. With the DOE agreement to provide the additional information, sufficient information should be available at the time of a potential license application for NRC to make a regulatory decision.

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

3.3.1.4.1.2 Data Are Sufficient for Model Justification

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the degradation of engineered barriers (degradation of the waste package) with respect to data being sufficient for model justification.

There are not enough data available for an accurate evaluation of dry-air oxidation and humid-air corrosion, but the data DOE used seem to be sufficient to bound the expected behavior. The assumption of parabolic growth of oxides on stainless steel and nickel-chromium-molybdenum alloys is not supported by either DOE data or independent tests performed outside the high-level waste disposal program (NRC, 2001). Parabolic oxidation kinetics, however, result in greater oxide penetration compared with either logarithmic or inverse logarithmic kinetics (Fehlner, 1986). At the temperatures expected for the proposed repository, complete oxide penetration of the Alloy 22 outer container by uniform oxidation is not expected. Physical processes that lead to accelerated oxidation rates, such as spalling or mechanical abrasion of the oxide layer, are not expected either. The DOE assumption of parabolic oxidation of Alloy 22 is bounding but should be supported by empirical evaluations of Alloy 22 and similar nickel-chromium-molybdenum alloys. An evaluation of the possibility of preferential oxidation at grain boundaries would be desirable based on the apparent susceptibility of nickel-base alloys to enhanced intergranular oxidation, which has been shown to be a factor in stress corrosion cracking of steam generator tubing (Bruemmer, et al., 2000). To address this issue, DOE agreed[4] to provide information on oxide film growth in air. Detailed clarifications stated here need to be included in the agreed-on information.

The approach used by DOE, assuming that the corrosion rates of Alloy 22 under humid-air conditions are the same as those for aqueous conditions, appears to be conservative. A comparison of aqueous and humid-air corrosion rates for Type 316L stainless steel (CRWMS M&O, 2000b) reveals that the humid-air corrosion rates are almost one order of magnitude less than the aqueous corrosion rates and thus supports the DOE approach.

General corrosion rates of Alloy 22 specimens exposed in the long-term corrosion test facility were calculated by measuring the weight loss of the specimens (American Society for Testing and Materials, 1997) after exposures of 6, 12, and 24 months. Weight gain was observed on 25 percent of the Alloy 22 specimens as a result of the deposition of silica (assumed to be amorphous SiO_2) on specimen surfaces. Data from specimens with weight gains were excluded from the distribution of corrosion rates that is equal to 0 nm/yr at the 0^{th} percentile, 27 nm/yr [1.06×10^{-3} mpy] at the 50^{th} percentile, 98 nm/yr [3.86×10^{-3} mpy] at the 90^{th} percentile, and 730 nm/yr [2.87×10^{-2} mpy] at the 100^{th} percentile. The distribution includes data from tests in a variety of solutions derived from J–13 water and included tests at 60 and 90 °C [140 and 194 °F], but is restricted to the 2-year exposure data. The abstracted general corrosion rate for the Alloy 22 outer container was found to be distributed between

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

10^{-6} and 7.3×10^{-5} mm/yr [3.9×10^{-5} and 2.9×10^{-3} mpy]. It was suggested, based on atomic force microscopy measurements, that the entire corrosion rate distribution can be corrected to take into account the weight gain caused by the deposited silicate by adding a value of 63 nm/yr [2.5×10^{-3} mpy] to the measured rates (CRWMS M&O, 2000b,g). The resulting distribution, that DOE defined as an alternative conservative model for waste package general corrosion, ranged from 4.0×10^{-6} to 1.8×10^{-4} mm/yr [1.6×10^{-4} to 7.1×10^{-3} mpy].

An enhancement factor, uniformly distributed between 1 and 2.5, is used to account for the corrosion rate of thermally aged Alloy 22. The value of the factor is based on the passive current density of the thermally aged specimen {700 °C [1,292 °F] for 173 hours} compared with that of an annealed specimen, both measured in potentiodynamic polarization tests (CRWMS M&O, 2000b).

The enhancement factor G_{MIC} is used to account for the acceleration of the corrosion rates as a result of microbial activity. For Type 316 nuclear grade stainless steel, a value of 10 is used for G_{MIC}, based on results obtained with Type 304 stainless steel. For Alloy 22, experimental results indicate a G_{MIC} of 2, based on the corrosion rate measured in short-term exposure tests (CRWMS M&O, 2000b,g). A value of G_{MIC} uniformly distributed between 1 and 2.0 is used in WAPDEG.

The distribution of localized corrosion rates is centered around the highest passive current density of 10 μA/cm² [9.2×10^{-4} A/ft²] that corresponds to a corrosion rate of 100 μm/yr [3.94 mpy]. The cumulative distribution of penetration rates for localized corrosion is equal to 12.7 μm/yr [0.5 mpy] for the 0th percentile, 127 μm/yr [5 mpy] for the 50th percentile, and 1,270 μm/yr [50 mpy] for the 100th percentile (CRWMS M&O, 2000b,g).

For the stress corrosion cracking of Alloy 22, crack propagation rates ranging from 2.1×10^{-11} to 7.6×10^{-12} m/s [8.27×10^{-10} to 3.0×10^{-10} in/s] were measured using a compact tension specimen at K_I = 30 MPa·m$^{1/2}$ [27.3 ksi·in$^{1/2}$] in an air-saturated alkaline solution (pH 13.4) with a composition similar to basic saturated water (Table 3.3.1-1) at 110 °C [230 °F] after a 3,585-hour exposure. These crack growth rates were used to determine the value of the repassivation parameter n (CRWMS M&O, 2000l). The parameter n is the exponent in the expression relating crack velocity with K_I in the slip dissolution/film rupture model. Because of the lack of sufficient data, the preexponential parameter A was considered to be equal to that reported for austenitic stainless steels in boiling water reactor environments. Assuming such a value for A, values of n ranging from 0.843 to 0.92 were then calculated from the measured crack growth rates listed previously. DOE recognizes that the variation of n as a function of environmental factors, which is one of the most important parameters in the model, is not available because of lack of experimental data. It should be noted that the range of values measured for n is the result of a single test conducted for 3,585 hours. Considering the uncertainty associated with the determination of n, values of 0.843 and 0.92 were selected to represent the lower and upper bounds of n using a uniform distribution (CRWMS M&O, 2000l). In the case of the stress intensity threshold model, a value of K_{Iscc} equal to 33 MPa·m$^{1/2}$ [30.3 ksi·in$^{1/2}$] was measured in N_2 deaerated 5-percent sodium chloride acidified to pH 2.7 at 90 °C [194 °F] (CRWMS M&O, 2000l). The value of 33 MPa·m$^{1/2}$ [30.3 ksi·in$^{1/2}$] with a standard deviation of 1.77 MPa·m$^{1/2}$ [1.61 ksi·in$^{1/2}$] was calculated from the results of duplicate tests using double cantilever beam specimens at 4 different initial K_I values ranging from 22 to 43 Mpa·m$^{1/2}$ [20 to 39 ksi·in$^{1/2}$].

In summary, the available data are not sufficient to justify the model abstractions for aqueous corrosion, in particular, for localized corrosion. The corrosion rates for general and localized corrosion, as well as the effect of changes in material conditions from fabrication processes (e.g., cold-working, welding, shop annealing, laser peening, and induction annealing) or environmental modifications as a result of microbial activity, do not include consideration of the complete range of environmental conditions that can be expected in the emplacement drifts. The solutions used in the tests, based on variations of J–13 Well water at 60 and 90 °C [140 and 194 °F], are not consistent with the environments predicted to result from the evolution of near-field processes (see Section 3.3.3). Lack of sufficient data weakens the justification of model abstractions (e.g., range of values assigned to enhancement factors). The enhancement factor for thermally aged specimens, based on limited short-term tests, implies that thermal aging will result only in an increased passive corrosion rate rather than in an increased susceptibility to localized or intergranular corrosion, as noted in other studies (NRC, 2001). The enhancement factor for microbially influenced corrosion, G_{MIC}, was calculated from the results of exposures to sterile and inoculated solutions (CRWMS M&O, 2000b,g). No information is provided on the possible preferential dissolution of alloying elements or on localized corrosion susceptibility as a result of microbial activity. In addition, the effects of temperature and environmental variations (e.g., pH, redox conditions, and ionic species) on the value of G_{MIC} are not available. To address all these concerns, DOE agreed[5] to provide sufficient information on aqueous corrosion, in particular, for localized corrosion. The agreed-on information will also include the effects of fabrication process and microbial activity and all credible environmental conditions.

3.3.1.4.1.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the degradation of engineered barriers (degradation of the waste package) with respect to data uncertainty being characterized and propagated through the model abstraction.

The most important implication of data uncertainty is related to the estimation of the distribution of waste package failure times. The importance of data uncertainty is also related to the contribution of specific corrosion processes to the overall performance and the propagation of data uncertainty in related and interdependent corrosion processes, as discussed next.

As noted in Section 3.3.1.4.1.1, humid-air corrosion is assumed to occur when relative humidity is greater than critical relative humidity. To define the characteristics of the environment in

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

contact with the surface of the waste package and the drip shield (dry versus humid air) and the corresponding corrosion process, the deliquescence point for $NaNO_3$ is used as the criterion for critical relative humidity (CRWMS M&O, 2000h). This choice is not justified, even though the deliquescence point for $NaNO_3$ seems to be the lowest among the salts that may be deposited on the surfaces of the waste package or the drip shield (CRWMS M&O, 2000h) because a mixture of salts usually has a lower deliquescence point than any of the individual salts that form the mixture. A relevant example is the $NaCl$-$NaNO_3$-KNO_3 system as shown in Table 3.3.1-2. For this system, the deliquescence point of the three salt mixture[6] is significantly lower than that of any of the individual salts. The lower deliquescence point or critical relative humidity implies that the waste package or drip shield may be subject to aqueous corrosion for a longer period of time when the temperature and the concentration of salts are both higher than those predicted. Additional details on this issue are provided in Section 3.3.3. To address this concern, DOE agreed[7] to provide information on the credible environmental conditions. More detailed clarifications stated here need to be included in the agreed-on information.

The DOE assumption of humid-air corrosion rates of Alloy 22 bounded by aqueous corrosion rates is acceptable. It would be useful to have additional data obtained outside the Yucca Mountain Project using information for Alloy 22 and similar nickel-chromium-molybdenum alloys. It appears that the uncertainty in the data will not lead to an erroneous evaluation of the effect of humid-air corrosion on waste package degradation. As the rates of aqueous corrosion are likely to encompass the humid air corrosion, NRC has no additional questions on this issue at this time.

Table 3.3.1-2. Deliquescence Point for Single Salts and Salt Mixtures at 16.5 °C [54.5 °F]

Salt(s)	Deliquescence Point
Pure NaCl	76 percent*
Pure $NaNO_3$	78 percent*
Pure KNO_3	95 percent*
Mixture of the listed salts (with a composition corresponding to a saturated solution of the three salts)	30.5 percent†

*CRWMS M&O. "Environment on the Surfaces of the Drip Shield and Waste Package Outer Barrier." ANL–EBS–MD–000001. Revision 00 ICN 01. Las Vegas, Nevada: CRWMS M&O. 2000.
†Weast, R.C., ed. *Handbook of Chemistry and Physics.* 54th Edition. Cleveland, Ohio: CRC Press. 1973.

[6]During the evaporation process, the composition of the dry salt formed as a result of losing the last amount of water would always have a composition corresponding to the saturated solution, no matter what the starting solution composition is, as long as it is within the chemical divide.

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

For aqueous corrosion, the DOE approach relies on passive dissolution rates of Alloy 22 determined via weight loss measurements. Because the passive corrosion rate of Alloy 22 is quite low, the change in mass is also small. For a typical 50- × 50- × 3.175-mm [1.97- × 1.97- × 0.125-in] test specimen with an area of 56.35 cm^2 [8.74 in^2] and a weight of 68.97 g [0.152 lb], a corrosion rate of 26.6 nm/yr [1.05×10^{-3} mpy] (DOE 50th percentile) is equivalent to a passive current density of 2.6×10^{-9} A/cm^2 [2.42×10^{-6} A/ft^2] or a mass loss rate of 0.00125 g/yr [0.000049 oz/yr]. For a 1-year exposure, the change in weight is less than 2×10^{-3} percent. Such small changes in weight can be determined provided there is not substantial interference from a competing process. In the case of the long-term corrosion test facility data, the deposition of silica was shown to interfere with the weight-loss data. The suggested correction (CRWMS M&O, 2000b,g) to the corrosion rate distribution {e.g., addition of 63 nm/yr [2.5×10^{-3} mpy]} may lead to a nonconservative estimation of the actual corrosion rates by overcorrecting the measured rates because the estimation does not account for the time-dependent changes in corrosion rate that must have occurred after the silica deposition. In addition, the value of the correction factor is more than twice the value of the median corrosion rate. An additional factor to consider is the use of a distribution in the corrosion rates that tends to give excessive weight in the computations of waste package life to the lowest corrosion rates within the distribution. On the contrary, the highest corrosion rates measured, if not accounted for in the distribution, would lead to container failure times much shorter that those currently predicted in the Total System Performance Assessment–Site Recommendation. To address this concern, DOE agreed[8] to provide justifications on the accurate measurements of corrosion rates, and their extrapolation and abstraction.

Higher corrosion rates have been observed for nickel-chromium-molybdenum alloys similar to Alloy 22 in a variety of environmental conditions relevant to the Yucca Mountain Project. Smailos (1993) reported corrosion rates of Alloy C–4 in brine environments containing 25.9 percent sodium chloride at 150 °C [302 °F] calculated, from weight loss measurements after 18-month exposures, to be in the range from 6×10^{-5} to 7×10^{-5} mm/yr [2.4×10^{-3} to 2.8×10^{-3} mpy]. In brines with 26.8 and 33 percent $MgCl_2$, the welded Alloy C-4 had a corrosion rate of 0.005 to 0.006 [0.2 to 2.4 mpy] . Bickford and Corbett (1985) measured corrosion rates of Alloy 22 in environments containing 20,000-p/m Cl^-; 2,300-p/m F^-; and 1,400-p/m SO_4^{2-}. In solutions with a pH of 1.6, the corrosion rates were 5 mm/yr [2 mpy] at 40 °C [104 °F] and 5 mm/yr [2 mpy] at 90 °C [194 °F], whereas, in solutions with pH 6, the corrosion rates were 5 mm/yr [2 mpy] at 40 °C [104 °F] and 0.012 mm/yr [0.47 mpy] at 90 °C [194 °F]. Harrar, et al. (1977, 1978) reported the corrosion rates of Alloys C-276 and 625 exposed to chloride containing groundwater at the Salton Sea geothermal field {100 °C [212°F] brine containing 12-percent chloride at a pH of 3.4}. General corrosion rates calculated using linear polarization were 0.0015 mm/yr [5.9×10^{-5} mpy] for Alloy C-276 and 0.007 mm/yr [2.8×10^{-4} mpy] for Alloy 625. In summary, the distribution of corrosion rates used for DOE in the WAPDEG calculations is lower than data reported in the literature, in some cases by more than one order of magnitude, for environments that appear to be relevant to the repository

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

conditions. To address this concern, DOE agreed[9] to provide justifications on the accurate measurements of corrosion rates, and their extrapolation and abstraction.

The relative corrosion rates of welded and base metal Alloy 22 were also determined using weight-loss specimens. Although the welded specimens are exposed along with the base alloy, the area of the welded region is quite small {approximately 10–15 cm^2 [1.6–2.35 in^2] and accounts for less than 25 percent of the total specimen-surface area. As a result, any accelerated corrosion rate of the welded region would be masked by the much larger area of the base alloy. To address this concern, DOE agreed[10] to use a larger surface area in corrosion testing, including welded samples cut from mockups.

The enhancement factor for the thermally aged specimens is based solely on short-term data and does not consider the effects of preferential corrosion that may occur at the grain boundary regions as indicated in previous investigations (Heubner, et al., 1989). Reductions in the $E_{critical}$ value are a strong indication that thermal aging increases the susceptibility of the alloy to localized corrosion, and more appropriate values of $E_{critical}$, such as crevice corrosion initiation and repassivation potentials, are necessary for a proper evaluation of thermal aging effects on localized corrosion. The increased current density, measured during an anodic polarization scan of an Alloy 22 specimen thermally aged for 173 hours at 700 °C [1,292 °F], was averaged over the entire exposed surface area. In light of the increased susceptibility of thermally aged nickel-chromium-molybdenum alloys to intergranular corrosion, the increased current density observed in the DOE test may be the result of preferential dissolution at grain boundaries rather than an overall increase in the corrosion rate. Such preferential attack, mainly confined to the grain boundary regions, would result in a true enhancement factor much greater than 2.5. To address this concern, DOE agreed[11] to provide updated information on the effects of thermal aging on corrosion.

Uncertainty in the data for the general corrosion rate of Alloy 22 also applies to the effects of long-term changes on the chemical composition and stability of oxide films. Previous investigations indicated that the composition of the passive oxide film becomes enriched in chromium and depleted in molybdenum and nickel (NRC, 2001). The long-term effects of preferential dissolution of alloying elements may include changes in the oxide film composition that, in turn, may alter the passive corrosion rate or promote susceptibility to localized corrosion. Information on the preferential dissolution of alloying elements has not been obtained from the specimens tested in the long-term corrosion test facility. To address this concern, DOE agreed[12] to provide information on the long-term behavior of passive films.

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

[10]Ibid.

[11]Ibid.

[12]Ibid.

Localized corrosion rates assumed by DOE, obtained from literature data using acidic chloride and acidic oxidizing chloride solutions, appear to correspond to measured corrosion penetration rates obtained in certain service environments, as reviewed by Cragnolino, et al. (1999). Smailos (1993) reported a maximum pit depth of 0.90 mm [0.035 in] in Alloy 625 after 18 months in 33 percent $MgCl_2$ at 150 °C [272 °F], corresponding to a localized corrosion penetration rate of 0.6 mm/yr [24 mpy]. Carter and Cramer (1974) reported that pit penetration rates for Alloy 625 were 0.22 mm/yr [8.7 mpy] after 45 days in 105 °C [221 °F] brine containing 155,000 p/m chloride with 30-p/m sulfur. Oldfield (1995) observed crevice corrosion of Alloys 625 and C-276 in both natural and chlorinated seawater at ambient temperature. The average penetration rate for Alloy 625 following a 2-year exposure was 0.049 mm/yr [1.9 mpy]. These observations clearly indicate the importance of defining conditions for the initiation and arrest of localized corrosion because these rates of penetration are several orders of magnitude greater than those corresponding to passive general corrosion and also are greater than those selected by DOE for localized corrosion. To address this concern, DOE agreed[13] to provide information on the environmental and electrochemical conditions for localized corrosion.

The DOE modeling of stress corrosion cracking of the Alloy 22 outer container considers a narrow range of expected waste package environments and is limited to the closure lid weld stresses. As noted, two stress corrosion cracking models, the threshold stress intensity model and the slip dissolution/film rupture model, are being used (CRWMS M&O, 2000l). In the first model, stress corrosion cracking susceptibility of Alloy 22 is evaluated using model parameters obtained from Lawrence Livermore National Laboratory data; whereas, in the second case, experimental data obtained at General Electric Corporation for Alloy 22 are combined with data reported for stainless steel in boiling water reactor environments. Evaluation of these two alternative models reveals that while a K_{Iscc} value of 33 $MPa{\cdot}m^{1/2}$ [30 $ksi{\cdot}in^{1/2}$], determined by Roy, et al. (1998), is adopted in the threshold model, the slip dissolution/film rupture model predicts crack propagation at K_I values less than the experimentally determined value of K_{Iscc}. It is claimed, however, that the General Electric Corporation data were obtained during cyclic loading conditions rather than constant load. Crack propagation rates for Alloy 22 are found to be extremely low, and the absence of crack growth under constant load conditions was confirmed experimentally.

The residual stress analyses performed by DOE, using a finite element method, indicate that given the calculated maximum stress intensity factors from weld residual stress and a K_{Iscc} determined by Lawrence Livermore National Laboratory, a radially oriented flaw perpendicular to weld may initiate stress corrosion cracking of the Alloy 22 outer container. In contrast, no stress corrosion cracking initiation at a circumferentially oriented flaw parallel to weld is expected based on the threshold value. These arguments are based on the threshold or minimum stress intensity criterion. K_{Iscc}, however, could be lower in a different environment than that tested (Speidel, 1981). The validity of K_{Iscc} as a bounding parameter for performance should be assessed through an appropriate combination of experimental and modeling work. K_{Iscc} values ranging from approximately 8 to 20 $MPa{\cdot}m^{1/2}$ [7.3 to 18.2 $ksi{\cdot}in^{1/2}$] have been

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

observed for Types 304, 304L, 316, and other similar austenitic stainless steels in chloride-containing solutions at temperatures ranging from 80 to 130 °C [176 to 266 °F] (Cragnolino and Sridhar, 1992). As expected, the values in the lower end of that range are observed with both increasing temperatures and chloride concentration. It is also recognized that K_{Iscc} values are affected by the electrode potential. On the basis of these observations, it is apparent that the composition of the environment is another constraint that must be considered when using K_{Iscc} as a bounding parameter for the initiation of stress corrosion cracking. To address this concern, DOE agreed[14] to provide stress corrosion cracking data for credible environmental conditions.

The effects of waste package fabrication processes (e.g., welding and heat treatments) on stress corrosion cracking of candidate container materials still remain major concerns. Residual stresses from waste package fabrication or applied stresses resulting from seismic events combined with the necessary environmental conditions may be sufficient to cause stress corrosion cracking of the outer container. If high residual stresses result from fabrication processes, the mechanical component necessary for stress corrosion cracking may be present in every waste package placed in the repository. As noted, DOE proposed postweld treatments to mitigate the effect of residual stresses. The effects of welding and postweld heat treatments on the stress corrosion cracking susceptibility of Alloy 22, as well as the respective K_{Iscc} values in the expected waste package environment, have not been evaluated. Additionally, the DOE stress corrosion cracking models consider weld residual stress the only source of stresses significant to stress corrosion cracking (CRWMS M&O, 2000b,l). Other sources of stress are assumed to be either insignificant such as dead load stress or temporary like seismic stress. Accordingly, the effects of other possible types of applied stresses in the repository have not been assessed. In particular, stresses generated at the line of contact of the waste package with the emplacement pallet should be evaluated. To address this concern, DOE agreed[15] to provide updated information on metallurgical conditions for stress corrosion cracking and its mitigation processes. More detailed clarifications stated here need to be included in the agreed-on information.

Data used to analyze the effects of initial defects on the performance of the waste package outer barrier (CRWMS M&O, 2000o) have uncertainties that have not been characterized nor propagated through the model abstraction. DOE estimates of the probabilities for initial defects in the waste package from various sources range from 10^{-8} to 10^{-3} per waste package. In the specific case of weld flaw, the probability of initial through-wall defect {e.g., defect size larger than 20 mm [0.79 in]} is estimated to be less than 10^{-11} per waste package for the top lid closure weld of Alloy 22. The consequence of this initial flaw is calculated as stress corrosion cracking growth. The effects of initial defects on other corrosion and mechanical failure processes were also ignored. Although surface intersecting flaws are more important for stress corrosion cracking than completely enclosed flaws, the stress and strain localization from the latter may adversely affect stress corrosion cracking, depending on the size and location of the

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

[15]Ibid.

flaw. Additionally, if one of the sources of defect is mis-heat treatment, the potential lowering of fracture toughness parameters because of precipitation of embrittling phases (μ-phase in Alloy 22), in combination with internal flaws and residual stresses, can cause mechanical fracture of the container as discussed in Section 3.3.2. To address this concern, DOE agreed[16] to information on stress corrosion cracking covering a full range of metallurgical conditions. Detailed clarifications stated here need to be included in the agreed-on information.

In the application of the slip dissolution/film rupture model to Alloy 22, DOE adopted values ranging from 0.843 to 0.92 for the repassivation slope, *n* (CRWMS M&O, 2000l). This range of values for *n* was calculated from a single experiment conducted for 3,385 hours during cyclic loading conditions R = 0.5–0.7, with frequency 0.001–0.003 Hz, at a maximum K_I = 30 MPa·m$^{1/2}$ [27.3 ksi·in$^{1/2}$]. Input for the model includes average crack growth rates ranging from 2.1×10^{-11} to 7.6×10^{-12} m/s [8.3×10^{-10} to 3.0×10^{-10} in/s] and the empirical relationship adopted from the work of Ford and Andresen (1988) on the stress corrosion cracking of austenitic stainless steels in boiling water reactor environments as previously reviewed by Sridhar, et al. (1993), in the empirical relationships developed by Ford and Andresen (1988), the two interdependent model parameters (*n* and *A*) used to define the crack propagation rate/crack tip strain rate relationship in the slip dissolution/film rupture model are dependent on material properties and the environment at the crack tip. From analysis of the extensive work conducted by Ford and Andresen (1988), it can be concluded that most of the final expressions for calculating crack propagation rates and crack tip strain rates requires the input of field data to adjust several of the parameters included in the model. This is particularly true in the case of the parameter *n*, but also applies to the preexponential coefficient *A*. The model parameters in the slip dissolution/film rupture model are largely empirical correlations on the basis of a combination of laboratory experimental results and field observations. Therefore, application of these empirical relationships to Alloy 22 requires a more complete database to limit propagation of the uncertainty characterizing currently available data into the modeling of stress corrosion cracking of Alloy 22. To address this concern, DOE agreed[17] to provide sufficient data on relevant parameters for stress corrosion cracking models.

Recently, Barkatt and Gorman (2000) reported stress corrosion cracking of Alloy 22 in concentrated J–13 Well water of pH 0.5 (acidified with hydrochloric acid) containing lead at relatively high concentrations (~1,000 p/m). Tests were conducted at 250 °C [452 °F] using U-bend specimens. These test conditions were extremely severe in terms of lead concentrations, temperature, and stress, and the results are preliminary; nevertheless, the possible detrimental effects of impurities such as lead, mercury, or arsenic require further evaluation. If the results are valid for the repository conditions, the current model abstraction for stress corrosion cracking will need reevaluation to account for the effects of these species.

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

[17]Ibid.

To address this concern, DOE agreed[18] to provide credible environmental conditions. Detailed clarifications stated here need to be included in the agreed-on information.

This section summarized characterization and propagation of data uncertainties. Various sources of the uncertainties were identified from the involved corrosion processes. They include credible environmental conditions, accurate measurements of corrosion rates, acceptable extrapolation and abstraction of laboratory data, and acceptable conditions for localized corrosion and stress corrosion cracking. The effects of thermal aging, fabrication processes (including welding), and microbial activity on corrosion were also evaluated.

As noted previously, DOE agreed to provide the needed information before any future license application being submitted.

3.3.1.4.1.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the degradation of engineered barriers (degradation of the waste package) with respect to model uncertainty being characterized and propagated through the model abstraction.

The distribution of passive corrosion rates used by DOE is not supported by the electrochemical measurements conducted within the Yucca Mountain project and is lower than corrosion rates measured in a variety of service environments. Combining electrochemical techniques with chemical analysis of alloying elements is a well-established method for measuring passive dissolution rates. The low passive corrosion rate of Alloy 22 is the result of formation of a protective chromium oxide passive film. Kirchheim, et al. (1989) reported a passive current density of 0.014 $\mu A/cm^2$ [1.3×10^{-5} A/ft^2] {corrosion rate of 9.68×10^{-5} mm y^{-1} [3.8×10^{-3} mpy]} for pure chromium in 1 N H_2SO_4. The rates for Ni-Cr-Mo alloys are expected to be higher, even in neutral chloride solutions simulating the aqueous environments contacting waste packages. To address this concern, DOE agreed[19] to conduct appropriate electrochemical tests or provide justification for the approach adopted in the measurements currently being conducted.

In addition, the corrosion rate data used by DOE do not consider the effects of long-term changes to the composition of the oxide films. Previous investigations (Lorang, et al., 1990) indicated that the composition of the oxide film, which acts as a barrier for mass transport, becomes enriched in chromium and depleted in molybdenum and nickel. The long-term effects of preferential dissolution of alloying elements may include changes to the oxide film composition that could, in turn, alter the passive corrosion rate or promote an increase in the susceptibility of the alloy to localized corrosion. Information on the preferential dissolution of

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

[19]Ibid.

alloying elements has not been obtained from long-term corrosion test facility specimens. To address this concern, DOE agreed[20] to provide information on the long-term behavior of passive films.

Determination of the Alloy 22 localized corrosion susceptibility by comparing the corrosion potentials and critical potentials measured in short-term tests may not be acceptable. Selection of the $E_{critical}$ should be based on the most likely corrosion mode for the alloy and must consider the environmental effects of temperature, solution chemistry, and the presence of microbes, as well as the effects of material property variations caused by fabrication, welding, thermal aging, and long-term evolution of the oxide film composition and characteristics. In addition, the range of environmental effects such as radiolysis and water chemistry, material factors such as formation of thermal oxide films, and the long-term evolution of the oxide film composition should be included in the bounding analyses of the E_{corr}. The present set of data used as criteria to evaluate the localized corrosion susceptibility of the outer container, as referenced in CRWMS M&O (2000b), is limited to $E_{critical}$ obtained in short-term tests. Confirmatory tests designed to determine the validity of the $E_{critical}$ approach seem to be necessary. To address this concern, DOE agreed[21] to provide information on the electrochemical and environmental conditions for localized corrosion.

Determination that the localized corrosion susceptibility of Alloy 22 is not affected by thermal aging based on the difference between the E_{corr} and the $E_{critical}$ may be nonconservative. The selected value of the $E_{critical}$, which may be a combination of pit initiation, transpassive dissolution and oxygen evolution, is misleading because it does not compare other possible values of $E_{critical}$ such as the initiation and repassivation potentials for crevice corrosion with E_{corr}. Reduction of the pit initiation potential observed for the thermally aged specimen is a strong indication that thermal aging reduces the localized corrosion susceptibility of Alloy 22. Previous investigations identified the formation of topologically closed-packed phases in both thermally aged (Heubner, et al., 1989) and welded (Cieslak, et al., 1986) Alloy 22. Observations of preferential initiation of localized corrosion in weldments and grain boundary attack of the thermally aged material (Heubner, et al., 1989), as well as a lower critical pitting temperature for welded Alloy 22 (Sridhar, 1990), do not support the DOE conclusion of no reduced susceptibility to localized corrosion after thermal aging. Reduction of the E_{corr} after thermal aging suggests an increase in the passive current density. As previously indicated, this increase may be a result of significantly enhanced dissolution at grain boundaries. To address this concern, DOE agreed[22] to provide evaluation of metallurgical conditions affecting localized corrosion, especially for thermally aged samples.

In addition to environmental effects, the DOE evaluation of the stress corrosion cracking susceptibility of Alloy 22 should consider the effects of variations in material properties,

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

[21]Ibid.

[22]Ibid.

fabrication and welding, and long-term exposure to elevated temperatures. These variations are not easily correlated with compositional variations or differences in mechanical properties. Segregation of alloying elements and the formation of topologically close-packed phases in the welded regions has been shown to occur for Alloy 22 (Cieslak, et al., 1986), and thermal aging has been shown to increase localized corrosion susceptibility (Heubner, et al., 1989). Long-term exposure of the waste package to elevated temperatures expected in the proposed repository may result in microstructural alterations that may be equivalent to aging for 100 hours at 700 °C [1,292 °F] (CRWMS M&O, 2000b). To address this concern, DOE agreed[23] to provide acceptable evaluation of metallurgical conditions for stress corrosion cracking, especially for thermally aged samples.

This section summarized characterization and propagation of model uncertainty. The sources of the uncertainties included the long-term behavior of passive film, and conditions for localized corrosion and stress corrosion cracking.

As noted previously, DOE agreed to provide the needed information before any future license application is submitted.

3.3.1.4.1.5 Model Abstraction Output Is Supported by Objective Comparisons

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the degradation of engineered barriers (degradation of the waste package) with respect to model abstraction output being supported by objective comparisons.

DOE data for the corrosion rates of Alloy 22, obtained in the long-term corrosion test facility, are not reliable because of the deposition of silica and the limitations of the weight loss measurements to evaluate the effects of welding. Additional tests, where interference from deposition processes do not occur, should be performed to confirm or correct the results obtained using long-term corrosion test facility specimens. Determination of passive corrosion rates from weight loss may be possible in solutions that do not contain dissolved silica, divalent cations such as calcium, or other species that can precipitate from solution and deposit on the test specimens. As an alternative to weight loss, steady-state anodic current density measurements obtained under potentiostatic conditions can be used to determine corrosion rates according to American Society for Testing and Materials G102 (American Society for Testing and Materials, 1999). A more substantiated discussion about the long-term validity of low passive corrosion rates of Alloy 22 needs to be justified using an appropriate combination of testing and calculations. The use of source data in the models appears to be inconsistent. To address this concern, DOE agreed[24] to provide accurate corrosion data and its acceptable extrapolation.

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

[24]Ibid.

Although the K_{Iscc} value determined by Roy, et al. (1998) is adopted in the threshold model, different source data for the crack growth rate are used in the slip dissolution/film rupture model. Not only the data have been obtained using different types of fracture mechanics specimens and test methods, but the environments are widely different in chemical composition, pH, and redox potential. In addition, the environments used to evaluate the stress corrosion cracking susceptibility of Alloy 22 using the stress corrosion cracking threshold model are not consistent with the environments expected on the drip shield and waste package (CRWMS M&O, 2000h). K_{Iscc} values used to determine stress corrosion cracking susceptibility should be based on measurements conducted in environments that may be expected in the proposed repository because K_{Iscc} values are strongly dependent on both the material and the environment (Speidel, 1981). At present, the slip dissolution/film rupture model for Alloy 22 uses a combination of parameters derived from stainless steel in boiling water reactor environments (Ford and Andresen, 1988; Ford, 1990) and limited amount of data obtained from laboratory tests (CRWMS M&O, 2000l). Although the model is theoretically based on fundamental parameters such as the repassivation rate, in practice, the critical parameters are empirically derived using a substantial volume of data obtained in boiling water reactor environments (Ford and Andresen, 1988; Ford, 1990) that are not available for Alloy 22 in the expected waste package environments. To address this concern, DOE agreed to provide supporting data bases for stress corrosion cracking models. Detailed clarifications stated here need to be included in the agreed-on information.

The effects of the postweld annealing treatment proposed for the dual lid waste package outer container on the stress corrosion cracking susceptibility of Alloy 22 should also be evaluated. The proposed annealing treatment relies on rapid heating and cooling cycles (CRWMS M&O, 2000b). Because only the end of the waste package is elevated to temperatures beyond 1,000 °C [1,802 °F], significant thermal gradients will exist that may result in the exposure of some portions of the waste package outer barrier to temperatures that favor the formation of detrimental topologically close-packed phases. Variations in the annealing parameters may exacerbate microstructural alterations and further reduce the stress corrosion cracking resistance of the alloy. There is no specific experience on laser peening of Alloy 22. To address this concern, DOE agreed[25] to provide additional information on postwelding processes for mitigating stress corrosion cracking.

Section 3.3.1.4.1.5 addresses uncertainties associated with accurate determinations of uniform corrosion rates, insufficient data base and rationales in using existing stress corrosion cracking models, and insufficient evaluation of welding effects on stress corrosion cracking.

As noted previously, DOE agreed to provide the needed information before any future license application is submitted.

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

3.3.1.4.2 Degradation of the Drip Shield

For undisturbed repository conditions, corrosion is also considered the primary degradation process of the drip shield. Because the drip shield was not included as an engineered barrier subsystem design feature in the viability assessment, there is only a single calculation showing the beneficial effect of the drip shield on waste package life and dose in DOE (1998a). In recent performance assessment sensitivity analysis calculations for the site recommendation, the beneficial effect of the drip shield on the predicted annual dose rate is only apparent after 50,000 years (CRWMS M&O, 2000a).

3.3.1.4.2.1 System Description and Model Integration Are Adequate

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.6.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the degradation of engineered barriers (degradation of the drip shield) with respect to system description and model integration.

DOE has documented the approach and technical basis for the abstraction of the degradation of the drip shield in total system performance assessment in a process model report (CRWMS M&O, 2000b) and supporting analysis and model reports. Use of a drip shield as a design option is intended to minimize the possibility of water dripping on containers. Corrosion of the containers can be enhanced by the presence of flowing liquid water that may facilitate localized penetration if the chemical composition of the water is sufficiently aggressive. In addition, liquid water can mobilize and advectively transport most radionuclides. Although moisture condensation between the waste package and the drip shield cannot be prevented, the purpose of the drip shield is to reduce water contact arising from fracture flow. Where active flowing fractures in the repository are coupled with sharp drift wall edges, seeps (drips) into the drift can occur. The principal function of the drip shield is to divert these drips from the waste package surface. The site recommendation design calls for an inverted U-shaped drip shield to be constructed with 1.5-cm [0.59-in]-thick Titanium Grade 7 (Ti-0.15Pd) or Grade 16 (Ti-0.05Pd) plates and structural members made of Titanium Grade 24 (Ti-6Al-4V-0.15Pd) for long-term structural support (CRWMS M&O, 2000p). The drip shield will be extended throughout the length of the emplacement drifts to enclose the top and sides of the waste package and will rest on top of the drift invert made of steel beams and filled up with ballast. The emplacements drifts will have steel sets and lagging (or, in some cases, rock bolts and mesh) for ground support instead of the concrete liner proposed in the viability assessment design.

The DOE approach consists of examining the possible environments to which the drip shield may be exposed (e.g., temperature and chemistry of incoming water) and evaluating the effects of these conditions on the possible degradation modes and rates for palladium-bearing titanium alloys. Degradation modes considered (CRWMS M&O, 2000b) include thermal embrittlement, dry-air oxidation, humid-air corrosion, uniform aqueous corrosion, localized (pitting and crevice) aqueous corrosion, and environmentally assisted cracking (consisting of stress corrosion cracking and hydrogen embrittlement or hydride-induced cracking).

The possibility for thermal embrittlement of titanium used in drip shield construction was excluded for further analysis because thermal embrittlement was considered to have a low probability of occurrence in the features, events, and processes analysis (CRWMS M&O, 2000f), discussed in Section 3.2.1. Mechanical degradation and collapse of the emplacement drifts, with potential effects on the temperature of the drip shield and moisture flow into the engineered barrier subsystems was also screened out (CRWMS M&O, 2000q). This type of drift degradation event may have an important effect on the integrity of the drip shield and should be considered, as discussed in detail in Section 3.3.2. To address this concern, DOE agreed[26] to provide information on the embrittlement of drip shield materials. Detailed clarifications stated here need to be included in the agreed-on information, especially related to the drip shield fabrication. DOE also agreed to provide sufficient information on the mechanical degradation of drip shields and the effects of drift collapse.

Environmentally assisted cracking was examined considering two main processes: stress corrosion cracking and hydride-induced cracking. The process model report, corresponding analysis and model reports, and other technical documents (CRWMS M&O, 2000b,l,r,s) made a clear distinction between stress corrosion cracking and hydride-induced cracking. Within this framework, the only viable source of stress needed for stress corrosion cracking results from rockfall because it is stated that the drip shield will be fully annealed after welding to minimize residual stresses. Two different models for evaluating stress corrosion crack propagation were considered—the stress intensity threshold model and the slip dissolution-film rupture model. The approach taken by DOE to evaluate hydride-induced cracking is based on the assumption that the dominant cathodic reaction occurring on the metal surface during passive (uniform) dissolution is hydrogen evolution, and it is assigned a reaction rate equal to the passive dissolution rate calculated from weight-loss coupon testing. Of the hydrogen gas produced from this cathodic reaction, a fraction (between 0.02 and 0.10) is postulated to enter into the metal as hydrogen atoms and precipitate as hydrides, which may then lead to a loss in ductility (e.g., hydride embrittlement). Hydride-induced cracking is said to be possible once a critical hydrogen concentration has been exceeded. Based on the uniform corrosion rates calculated from weight-loss coupon testing and assumptions regarding the fraction of hydrogen eventually absorbed into the metal lattice, it was concluded that hydride-induced cracking does not have a significant effect on the drip shield life expectancy during the 10,000-year performance period.

Additional examination of possible galvanic interactions with iron-based components in the repository (e.g., rock bolts, steel supports, and gantry rail) led DOE to suggest that only localized areas of galvanic interaction were possible. Given that the cathode (drip shield) to anode (steel component) area ratios would be large, it is assumed that any hydrogen produced would be mostly absorbed in a large volume of titanium such that the concentration would be low. In any event, the consequence for both stress corrosion cracking and hydride-induced cracking was considered to be low because any cracks that developed would be plugged by corrosion products and, therefore, would not be available for the transport of water and subsequent dripping onto the waste package.

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

The dry-air oxidation and the humid-air and aqueous corrosion processes of the drip shield are integrated in WAPDEG (CRWMS M&O, 2000m) and the model abstraction used for these processes is identical to that used for calculating the lifetime of the waste packages. However, a logarithmic growth law was considered as appropriate for the dry-air oxidation of titanium alloys instead of the parabolic law used for the outer waste package container (CRWMS M&O, 2000b,t). A similar criterion to that used in the case of the waste package was applied for the initiation of humid-air and general aqueous corrosion. The general corrosion rates used for these two processes were derived from weight-loss data obtained from the long-term corrosion test facility using Titanium Grade 16 instead of Titanium Grade 7.

As for Alloy 22, localized corrosion of titanium alloys is assumed to occur when the E_{corr} is greater than the $E_{critical}$. Only crevice corrosion is considered because pitting corrosion is disregarded as a plausible degradation process because it was not observed in the long-term corrosion test facility tests. Initiation and threshold potentials were obtained in cyclic potentiodynamic polarization tests in a variety of electrolytes based on modifications of J–13 Well water. The difference between $E_{critical}$ and E_{corr} for each solution tested was plotted as a function of temperature. The difference between $E_{critical}$ and E_{corr} was sufficiently large to preclude the occurrence of crevice corrosion for the range of conditions tested.

In summary, the description of the drip shield materials is adequate for consideration of the corrosion processes affecting performance; however, many details regarding fabrication (e.g., welding, postweld treatments) will be needed for performance assessment at the time of license application. Whereas the description of likely corrosion processes is sufficient, many aspects of model abstraction and integration have limitations. Uncertainties in the composition of the water contacting the drip shield (e.g., fluoride content) may have a significant effect on performance of the drip shield and its expected function. To address these concerns, DOE agreed[27] to provide sufficient information on detailed fabrication processes, model abstraction and integration of corrosion processes, and credible environmental conditions including the composition of the contacting water (e.g., fluoride content).

3.3.1.4.2.2 Data Are Sufficient for Model Justification

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the degradation of engineered barriers (degradation of the drip shield) with respect to data being sufficient for model justification.

There are not enough data available to accurately evaluate dry-air oxidation and humid-air corrosion of the drip shield, but the data DOE used seem sufficient for bounding the expected behavior. According to the waste package degradation process model report and the general and localized corrosions of the drip shield analysis and model report (CRWMS M&O, 2000b,t), Titanium Grade 16 coupons were exposed for 1 year to several aqueous solutions that were

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

variants of J–13 Well water. Tests showed there was little influence of temperature from 60 to 90 °C [140 to 164 °F] nor was there a significant influence of the testing environment. A wide variation in the measured weight loss, resulting in corrosion rates of ~ −1,700 to 150 nm/yr [6.7×10^{-2} to 5.9×10^{-3} mpy], was reported, however. It is apparent from the negative values that the data include specimens exhibiting significant weight gain. The variability was explained as resulting from differences in the postexposure cleaning procedures used to remove corrosion product buildup. Similar tests were conducted using creviced specimens with no significant attack observed under the crevice former. In this case, rates ranging from −350 to 350 nm/yr [-1.4×10^{-2} to 1.4×10^{-2} mpy] were calculated. Because it was noted that the corrosion rates were similar for the uniform corrosion coupons and the crevice corrosion coupons, it was assumed that the main corrosion mode for the creviced specimens was also uniform passive corrosion of the exposed surfaces. Data from specimens exhibiting weight gain were excluded from the cumulative distribution function of corrosion rates. Based on the maximum corrosion rates observed {350 nm/yr [1.4×10^{-2} mpy] for creviced specimens}, it was concluded that failure of titanium alloy drip shields would be unlikely within the 10,000-year performance period.

A limited set of cyclic potentiodynamic polarization experiments was also performed to examine localized corrosion susceptibility. Based on experiments conducted in simulated saturated water at 120 °C [218 °F] and in simulated J–13 concentrated water at 90 °C [164 °F] (the nominal compositions for these solutions are shown in Table 3.3.1-1), no localized corrosion was noted even when polarization was conducted to 2.5 $V_{Ag/AgCl}$. A critical threshold potential was observed in the polarization scans near 1 $V_{Ag/AgCl}$ and was believed to be associated with oxygen evolution (CRWMS M&O, 2000t).

In summary, the available data, although sufficient for justification of the uniform corrosion model abstraction, do not incorporate the effects of fabrication processes nor the complete range of environmental conditions that can be expected in the emplacement drifts. In particular, the potential detrimental effect of fluoride anions in accelerating the dissolution rate of titanium alloys above a certain threshold concentration (Brossia and Cragnolino, 2001a,b) is not considered. Furthermore, the possible increase of hydrogen uptake by Titanium Grade 7 in the presence of fluoride leading to enhanced susceptibility to hydride-induced cracking has not been evaluated. To address these concerns, DOE agreed[28] to provide sufficient information on detailed fabrication processes and credible environmental conditions, including the composition of the contacting water. In particular, DOE agreed to address the potential detrimental effect of fluoride anions leading to accelerating the drip shield dissolution and hydrogen uptake/hydride cracking of drip shields from the accelerated dissolution.

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

3.3.1.4.2.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the degradation of engineered barriers (degradation of the drip shield) with respect to data uncertainty being characterized and propagated through the model abstraction.

As is the case of the waste package outer container, the most important implication of data uncertainty is related to the estimation of the distribution of drip shield failure times. It should be noted that the maximum error in the determination of corrosion rates from weight-loss measurements in the case of titanium alloy is more than two times that of Alloy 22. The difference can be attributed mostly to differences in density. The main source of uncertainties, however, is related to variation in environmental conditions promoting accelerated corrosion rates.

Though considerable data have been obtained examining the possibility and rates associated with uniform and localized corrosion, several areas of uncertainty still exist. The low corrosion rates measured from weight-loss experiments need to be confirmed with other tests designed to sensitively measure the passive corrosion rate. This confirmation is particularly important because it appears there is an inconsistency between the analysis and model report (CRWMS M&O, 2000t) and the process model report (CRWMS M&O, 2000b). This analysis and model report claims that the weight-loss measurements are at or below the reliable detection limit, yet these values are used for life prediction purposes in the process model report. To address this concern, DOE agreed[29] to provide sufficient data on the uniform corrosion from alternative test methods.

In addition, uncertainties related to the presence of fluoride in the waters contacting the drip shield can lead to much higher rates of uniform corrosion that, in turn, can result in higher absorption rates of hydrogen by the titanium alloys. In this case, the propagation of data uncertainty can affect the evaluation of the potential occurrence of delayed hydrogen cracking as a coupled failure mode. To address this concern, DOE agreed[30] to provide sufficient information on the fluoride concentration of the groundwater in contact with drip shields and its effects on accelerated drip shield corrosion and hydrogen uptake/hydride cracking.

Error propagation from data uncertainties was considered to originate mainly from variations in environmental conditions, low sensitivity in the measurement of uniform corrosion, and possible acceleration of uniform corrosion and hydride embrittlement in the presence of fluoride ions.

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

[30]Ibid.

In summary, as noted previously, DOE agreed to provide the needed information before any future license application is submitted.

3.3.1.4.2.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the degradation of engineered barriers (degradation of the drip shield) with respect to model uncertainty being characterized and propagated through the model abstraction.

The corrosion rates measured (approximately 10 to a few hundreds of nanometers per year) using weight-loss methods, especially given the uncertainties concerning cleaning procedures, may be unreliable and nonconservative. Furthermore, in the analysis and model report (CRWMS M&O, 2000t) it was concluded that the majority of the weight-loss measurements during coupon exposure tests were at or below the level of detection. Based on electrochemical corrosion tests, much higher passive dissolution rates were observed (at least a factor of 30 times greater and, in some cases, more than 400 times greater), which could lead to a more conservative estimate of the drip shield life. DOE has not considered alternative models for general passive corrosion. The model used is empirical and based only on the experimental determination of corrosion rates (CRWMS M&O, 2000t). As a result, data uncertainty (note the elimination of data exhibiting weight gain) may render model validation unreliable affecting the confidence to predict life for thousands of years. To address this concern, DOE agreed[31] to provide sufficient data on uniform corrosion from more sensitive and alternative test methods.

This issue is also important in relation to the mechanical disruption of the engineered barriers integrated subissue as described in Section 3.3.2. The effect of rockfall calculations on mechanical failure of the drip shield will be affected by consideration of the drip shield wall thinning because of uniform corrosion and simultaneous hydrogen absorption leading to hydride precipitation and embrittlement of titanium alloys. To address this concern, DOE agreed[32] to provide sufficient information on the effect of wall thinning from corrosion and hydride embrittlement on the mechanical failure induced by rockfall.

The rates of DOE drip shield uniform corrosion are neither consistent among different test methods nor consider alternative models. The inaccurate assessment of uniform corrosion rate will lead to the inaccurate prediction of the drip shield mechanical failure by the thinning of the drip shield wall with the impact of rockfalls.

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

[32]Ibid.

In summary, as noted previously, DOE agreed to provide the needed information before any future license application is submitted.

3.3.1.4.2.5 Model Abstraction Output Is Supported by Objective Comparisons

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess the degradation of engineered barriers (degradation of the drip shield) with respect to model abstraction output being supported by objective comparisons.

Though not considered important by DOE, thermal embrittlement of titanium alloys has been reported based on thermally driven redistribution of nearly insoluble impurities from grain interiors to grain boundaries (Nesterova, et al., 1980). This redistribution results in embrittlement of the material with negligible change in strength (though wide variations in ductility are observed) and increased intergranular fracture. Such segregation tends to result in precipitation of finely dispersed particles at the grain boundaries. For commercial purity titanium and α-titanium alloys that contain nickel and iron as impurities, these precipitates have been identified as $Ti_2(Fe,Ni)$. Embrittlement has been noted at temperatures as low as 350 °C [662 °F] after 500 hours. The possibility of embrittlement at lower temperatures when exposed for longer periods has not been examined, however. DOE abstraction analyses of hydrogen embrittlement of titanium alloys could be used to capture any possible effects of thermal embrittlement on predicted drip shield life expectancy. DOE agreed[33] to address this concern and needs to include detailed clarifications stated here in the agreed-on information on hydride embrittlement.

Of possibly greater importance is the lack of experimental work examining the possible detrimental effects of fluoride on the corrosion behavior of titanium. Though fluoride was present in some test environments at low levels, the presence of other species, such as calcium and silicon, may have limited the concentration of free fluoride available for complexation with titanium (Schutz and Grauman, 1985) and masked the evaluation of any accelerating effect of fluoride. To address this concern, DOE agreed[34] to provide sufficient information on the fluoride concentration of the groundwater in contact with drip shields and its potential effect on corrosion.

From the perspective of localized corrosion, though little or no localized corrosion has been observed thus far, the localized corrosion behavior of titanium-palladium alloys has not been extensively studied. It has been observed that, under relatively aggressive conditions, these materials still exhibit high crevice corrosion resistance (Brossia and Cragnolino, 2001a,b). In the presence of fluoride, however, significant attack has been reported, and, in fact, some crevice corrosion in chloride-fluoride environments has been observed (Brossia and Cragnolino,

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

[34]Ibid.

2001a). In addition, the possible detrimental effects of fabrication methods, such as weldments, have not been evaluated and further evaluation should be provided once the design has been finalized. To address this concern, DOE agreed[35] to provide sufficient data and rationales in assessing the susceptibility of drip shields to localized corrosion.

Environmentally assisted cracking of titanium-palladium alloys has not been extensively examined. As noted, it is generally accepted that environmentally assisted cracking of titanium alloys occurs through a hydrogen embrittlement-type mechanism probably related to hydride precipitation and cracking. DOE, however, considers stress corrosion cracking and hydride-induced cracking to be separate mechanisms. In fact, DOE even is considering two possible models for stress corrosion cracking (threshold stress intensity and slip-film dissolution). It is unclear how these stress corrosion cracking models fit into the generally accepted mechanistic understanding of hydrogen-embrittlement-based environmentally assisted cracking of titanium alloys. DOE should clarify if it plans to use these models to predict environmentally assisted cracking of the Titanium Grade 7 drip shield. With regard to hydride-induced cracking of the drip shield, DOE's recent change to use the minimum hydrogen concentration necessary for hydride-induced cracking based on limited experimental work using Titanium Grade16 (CRWMS M&O, 2000r) may be more realistic but less conservative than the previous efforts using the values for commercial purity titanium. Given the relative lack of data in this area on titanium-palladium alloys and the uncertainty surrounding the calculations, a more conservative approach may be more adequate. To address this concern, DOE agreed[36] to provide sufficient data and rationales for the possibility of drip shield stress corrosion cracking.

Additional technical bases for the fraction of hydrogen absorbed by titanium during corrosion processes have been provided (CRWMS M&O, 2000s). The effects that palladium may have on this value should be evaluated further, especially given the catalytic effects of palladium on hydrogen generation and the reported increases in absorbed hydrogen at constant corrosion rates for palladium-bearing alloys compared with nonpalladium-titanium alloys (Fukuzuka, et al., 1980). The technical basis for the fraction of hydrogen absorbed, especially considering the well-known catalytic properties of palladium for hydrogen generation, however, needs to be strengthened. In addition, reliance on the passive corrosion rates measured from weight loss coupons may lead to a nonconservative estimate of the quantity of hydrogen absorbed. This estimate suggests that hydride-induced cracking of titanium may occur during anticipated repository conditions. It is suggested DOE examine the possibility of enhanced hydrogen uptake and absorption in the palladium-bearing titanium alloys, especially Grade 7 rather than Grade 16, because the differences in the palladium content of these materials could make a difference in the measured hydrogen uptake rates. The possibility of enhanced hydrogen update in the presence of fluoride through destabilization of the TiO_2 oxide should be evaluated also. It is recommended that DOE confirms the low corrosion rates measured from weight-loss experiments and from polarization data with long-term electrochemical tests or other techniques

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

[36]Ibid.

designed to sensitively measure the passive corrosion rate. To address these concerns, DOE agreed[37] to provide sufficient data and rationales for the efficiency of hydrogen uptake along with the sensitive measurement of associated corrosion rates.

The belief that stress corrosion cracking and hydride-induced cracking of the drip shield have low consequences because of presumed crack plugging by corrosion or calciferous deposits should be reevaluated further. Though it may be possible that any cracks forming on the drip shield eventually will be plugged such that no water transport through the crack is possible, the consequence of the crack presence on subsequent rockfall events should be examined. In such cases, it might be envisioned that an existing crack acts as the nucleation point for a substantial opening in the drip shield. To address this concern, DOE agreed[38] to provide sufficient information on the potential effects of crack plugging by corrosion or by calciferous deposits on the further development of stress corrosion cracking.

Sufficient data and rationales are required for the verification of the model abstraction in the drip shield performance. Thermal embrittlement may occur by the formation of secondary phases. The accurate assessment of fluoride ion concentration on the drip shield surface may exclude fluoride-induced fast drip shield corrosion or hydride embrittlement. The likelihood of drip shield susceptibility to localized corrosion needs to be better assessed, especially with respect to drip shield fabrication. The DOE assessment of the environmentally assisted cracking of drip shields is unclear regarding critical hydrogen concentration and the hydrogen uptake process, and the proposed mechanism for crack plugging by corrosion or calciferous deposits as means for crack arrest.

In summary, as noted previously, DOE agreed to provide the needed information before any future license application is submitted.

3.3.1.4.3 Criticality Within the Waste Package

DOE screened the occurrence of nuclear criticality for commercial spent fuel, normal conditions, and seismic events from the Total System Performance Assessment–Site Recommendation based on the lack of waste package breach or failure at any time during the first 10,000 years of postclosure (CRWMS M&O, 2000u). For igneous events, DOE screened the occurrence of criticality based on a low probability of formation of a critical configuration. The basis for this screening has been documented in CRWMS M&O (2000v). NRC concerns regarding the DOE screening argument for nuclear criticality are discussed in Section 3.2.2. Identification of Events with Probabilities Greater Than 10^{-8} Per Year. Per an agreement made

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

[38]Ibid.

during the DOE and NRC Technical Exchange on Criticality,[39] DOE committed to performing a what-if analysis, using the topical report approach, which would simulate the consequence of a criticality event. Discussion of criticality in the following sections relates to the topical report DOE developed to describe the methodology that will be used to assess the probability and consequences of an in-package criticality event within the repository system (DOE, 1998b). NRC reviewed this topical report and documented the results in a safety evaluation report (NRC, 2000d). The safety evaluation report contains 28 open items on the methodology, which, when closed, will document NRC acceptance of the proposed methodology to address criticality in the repository system. Per an agreement made during the DOE and NRC Technical Exchange on Criticality, DOE provided NRC with Revision 1 of this topical report, intended to address 27 of these open items (DOE, 2000). In an NRC letter dated December 10, 2001, NRC stated it accepted Revision 01 of the topical report for detailed technical review. It is expected that the NRC review will be completed by the end of 2002. If this new revision of the topical report is found acceptable, it will provide confidence that DOE will be able to address the effects of criticality on the performance of the repository system in any potential license application even if DOE is unable to support arguments for screening criticality from the total system performance assessment. The remaining open item on burnup measurements was discussed at the DOE and NRC Technical Exchange on Preclosure Safety.[40]

3.3.1.4.3.1 System Description and Model Integration Are Adequate

Overall, the current information, along with the agreements reached between the DOE and NRC (Section 3.3.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess degradation of engineered barriers (criticality within the waste package) with respect to system description and model integration.

The open items associated with the DOE topical report on criticality include many issues related to the in-package criticality model: (i) development of a modeling approach for igneous-activity-induced criticality, (ii) losses of radionuclides from intact assemblies through pinholes and cracks in the cladding and establishment of the uncertainty associated with this loss, (iii) inclusion of a criticality margin, (iv) cross-dependency of configuration parameters for k_{eff} regression equations, (v) provision of a multi-parameter approach in bias-trending analyses, (vi) defense of method used for extending trends, (vii) development of a methodology to determine steady-state criticality consequences for nonaqueous moderators, (viii) addition of consequences other than radionuclide inventory increase to the steady-state criticality consequence model, (ix) description of the interface between the criticality topical report analyses and the total system performance assessment criticality risk analysis, and (x) physical verification of burnup levels of spent nuclear fuel (NRC, 2000d).

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[39]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Key Technical Issue: Criticality (October 23–24, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[40]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Key Technical Issue: Pre-Closure Safety (July 24–26, 2001)." Letter (August 14) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

As noted above, if the new revision of the topical report is found acceptable, it will provide confidence that DOE will be able to address the effects of criticality on the performance of the repository system in any potential license application even if DOE is unable to support arguments for screening criticality from the total system performance assessment.

3.3.1.4.3.2 Data Are Sufficient for Model Justification

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess degradation of engineered barriers (criticality within the waste package) with respect to data being sufficient for model justification.

The open items associated with the DOE topical report on criticality include several issues related to the sufficiency of data supporting the in-package criticality model, including the DOE needs to use cross-sectional data at the temperature of the waste package or critical benchmarks and DOE must provide justification for the correction factors developed for boron remaining in solution (NRC, 2000d). In the DOE and NRC Technical Exchange on Criticality, DOE indicated that examples of data that would be used in the criticality analyses for the quantity and alternative forms of corrosion products in the waste package and radionuclide release from small cracks in cladding could be found in several reports (CRWMS M&O, 1998b, 1999b,c, 2000w,x,y; Wilson, 1990). Additionally, DOE indicated that additional data would be located in the validation reports for the inventory, neutronics, and geochemistry computer codes that will be used in the criticality modeling. DOE agreed [41] to provide these validation reports to NRC before submitting the license application for the proposed Yucca Mountain repository.

As noted previously, if the new revision of the topical report is found acceptable, it will provide confidence that DOE will be able to address the effects of criticality on the performance of the repository system in any potential license application even if DOE is unable to support arguments for screening criticality from the total system performance assessment.

3.3.1.4.3.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess degradation of engineered barriers (criticality within the waste package) with respect to data uncertainty being characterized and propagated through the model abstraction.

The open items associated with the DOE topical report on criticality include several issues related to the assessment of data uncertainty in the in-package criticality model: (i) DOE needs

[41]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Key Technical Issue: Criticality (October 23–24, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

to account for bias and uncertainty in the isotopic depletion model, (ii) DOE must account for all types of uncertainty and bias in the criticality analysis, (iii) DOE must include the isotopic bias and uncertainty in developing the critical limit, and (iv) DOE must include uncertainty introduced by the use of a regression equation and look-up tables (NRC, 2000d). In the DOE and NRC Technical Exchange on Criticality, DOE indicated that examples of the consideration of data uncertainty that would be used in the criticality analyses for the quantity and alternative forms of corrosion products in the waste package and radionuclide release from small cracks in cladding could be found in several reports (CRWMS M&O, 1998b, 1999b,c, 2000w,x,y; Wilson, 1990). Additionally, DOE indicated that quantification of data uncertainty would be located in the validation reports for the inventory, neutronics, and geochemistry computer codes that will be used in the criticality modeling. DOE agreed [42] to provide these validation reports to NRC before submitting the license application for the proposed Yucca Mountain repository.

As noted previously, if the new revision of the topical report is found acceptable, it will provide confidence that DOE will be able to address the effects of criticality on the performance of the repository system in any potential license application even if DOE is unable to support arguments for screening criticality from the total system performance assessment.

3.3.1.4.3.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess degradation of engineered barriers (criticality within the waste package) with respect to model uncertainty being characterized and propagated through the model abstraction.

The open items associated with the DOE topical report on criticality include one issue related to the assessment of model uncertainty in the in-package criticality model, demonstrating the adequacy of using a one-dimensional point-depletion calculation in the depletion analyses instead of two- or three-dimensional models (NRC, 2000d). In the DOE and NRC Technical Exchange on Criticality, DOE indicated that the validation reports will support the inventory computer code in the criticality modeling. DOE agreed [43] to provide these validation reports to NRC before submitting the license application for the proposed Yucca Mountain repository.

As noted previously, if the new revision of the topical report is found acceptable, it will provide confidence that DOE will be able to address the effects of criticality on the performance of the repository system in any potential license application even if DOE is unable to support arguments for screening criticality from the total system performance assessment.

[42]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Key Technical Issue: Criticality (October 23–24, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[43]Ibid.

3.3.1.4.3.5 Model Abstraction Output Is Supported by Objective Comparisons

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.1.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess degradation of engineered barriers (criticality within the waste package) with respect to model abstraction output being supported by objective comparisons.

Open items associated with the DOE topical report on criticality include issues related to the support of models used in the in-package criticality model, including DOE must validate the regression equation or look-up table for all ranges of configurations and waste form parameters affecting k_{eff} and that DOE needs to develop a validation approach for the power model for steady-state criticality consequences (NRC, 2000d). In the DOE and NRC Technical Exchange on Criticality, DOE indicated that the justification of the models used in the criticality analyses would be located in the validation reports for the inventory, neutronics, and geochemistry computer codes. DOE agreed[44] to provide these validation reports to NRC before submitting the license application for the proposed Yucca Mountain repository.

As noted previously, if the new revision of the topical report is found acceptable, it will provide confidence that DOE will be able to address the effects of criticality on the performance of the repository system in any potential license application even if DOE is unable to support arguments for screening criticality from the total system performance assessment.

3.3.1.5 Status and Path Forward

Table 3.3.1-1 provides the status of all key technical issue subissues, referenced in Section 3.3.1.2, for the Degradation of Engineered Barriers Integrated Subissue. The table also provides the related DOE and NRC agreements pertaining to the Degradation of Engineered Barriers Integrated Subissue. The agreements listed in the table are associated with one or all five generic acceptance criteria discussed in Section 3.3.1.4. Note that the status and the detailed agreements (or path forward) pertaining to all the key technical issue subissues are provided in Table 1.1-3 and Appendix A.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (through specified testing, analyses, and the like), acceptably addresses the NRC questions so that no information beyond that provided, or agreed to, will likely be required at the time of a potential license application.

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[44]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Key Technical Issue: Criticality (October 23–24, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

Table 3.3.1-3. Related Key Technical Issue Subissues and Agreements			
Key Technical Issue	**Subissue**	**Status**	**Related Agreements***
Container Life and Source Term	Subissue 1—The Effects of Corrosion Processes on the Lifetime of Containers	Closed-Pending	CLST.1.01 through CLST.1.17
	Subissue 2—The Effects of Phase Instability of Materials and Initial Defects on Mechanical Failure and Lifetime of Containers	Closed-Pending	CLST.2.04 through CLST.2.08
	Subissue 5—The Effect of In-Package Criticality on Waste Package and Engineered Barrier Subsystem Performance	Closed-Pending	CLST.5.01 CLST.5.03 through CLST.5.07
	Subissue 6—The Effects of Alternate Engineered Barrier Subsystem Design Features on Container Lifetime and Radionuclide Release from the Engineered Barrier Subsystem	Closed-Pending	CLST.6.01 through CLST.6.04
Thermal Effects on Flow	Subissue 2—Thermal Effects on Temperature, Humidity, Saturation, and Flux	Closed-Pending	TEF.2.03 TEF.2.04 TEF.2.09
Evolution of the Near-Field Environment	Subissue 2—Effects of Coupled Thermal-Hydrologic-Chemical Processes on Waste Package Chemical Environment	Closed-Pending	ENFE.2.04 ENFE.2.14
	Subissue 3—Effects of Coupled Thermal-Hydrologic-Chemical Processes on Chemical Environment for Radionuclide Release	Closed-Pending	ENFE.3.01
	Subissue 5—Effects of Coupled Thermal-Hydrologic-Chemical Processes on Potential Nuclear Criticality in the Near-Field	Closed-Pending	ENFE.5.03
Repository Design and Thermal-Mechanical Effects	Subissue 3—Thermal-mechanical Effects on Underground Facility Design and Performance	Closed-Pending	RDTME.3.18
Total System Performance Assessment and Integration	Subissue 1—System Description and Demonstration of Multiple Barriers	Closed-Pending	None
	Subissue 2—Scenario Analysis and Event Probability	Closed-Pending	TSPAI.2.01 TSPAI.2.02 TSPAI.2.04

Table 3.3.1-3. Related Key Technical Issue Subissues and Agreements (continued)			
Key Technical Issue	Subissue	Status	Related Agreements*
Total System Performance Assessment and Integration	Subissue 3—Model Abstraction	Closed-Pending	TSPAI.3.01 through TSPAI.3.05
	Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards	Closed-Pending	None
*Related DOE and NRC agreements are associated with one or all five generic acceptance criteria.			
Note: Key Technical Issue Agreement GEN.1.01 pertains to multiple integrated subissues, as well as some specific issues related to this integrated subissue.			

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3.3.2 Mechanical Disruption of Engineered Barriers

3.3.2.1 Description of Issue

The Mechanical Disruption of Engineered Barriers Integrated Subissue addresses the DOE total system performance assessment of engineered barriers subjected to mechanically disruptive events. Engineered barriers include the emplacement drift, waste package, waste package pallet, and drip shield and drift invert system. Although engineered backfill is not presently included in the engineered barrier subsystem design, it may be placed within the emplacement drifts of the proposed geologic repository for commercial spent nuclear fuel and high-level waste. If used, engineered backfill would also be assessed to determine how its performance characteristics and interactions with other engineered barrier subsystem components would be affected by mechanically disruptive events. The potential disruptive events to be addressed by the Mechanical Disruption of Engineered Barriers Integrated Subissue review are igneous intrusion, faulting, seismicity, rockfall and drift collapse, and criticality. The relationship between this integrated subissue to other integrated subissues is depicted in Figure 3.3.2-1. The overall organization and identification of all the integrated subissues is depicted in Figure 1.1.2.

The DOE description and technical bases for the analyses of mechanical disruption of engineered barriers model abstraction are documented in various process model reports, analysis and model reports, system description documents, and calculation reports. These documents, which are identified in the appropriate subsections that follow, are reviewed to the extent that they contain (i) process-level models, data, and analyses that support the abstracted models used by DOE in the total system performance assessment of the engineered barrier subsystem when subjected to mechanically disruptive events and (ii) screening arguments used to justify the exclusion of mechanical disruption of engineered barriers processes from consideration.

With the exception of igneous activity, DOE screened out all potential disruptive events from consideration of the repository total system performance assessment based on low-probability and low-consequence arguments. Igneous effects accounted for in the mechanical disruption of engineered barriers model abstraction are presently limited by DOE to interactions between basaltic magma and waste packages not located along a magma flow path to the surface. Waste package response to magma flowing to the surface (i.e., in the subvolcanic conduit) is evaluated as part of the Volcanic Disruption of Waste Packages Integrated Subissue. Key processes associated with the mechanical disruption of engineered barriers by igneous intrusion are (i) basaltic magma flows into proposed repository drifts, (ii) engineered barrier component response to basaltic magma exposure, and (iii) cooling of the basalt and engineered barrier subsystem, allowing reestablishment of long-term hydrologic transport processes.

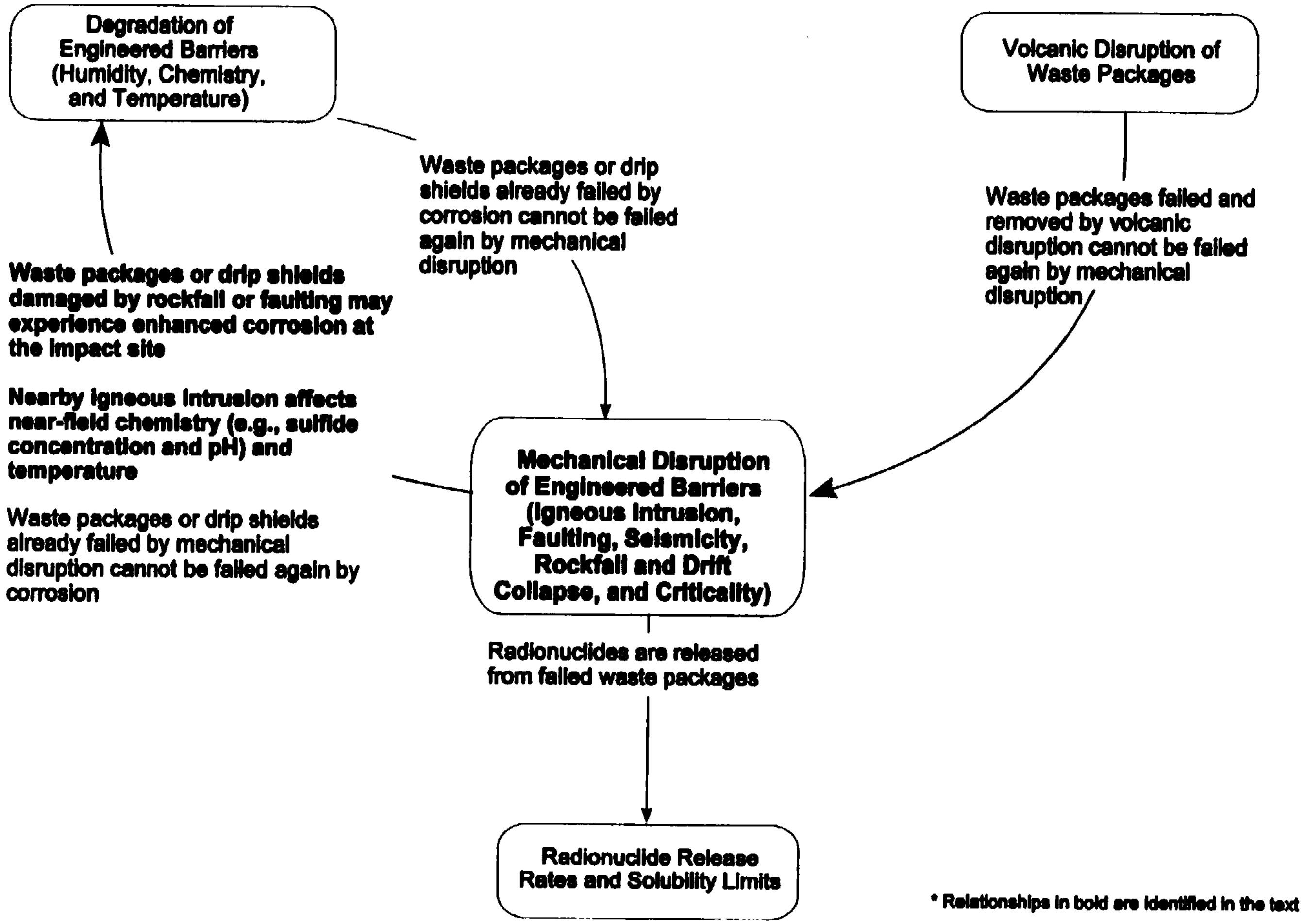

Figure 3.3.2-1. Diagram Illustrating the Relationship Between Mechanical Disruption of Engineered Barriers and Other Integrated Subissues

3.3.2.2 Relationship to Key Technical Issue Subissues

The Mechanical Disruption of Engineered Barriers Integrated Subissue incorporates subject matter previously captured in the following key technical issue subissues.

- Container Life and Source Term: Subissue 1—Effects of Corrosion Processes on the Lifetime of the Containers (NRC, 2001)
- Container Life and Source Term: Subissue 2—Effects of Phase Instability of Materials and Initial Defects on the Mechanical Failure and Lifetime of the Containers (NRC, 2001)
- Container Life and Source Term: Subissue 5—Effect of In-Package Criticality on Waste Package and Engineered Barrier Subsystem Performance (NRC, 2001)
- Container Life and Source Term: Subissue 6—Effect of Alternate Engineered Barrier Subsystem Design Features on Container Lifetime and Radionuclide Release from the Engineered Barrier Subsystem (NRC, 2001)
- Igneous Activity: Subissue 1—Probability of Future Activity (NRC, 1999a)
- Igneous Activity: Subissue 2—Consequences of Igneous Activity (NRC, 1999a)
- Repository Design and Thermal Mechanical Effects: Subissue 1—Implementation of an Effective Design Control Process within the Overall Quality Assurance Program (NRC, 2000a)
- Repository Design and Thermal Mechanical Effects: Subissue 2—Design of the Geologic Repository Operations Area for the Effects of Seismic Events and Direct Fault Disruption (NRC, 2000a)
- Repository Design and Thermal Mechanical Effects: Subissue 3—Thermal-Mechanical Effects on Underground Facility Design and Performance (NRC, 2000a)
- Structural Deformation and Seismicity: Subissue 1—Faulting (NRC, 1999b)
- Structural Deformation and Seismicity: Subissue 2—Seismicity (NRC, 1999b)
- Structural Deformation and Seismicity: Subissue 3—Fracturing and Structural Framework of the Geologic Setting (NRC, 1999b)
- Structural Deformation and Seismicity: Subissue 4—Tectonic Framework of the Geologic Setting (NRC, 1999b)
- Total System Performance Assessment and Integration: Subissue 1—System Description and Documentation of Multiple Barriers (NRC, 2000b)

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- Total System Performance Assessment and Integration: Subissue 2—Scenario Analysis and Event Probability (NRC, 2000b)
- Total System Performance Assessment and Integration: Subissue 3—Model Abstraction (NRC, 2000b)
- Total System Performance Assessment and Integration: Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards (NRC, 2000b)

The key technical issue subissues formed the bases for the previous version of the issue resolution status reports and also were the bases for technical exchanges with DOE where agreements were reached on what additional information DOE needed to provide to resolve the subissues. The resolution status of this integrated subissue is based on the resolution status of each of the contributing key technical issue subissues. The subsequent sections incorporate applicable portions of these key technical issue subissues, however, no effort was made to explicitly identify each subissue.

3.3.2.3 Importance to Postclosure Performance

One aspect of risk informing the NRC review was to determine how this integrated subissue is related to the DOE repository safety strategy. Specifically, the DOE Repository Safety Strategy (CRWMS M&O, 2000a) acknowledges that mechanical disruption of engineered barriers will affect the long-term risks of the proposed repository to the public health and safety. Both the performance of the waste package and that of the drip shield and drift invert system are listed among the eight principal factors for the postclosure safety case (CRWMS M&O, 2000a).

The Yucca Mountain area, which lies within the Basin and Range tectonic province of the western Cordillera, has been seismically, tectonically, and volcanically active on the timescale of a geologic repository. Future seismotectonic and volcanic activities could affect the stability of both the natural and engineered barrier subsystems of the repository.

The Total System Performance Assessment–Site Recommendation (CRWMS M&O, 2000b) reports no radiological risk in 10,000 years from the basecase repository. Based on the DOE analyses, intrusive igneous activity has a probability weighted risk of approximately 1 μSv/yr [0.1 mrem/yr] and is classified by DOE as a principle factor (CRWMS M&O, 2000b). This risk value increases by approximately one order of magnitude when a probability value of 1×10^{-7} (NRC, 1999a) is used. DOE agreed[1] to include, for its licensing case, the results of a single-point sensitivity analysis for extrusive and intrusive igneous processes at 1×10^{-7}. In a later DOE analysis, the risk from intrusive igneous activity decreased by approximately one order of magnitude to 0.1 μSv/yr [0.01 mrem/yr] in Bechtel SAIC Company, LLC (2001a), based primarily on changes to radionuclide solubility and transport models. With the exception of

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[1]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (August 29–31, 2000)." Letter (October 23) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

volcanism, this level of intrusive risk clearly exceeds calculated risks from other postclosure features, events, and processes in Bechtel SAIC Company, LLC (2001a,b).

Staff raised concerns with the technical bases used by DOE to evaluate both extrusive and intrusive igneous activities in the Total System Performance Assessment–Site Recommendation[2] (Hill and Connor, 2000). Analyses presented in, for example, NRC (1999a) also demonstrate that probability-weighted risk from postclosure volcanism may be on the order of 10 μSv/yr [1 mrem/yr], with significant uncertainties associated with this value. Further, processes of magma-repository-waste package interactions affect the amount of radionuclide potentially released by groundwater pathways. This interaction directly controls the amount and character of high-level waste potentially available for subsequent hydrologic transport (see Section 3.3.2.4.1 for detailed discussion).

Because postclosure performance requirements rely on continued functionality of the waste package and drip shield and drift invert system, DOE committed to designing these engineered barrier subsystem components to withstand the effects of vibratory ground motion caused by earthquakes and the potential loads arising from drift degradation (i.e., rock block impacts and drift collapse). Although the engineered barrier subsystem design has yet to be finalized, DOE screened out nearly all the primary and secondary features, events, and processes pertaining to vibratory ground motion and drift degradation from consideration in the Total System Performance Assessment Code based on the aforementioned design commitment. The only features, events, and processes pertaining to seismic and drift degradation loads accounted for in the DOE Total System Performance Assessment Code is the potential failure of the commercial spent nuclear fuel cladding caused by vibratory ground motion (CRWMS M&O, 2000c). This scenario is included in the total system performance assessment basecase, and DOE concluded that seismically induced cladding failure does not contribute to dose (CRWMS M&O, 2000c) because the waste packages will remain intact for the entire regulatory period regardless of any cladding failures. The staff reviewed the DOE cladding failure analyses and identified several deficiencies (see Section 3.3.4.4.3).

Criticality is also included in the Mechanical Disruption of Engineered Barriers Integrated Subissue discussion for two reasons. The first reason is an in-package criticality event may cause significant mechanical degradation or outright failure of the waste form and waste package. The second reason is a criticality event could be initiated as a result of another, unrelated mechanically disruptive event (e.g., rockfall). For the second case, the extent of the damage caused by the original disruptive event could be significantly magnified if criticality were to occur as a related consequence.

3.3.2.4 Technical Basis

NRC developed a plan (2002) consistent with acceptable criteria and review methods found in previous issue resolution status reports. A review of the DOE approach for including

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[2]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (August 29–31, 2000)." Letter (October 23) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

mechanical disruption of engineered barriers in the total system performance assessment abstraction is provided in the following subsections. For the sake of clarity, the technical basis for the staff comments will be presented within individual subsections for each mechanically disruptive event being reviewed (i.e., igneous intrusion, faulting, seismicity, rockfall and drift collapse, and criticality). Each of these subsections, in turn, have been subdivided and organized according to the five acceptance criteria identified in Section 1.5: (i) System Description and Model Integration Are Adequate, (ii) Data Are Sufficient for Model Justification, (iii) Data Uncertainty Is Characterized and Propagated Through the Model Abstraction, (iv) Model Uncertainty Is Characterized and Propagated Through the Model Abstraction, and (v) Model Abstraction Output Is Supported by Objective Comparisons.

3.3.2.4.1 Igneous Intrusion

3.3.2.4.1.1 System Description and Model Integration Are Adequate

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.2.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess mechanical disruption of engineered barriers (i.e., igneous intrusion) with respect to system description and model integration.

This subsection provides a review of the system description and model integration of the DOE igneous intrusion abstraction for the Total System Performance Assessment–Site Recommendation (CRWMS M&O, 2000b). The DOE description and technical basis for the igneous intrusion abstraction are documented in CRWMS M&O (2000d) and three supporting analysis and model reports (CRWMS M&O, 2000e–g). Calculation report documents (CRWMS M&O, 2000h,i) also provide information relevant to this review.

The DOE approach to evaluating igneous disruption of waste packages involves several conceptual models. Models for magma ascent and initial interaction with proposed repository drifts are discussed in Section 3.3.10 of this report. For the mechanical disruption of engineered barriers, the DOE models begin with the assumption that basaltic magma has flowed into all drifts directly intersected by an ascending dike [e.g., CRWMS M&O (2000e)].

DOE currently assumes only three waste packages on either side of an igneous intrusion (i.e., Zone 1) are damaged to the extent that the waste package provides no impediment for subsequent hydrologic flow and transport (CRWMS M&O, 2000e,f). Staff agree these models consider a sufficient range of interrelated processes to support this conclusion. The remaining waste packages in an intersected drift (i.e., Zone 2), however, have only limited damage resulting from end-cap failure caused by internal pressurization effects (CRWMS M&O, 2000f,h). Although the spent nuclear fuel cladding degraded completely in the Zone 2 waste packages because thermal effects, waste can be mobilized only by water circulation through a limited number of relatively small openings along the waste package lid. Consideration of the full range of physical conditions associated with igneous events would result in much more extensive damage to Zone 2 waste packages than currently modeled by

DOE. To address this concern, DOE agreed[3] to evaluate waste package performance for the duration of the igneous event if the model abstraction takes credit for engineered barriers providing delay in radionuclide release.

Simple calculations show affected waste packages will likely remain exposed to hot {temperatures approximately 1,100 °C [2,012 °F]} basaltic magma for at least 480 hours (NRC, 1999a; CRWMS M&O, 2000e). The yield stress of Alloy 22 decreases from 370 Mpa [54 ksi] at room temperature to 213 MPa [31 ksi] at 760 °C [1,400 °F] (Haynes International, 1988). Similarly, the ultimate tensile strength of the alloy decreases from 786 MPa [114 ksi] at room temperature to 524 MPa [76 ksi] at 760 °C [1,400 °F]. Although the mechanical property data at higher temperatures are not available from the alloy manufacturer literature, the yield stress and ultimate tensile strength at temperatures above 760 °C [1,400 °F] can be estimated (CRWMS M&O, 2000h). At 1,100 °C [2,012 °F], the ultimate tensile strength is estimated to be 226 MPa [33 ksi] (CRWMS M&O, 2000h), and the yield stress is estimated to be 91 MPa [13 ksi]. The ductility of the alloy is not a function of temperature in the range 25–760 °C [77–1,400 °F]. A marked decrease in ductility above 760 °C [1,400 °F] is not expected for this material. After exposure to temperatures in the range 600–900 °C [1,112–1,652 °F], Alloy 22 undergoes microstructural changes that can result in a significant reduction in ductility at subsequently lower temperatures (Summers, et al., 1999; Rebak, et al., 2000). The loss of ductility may increase the susceptibility of the material to mechanical failure as a result of rockfall or seismic events after the intrusive event. The mechanical properties used by DOE when assessing the potential damage that a waste package might incur as a result of interactions with magma (CRWMS M&O, 2000e,h) do not account for these rapidly induced aging effects, which will produce nonlinear trends in mechanical properties. In addition, Alloy 316 nuclear grade stainless steel, which is used to construct the waste package inner barrier, has approximately 30 percent greater thermal expansivity than materials analogous to Alloy 22 (American Society of Mechanical Engineers, 2001), which is used to construct the waste package outer barrier. For the current waste package design, which uses a narrow gap between the inner and outer barriers, these differences in thermal expansion will create tensile stresses in the waste package outer barrier when subjected to magmatic temperatures. Exposure to magmatic temperatures also causes significant gas pressures within the confines of the waste package (CRWMS M&O, 2000e,h). The combined effects of differential thermal expansion and internal gas pressurization should be considered when assessing waste package response to magmatic temperature exposure. In addition, CRWMS M&O (2000e) concludes that drifts intersected by a dike will be blocked at the ends and will fill with magma until fluid pressures are high enough to fracture the drift roof and allow ascent of basaltic magma to continue. Analyses in CRWMS M&O (2000e,h) have not evaluated waste package response to dynamic external pressures in the 3–7 MPa [435–1,015 psi] range as discussed in CRWMS M&O (2000e). To address these

[3]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (August 29–31, 2000)." Letter (October 23) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

concerns, DOE agreed[4] to evaluate waste package response to stresses from thermal and mechanical effects associated with exposure to basaltic magma. This evaluation will include (i) appropriate at-conditions waste package material strength properties and magma flow paths for the likely duration of an igneous event and (ii) aging effects on waste package material strength properties when exposed to basaltic magmatic conditions for the likely duration of an igneous event.

Analyses in CRWMS M&O (2000e,h) also do not consider potentially adverse high-temperature corrosion processes in response to magmatic degassing or contact with basaltic magma. Cooling or depressurized basaltic magma exsolves significant amounts of gas, which is dominantly water with subordinate amounts of carbon, sulfur, and fluorine species (CRWMS M&O, 2000g). Some fraction of the exsolved gas will likely flow into drift-wall fractures not sealed by magma because the air in these fractures are at pressures close to atmospheric pressure (e.g., Rousseau, et al., 1999). The remainder of the exsolved gasses will flow into available openings in nonintersected drifts, including potential voids in backfilled materials. By analogy with basaltic lava flows, degassing may occur for years, potentially decades, after the eruption has ceased. Although the model in CRWMS M&O (2000e) appears to overestimate gas flow, the report concludes that "... the volume of gas arriving at a container is not directly a limiting factor in corrosion." Corrosion of the waste packages and drip shields by magmatic gas, however, is not considered in subsequent models (CRWMS M&O, 2000b,f). This process may be potentially important because magmatic gasses could extend well beyond the boundaries of magma flow in the drifts if drift ends are not completely sealed (i.e., into Zone 3). The potential exists for accelerated degradation of waste packages and drip shields exposed to magmatically derived gases, even if the waste packages and drip shields are not in direct contact with basaltic magma, as in Zones 1 and 2. To address this concern, DOE agreed[5] to evaluate the response of Zone 3 waste packages, or waste packages covered by backfill or rockfall, if exposed to magmatic gasses at conditions appropriate for an igneous event.

Although CRWMS M&O (2000b) concludes no significant natural backfill should occur within 10,000 years, staff recognize the presence of natural or engineered backfill will affect the extent of magma flow into drifts. Limited intrusion into backfilled drifts, however, will still result in the rapid emplacement of some volume of basaltic magma. Some waste packages may be separated from direct contact with this emplaced magma by backfill or rubble. Nevertheless, during the igneous event, basaltic magma will cool against this material and degas. These processes will likely result in coupled thermal and chemical effects on some waste packages in backfill extending beyond Zone 1 of CRWMS M&O (2000e). An appropriate range of temperatures, pressures, and gas geochemical effects has not been evaluated for waste packages in backfilled drifts outlined in CRWMS M&O (2000e,i). The potential exists for accelerated degradation of waste packages and drip shields exposed to high temperatures and magmatically derived gases, even if the waste packages and drip shields are not in direct

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[4]Reamer C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[5]Ibid.

contact with basaltic magma in Zones 1 and 2. To address this concern, DOE agreed[6] to evaluate the response of Zone 3 waste packages, or waste packages covered by backfill or rockfall, if exposed to magmatic gasses at thermal conditions appropriate for an igneous event.

In summary, internal gas pressurization and differential thermal expansion at elevated temperatures, coupled with the large dynamic loads of the overlying magma, aging effects on mechanical strength, and adverse geochemical effects appear sufficient to breach currently proposed waste packages located in DOE Zone 2 during basaltic igneous events. There is insufficient technical bases to conclude that any barrier to subsequent hydrologic transport processes remains for waste packages in Zone 2. Models for basalt degassing also show that corrosion induced by exposure to magmatic gasses may extend beyond direct damage Zones 1 and 2 and could potentially affect all remaining waste packages in Zone 3 (CRWMS M&O, 2000e). If all waste packages in Zone 2 are wholly damaged, there is likely a one order-of-magnitude increase in the source term for subsequent hydrologic transport (CRWMS M&O, 2000e,f). This increase in source term may increase probability-weighted risk significantly above 10 μSv/yr [1 mrem/yr]. The current information and the agreements reached between DOE and NRC (Section 3.3.2.5) are sufficient to ensure the necessary information will be available at the time of a potential license application to address these concerns.

3.3.2.4.1.2 Data Are Sufficient for Model Justification

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.2.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess mechanical disruption of engineered barriers (i.e., igneous intrusion) with respect to sufficient data for model justification.

To support models for waste package resilience during igneous events, data are needed for proposed waste package alloys for the following conditions:

- Material strength properties at magmatic temperatures {i.e., around 1,100 °C [2,012 °F]} for dynamic load conditions appropriate for the potential duration of basaltic igneous events {i.e., recurring pressure variations on order of 0.1–10 MPa [14.5–1,450 psi]}

- Changes in waste package material properties caused by continued exposure to magmatic conditions for the likely duration of basaltic igneous events (i.e., time of exposure at least 500 hours)

- Geochemical effects on waste package properties from cooling and degassing magma in direct contact with waste packages and for waste packages located beyond the zone of direct magma contact

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[6]Reamer C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

Limited available data indicate internal gas pressurization and differential thermal expansion at beyond design temperatures, coupled with the dynamic load of the overlying magma, secondary phase precipitation, and potential geochemical effects, appear sufficient to breach currently proposed waste packages located in Zone 2 (CRWMS M&O, 2000d) during basaltic igneous events. In addition, gasses produced from cooling magma appear potentially corrosive to proposed waste package alloys (CRWMS M&O, 2000e). These gasses will likely affect long-term performance of waste packages located in Zone 3 (CRWMS M&O, 2000e). To address these concerns, DOE agreed[7] to evaluate waste package response to stresses from thermal-mechanical effects associated with exposure to basaltic magma. This evaluation will include (i) appropriate at-conditions waste package material strength properties and magma flow paths for the likely duration of an igneous event, (ii) aging effects on waste package material strength properties when exposed to basaltic magmatic conditions for the likely duration of an igneous event, and (iii) evaluation of the response of Zone 3 waste packages, or waste packages covered by backfill or rockfall, if exposed to magmatic gasses at conditions appropriate for an igneous event.

In summary, data used by DOE are insufficient to justify model conclusions for limited waste package damage in Zone 2 of an igneous event or to evaluate the extent of waste package degradation caused by magmatic degassing following an igneous event (e.g., CRWMS M&O, 2000e,h). In addition, currently available data (e.g., Summers, et al., 1999; Rebak, et al., 2000; Haynes International, 2001) do not evaluate conditions representative of basaltic igneous events. DOE plans to provide an additional evaluation of thermal-mechanical effects on waste package damage in an update to CRWMS M&O (2000e). The current information and the agreements reached between DOE and NRC (Section 3.3.2.5) are sufficient to ensure the necessary information will be available at the time of a potential license application to address these concerns.

3.3.2.4.1.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.2.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess mechanical disruption of engineered barriers (i.e., igneous intrusion) with respect to the characterization and propagation of data uncertainty through the model abstraction.

The number of waste packages directly intersected by a basaltic dike is calculated using a range of dike characteristics summarized in CRWMS M&O (2000a,g). Current total system performance assessment models sample a range of dike length and orientations and the number of dikes per igneous event. These parameter ranges appear reasonably consistent with the underlying technical basis (CRWMS M&O, 1996). Using simple geometric relationships, models then calculate the number of drifts intersected by each sampled dike

[7]Reamer C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

event. For each intersected drift, three waste packages on either side of the dike are assumed to fail on contact with basaltic magma (i.e., Zone 1), whereas the remaining waste packages in the drift (i.e., Zone 2) are assumed to have limited damage (CRWMS M&O, 2000e,g,h). The range sampled in CRWMS M&O (2000b) for the number of waste packages impacted in Zones 1 and 2 is the simple product of the number of drifts intersected per intrusive event and the number of waste packages within each defined geometric zone.

DOE performed a limited number of sensitivity calculations in the Total System Performance Assessment–Site Recommendation relative to mechanical disruption of engineered barriers (CRWMS M&O, 2000b). Small variations in the number of waste packages failed in Zone 1 (i.e., 108 at the 5th percentile, 219 at the 95th percentile) had about a factor of two variation in the probability-weighted dose. Based on this sensitivity, an order-of-magnitude increase in dose is likely for an order-of-magnitude increase in the number of waste packages wholly damaged during an intrusive igneous event. Varying the aperture of end-cap openings in Zone 2 packages from 3.5 to 30 cm^2 [0.54–4.7 in^2] had negligible effects on the probability-weighted dose (CRWMS M&O, 2000b). Large increases in the number of waste packages partially damaged in Zone 2 also had negligible effects on the probability-weighted dose.

Although the processes of magma-waste package interaction are highly complex, DOE developed a deterministic model for waste package damage (CRWMS M&O, 2000b,e,h). Uncertain parameter values, such as waste package material strength properties at sustained temperatures, are not sampled in these models. If DOE develops process-level models to evaluate waste package resilience to igneous events, data uncertainty will need to be evaluated. To address this concern, DOE agreed[8] to evaluate waste package response to stresses from thermal and mechanical effects associated with exposure to basaltic magma. This evaluation will include (i) appropriate at-conditions waste package material strength properties and magma flow paths for the likely duration of an igneous event and (ii) aging effects on waste package material strength properties when exposed to basaltic magmatic conditions for the likely duration of an igneous event.

3.3.2.4.1.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.2.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess mechanical disruption of engineered barriers (i.e., Igneous Intrusion) with respect to the characterization and propagation of model uncertainty through the model abstraction.

CRWMS M&O (2000b) presents several alternative conceptual models for magma flow into open or backfilled drifts. The performance implications of these alternative models, however, are not discussed. For example, CRWMS M&O (2000e) discusses multiple-flow modes that

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[8]Reamer C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

pyroclastic flows or liquid magma could follow that result in different rates and extent of magma interaction within and between proposed repository drifts. Only one of those models is evaluated within Total System Performance Assessment–Site Recommendation, which is for flow into and repressurization within each discretely intersected drift (CRWMS M&O, 2000b). A model is developed in CRWMS M&O (2000e) for evolution of potentially corrosive gasses from cooling basaltic magma. Although the gas-flow rate is probably overestimated in this model, the process of degassing-induced corrosion appears supportable based on this model (CRWMS M&O, 2000e). The potential effects of degassing-induced corrosion on waste package performance, however, are not evaluated. Calculations in CRWMS M&O (2000h) assume the waste package walls are a single metal alloy and, thus, do not evaluate the potential for differential thermal expansion or consider that waste packages will be subjected to igneous conditions for many hundreds of hours during the intrusive event. Each of these models has clear alternatives, such as the use of different composition alloys for canister walls and prolonged exposure to igneous conditions, which are expected to affect total system performance significantly. To address these concerns, DOE agreed[9] to evaluate waste package response to stresses from thermal and mechanical effects associated with exposure to basaltic magma. This evaluation will include (i) appropriate at-conditions waste package material strength properties and magma flow paths for the likely duration of an igneous event and (ii) aging effects on waste package material strength properties when exposed to basaltic magmatic conditions for the likely duration of an igneous event.

In summary, alternative conceptual models consistent with available information are not evaluated within the context of total system performance. Uncertainties with existing conceptual models are not quantified or discussed, and the potential effects of these uncertainties are not evaluated . The current information and the agreements reached between DOE and NRC (Section 3.3.2.5) are sufficient to ensure the necessary information will be available at the time of a potential license application to address these concerns.

3.3.2.4.1.5 Model Abstraction Output Is Supported by Objective Comparisons

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.2.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess mechanical disruption of engineered barriers (i.e., igneous intrusion) with respect to model abstraction output being supported by objective comparisons.

Models relevant to igneous effects on mechanical disruption of waste packages in CRWMS M&O (2000a,b,d–i) have not been compared to detailed process-level models, appropriate laboratory or field tests, or natural analogs. Models for the flow of magma into repository drifts (CRWMS M&O, 2000g) are critically dependent on sustaining a debris plug at the end of each intersected drift. The abstracted models used to calculate pressures in the magma-drift system will need to be supported in conjunction with an analysis of debris plug

[9]Reamer C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

strength before magma flow can be wholly restricted to within an intersected drift. Models that conclude only a limited extent of damage to Zone 2 waste packages will need significant support, including evaluation of an appropriate range of physical conditions and duration of conditions associated with basaltic igneous events. The potential effects of degassing-induced corrosion will also need to be evaluated and verified for all potentially impacted waste packages. Once potential inconsistencies between the abstracted models and comparative data are explained and quantified, the resulting uncertainties will need to be included in total system performance assessment model results.

To address these concerns, DOE agreed[10] to evaluate waste package response to stresses from thermal and mechanical effects associated with exposure to basaltic magma. In addition, DOE agreed to evaluate the response of Zone 3 waste packages, or waste packages covered by backfill or rockfall, if exposed to magmatic gasses at conditions appropriate for igneous events.

3.3.2.4.2 Faulting

3.3.2.4.2.1 System Description and Model Integration Are Adequate

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.2.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess mechanical disruption of engineered barriers (i.e., faulting) with respect to system description and model integration.

DOE excluded all effects of faulting from consideration in the Total System Performance Assessment–Site Recommendation based on the features, events, and processes analyses (CRWMS M&O, 2000c). The exclusion of features, events, and processes related to faulting is primarily based on DOE conclusions of low probability. DOE assumes design parameters can be used to screen features, events, and processes based on low probability if the repository design eliminates or alleviates the features, events, and processes (CRWMS M&O, 2000c, Assumption 5.2). For faulting, the design parameters are fault-setback distances. DOE will position emplacement drifts and waste packages away from faults with future fault slip potential. The setback distance will have to be enough to ensure that faulting will not impact the engineered components. The amount of setback was determined from mechanical and theoretical considerations of fault zone behavior (CRWMS M&O, 2000j).

Determination of appropriate design parameters for faulting, including setback distances, was derived using results from the DOE fault displacement hazard assessment. The probabilistic fault displacement hazard assessment was constructed through the expert elicitation used by DOE to develop a probabilistic seismic hazard analysis (CRWMS M&O, 1998; Stepp, et al., 2001). The expert elicitation results were based on the findings of six expert teams, each consisting of three geoscientists. Fault displacement analyses evaluates the

[10]Reamer C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Igneous Activity (September 5, 2001)." Letter (September 12) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

potential hazards of an active fault intersecting vital components of the engineered barrier subsystem, especially waste packages.

For this evaluation of faulting, principal and secondary (or distributed) faulting were considered (as defined in dePolo, et al., 1991). Principal faulting refers to displacement along the main fault zone responsible for the release of seismic energy (i.e., an earthquake). At Yucca Mountain, principal faulting is assumed to occur only along principal faults, mainly block-bounding faults like the Solitario Canyon and Paintbrush Canyon faults. In contrast, secondary or distributed faulting is defined as a rupture of smaller faults, such as the Ghost Dance fault, that occurs in response to the rupture in the vicinity of the principal fault. These two subsets of faults are not mutually exclusive. Faults capable of principal rupture also can undergo secondary faulting in response to faulting on another principal fault. Because principal and secondary faults pose a potential risk to repository performance, both types were considered by DOE. NRC (1999b) provided a review of the methodology used by the DOE expert elicitation to develop an appropriate probabilistic fault displacement hazard assessment.

Staff consider that DOE used conservative assumptions for estimating the probability of faulting and the associated effects on waste packages (NRC, 1999b). The current screening argument used by DOE to exclude faulting from the total system performance assessment in the features, events, and processes analyses and the inputs of fault displacement to the setback calculations (CRWMS M&O, 2000j), however, does not provide an adequate technical basis for staff to consider this subissue closed. In the screening, DOE (CRWMS M&O, 2000c, Assumption 5.5), assumes the median fault displacement values, rather than the mean values, are a more accurate predictor of faulting for low probability faulting events. Assumption 5.5 defined low probability events as those with annual probabilities less than 10^{-6} per year. To address this concern, DOE[11] agreed to provide the appropriate technical basis for use of the median or reevaluation of the features, events, and processes screening based on the mean values, according to the Structural Deformation and Seismicity Key Technical Issue Technical Exchange agreements.

3.3.2.4.2.2 Data Are Sufficient for Model Justification

Overall, the current information is sufficient to assess mechanical disruption of engineered barriers (i.e., faulting) with respect to sufficient data for model justification.

DOE adequately evaluated the nature and amount of faulting and the appropriate range of both principal and secondary faulting hazard sources within the repository block. In addition, DOE adequately determined fault geometry applicable to developing the probabilistic fault displacement hazard assessment. Given present knowledge, the DOE interpretations of faulting from surficial and underground mapping, as presented in CRWMS M&O (1998), are geologically consistent and reasonable.

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

The experts adequately noted faults as primary or secondary for probabilistic fault displacement hazard assessment. Some fault data taken by DOE from surface outcrops and from the exploratory studies facilities have been confirmed by independent checks by the NRC staff (NRC, 1999b). The variation of fault orientation data is within acceptable limits for normal geologic work. Field checks of fault locations, orientations, displacements, and other selected geometric features are generally in close agreement with the DOE observations and interpretations.

3.3.2.4.2.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.2.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess mechanical disruption of engineered barriers (i.e., faulting) with respect to the characterization and propagation of data uncertainty through the model abstraction.

DOE has not yet provided information needed to justify the probability distributions and bounding assumptions of faulting or reasonably to account for the associated uncertainties and variabilities. DOE developed models of faulting (CRWMS M&O, 2000j) based on a probabilistic fault displacement hazard assessment (CRWMS M&O, 1998; Stepp, et al., 2001). In those models, values for fault displacements for probabilities less than 10^{-6} annual exceedance per year are based on the median rather than the mean values from the probabilistic fault displacement hazard assessment curves (CRWMS M&O, 2000c, Assumption 5.5). As discussed in Section 3.3.2.4.2.1, use of the median rather than the mean values is not supported by sufficient technical basis (also see Section 3.2.2). To address this concern, DOE agreed[12] to provide the necessary information or use the mean in future analyses.

3.3.2.4.2.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

At the time this report was prepared, the effects of faulting were excluded from the total system performance assessment model abstraction for disruptive events. DOE agreed to address the NRC concerns, as discussed in Sections 3.3.2.4.2.1 and 3.3.2.4.2.3. Depending on the resolution of these concerns, the effects of faulting will be included or excluded from the total system performance assessment model abstraction for disruptive events.

3.3.2.4.2.5 Model Abstraction Output Is Supported by Objective Comparisons

At the time this report was prepared, the effects of faulting were excluded from the total system performance assessment model abstraction for disruptive events. DOE agreed to address the NRC concerns, as discussed in Sections 3.3.2.4.2.1 and 3.3.2.4.2.3. Depending on the

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

resolution of these concerns, the effects of faulting will be included or excluded from the total system performance assessment model abstraction for disruptive events.

3.3.2.4.3 Seismicity

3.3.2.4.3.1 System Description and Model Integration Are Adequate

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.2.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess mechanical disruption of engineered barriers (i.e., seismicity) with respect to system description and model integration.

The DOE calculation of seismic effects on the engineered barrier subsystem relies in part on the input seismic loads calculated from the DOE probabilistic seismic hazard analysis. The probabilistic seismic hazard analysis methodology has been identified by NRC in 10 CFR 100.23 as an appropriate approach to address uncertainties associated with ground motions. DOE outlined the methodology used for its probabilistic seismic hazard analysis in DOE (1994), which was accepted, in principle, by NRC.[13] The methodologies discussed in NRC (1997) also offer acceptable approaches for evaluating the probabilistic seismic hazard at Yucca Mountain. For postclosure performance, the seismic hazard curve is an important input parameter for assessing rockfall and drift collapse in the emplacement drifts because of earthquake-induced ground shaking.

See the discussion on Effect of Rockfall and Drift Collapse in Section 3.3.2.4.4.1 for the staff assessment of consequences to the engineered barrier subsystem components caused by seismic events. The following sections discuss the elements of a probabilistic seismic hazard analysis.

<u>Seismic Source Characterization</u>

DOE characterized seismic sources in CRWMS M&O (1998) and in Stepp, et al. (2001). In this analysis, DOE used six teams of experts. Each team consisted of three specialized geoscientists with expertise in either paleoseismology, Basin and Range structural geology, or Basin and Range seismology. To assess seismic sources, the teams mainly relied on information provided by the U.S. Geological Survey, DOE, and related Yucca Mountain studies augmented by published literature. In addition, the teams assembled for six workshops, at which the experts exchanged information on seismic sources and participated in additional discussions with other external experts. Details of the workshops are given in the probabilistic seismic hazard analysis final report (CRWMS M&O, 1998; Stepp, et al., 2001). Elicitation methodology and related issues are treated separately in Section 5.4, Expert Elicitation Acceptance Criteria.

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

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

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

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

Fault sources were identified by the expert teams from published U.S. Geological Survey and DOE maps and reports (U.S. Geological Survey, 1996; Piety, 1995; Anderson, et al., 1995a,b; Simons, et al., 1995), published scientific literature (Scott, 1990; Zhang, et al., 1990; Reheis and Dixon, 1996; Reheis and Sawyer, 1997), and CNWRA publications (Ferrill, et al., 1996; McKague, et al., 1996). In addition, the experts benefitted from detailed discussions at several of the probabilistic seismic hazard analysis workshops, in which summaries of fault sources and tectonic models were presented by project and external scientific experts. The expert teams also visited many of the sources during a field trip held as part of Probabilistic Seismic Hazard Analysis Workshop #3 (November 18–21, 1996).

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

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

In contrast to fault sources, areal sources represent areas of distributed or background seismicity in which no geologic or geophysical evidence can tie earthquakes to known faults. In this way, areal sources account for earthquakes that occur on unidentified or unidentifiable fault sources. Areal sources are typically developed to represent earthquakes with magnitudes that may not necessarily cause surface rupture.

In the DOE probabilistic seismic hazard analysis (CRWMS M&O, 1998; Stepp, et al., 2001), experts relied on empirical relationships that relate surface rupture to earthquake magnitude (e.g., Wells and Coppersmith, 1993, 1994; dePolo, 1994; U.S. Geological Survey, 1996; CRWMS M&O, 1998, Figure 4-11; Stepp, et al., 2001). Given these data, there is greater than an 80-percent probability that **M**6.5 earthquakes will rupture the surface, while there is less than a 20-percent chance that **M**5.5 earthquakes will rupture the surface.

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

Historic Seismicity: The DOE facilitation team provided a single earthquake catalog to the expert teams. This catalog was compiled from 12 regional catalogs (CRWMS M&O, 1998, p. G-2). The initial catalog contained 271,223 earthquakes of **M**0.5 and larger for the period 1868–1996. This initial catalog was modified in three ways. First, all the magnitudes were converted to moment magnitude ($\mathbf{M}_W$). 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.

Maximum Magnitude: The maximum magnitude earthquake is the largest earthquake that can be produced on a fault or in an areal source, regardless of its frequency of occurrence. For fault sources, the expert teams used empirical scaling relationships that relate maximum magnitude to the physical dimensions of the fault. Maximum magnitude was derived from fault length, rupture area, maximum surface displacement, and average surface displacement. In some cases, the expert teams modified their maximum magnitude estimate by considering slip rate as well as rupture dimensions following Anderson, et al. (1996). In addition, the experts considered rupture area and average slip on the fault to estimate seismic moment, which was then converted to maximum magnitude using the relationships in Hanks and Kanamori (1979). For areal sources, the experts estimated the maximum magnitude earthquake based on the largest fault in the areal source not explicitly modeled as a fault source. Alternatively, the experts relied on the empirical relationships that relate surface rupture to earthquake magnitude based on empirical data (e.g., Wells and Coppersmith, 1994; dePolo, 1994; U.S. Geological Survey, 1996; CRWMS M&O, 1998, Figure 4-11).

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

Earthquake Recurrence

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

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

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

Ground Motion Attenuation

In a probabilistic seismic hazard analysis, ground motion attenuation models (i.e., mathematical relationships between ground motion and earthquake magnitude, distance, site conditions, and style of faulting) are required to estimate the levels of ground motion that may occur at a site. An expert elicitation process was used (CRWMS M&O, 1998) to develop ground motion estimates for the Yucca Mountain probabilistic seismic hazard analysis. Because of the limited availability of sufficient strong motion data to develop robust empirical ground models specific to the regional and local geologic conditions at Yucca Mountain and the seismologic characteristics of nearby active faults, a group of ground motion experts convened to evaluate input for developing a probabilistic ground motion model specific to the regional conditions of the western Basin and Range, in proximity to Yucca Mountain. In the context of these circumstances, expert elicitation is reasonable and appropriate (NRC, 1997). In addition, an expert elicitation provides the opportunity to incorporate supplementary sources of information into the development of ground motion models such as expert interpretations of related and indirect information on strong ground motion.

In the Yucca Mountain probabilistic seismic hazard analysis, the experts were to provide input (i.e., data, scientific interpretations, and estimates of parameter uncertainties) as part of the development of a probabilistic ground motion attenuation model. Consistent with the overall approach in the probabilistic seismic hazard analysis, the probabilistic ground motion attenuation model includes estimates of aleatory and epistemic uncertainties in ground motion levels. The aleatory uncertainty quantifies the inherent or natural randomness of ground motions (e.g., variability not explained by the ground motion model). The aleatory or random uncertainty is a probabilistic variable that results from natural physical processes and is inherent to the unpredictable nature of future events. For example, the size, location, and time of the next earthquake and the details of the ground motion are examples of quantities considered aleatory. Aleatory uncertainty cannot be reduced by collecting additional data. Epistemic uncertainty quantifies the uncertainty associated with the estimate of model parameters that are the result of limited data and lack of knowledge about parameters such as the physical processes involved in fault rupture and its energy release properties and the resultant wave propagation characteristics. In the Yucca Mountain probabilistic seismic hazard analysis, a probabilistic ground motion model was developed by each of seven ground motion experts. In aggregate, the seven models were intended to represent (probabilistically) the current state of knowledge with regard to ground motions that can occur at the Yucca Mountain site because of earthquakes.

Elements of the Probabilistic Ground Motion Model

The probabilistic ground motion model used in the Yucca Mountain probabilistic seismic hazard analysis predicts aleatory and epistemic uncertainties in ground motion as a function of earthquake magnitude, source-site distance, and style of faulting. The model consists of the following the elements:

- Ground Motion—Mathematical relationship that defines the variation of the mean log (median) ground motion (denoted as μ) as a function of earthquake magnitude, source-site distance, and style of faulting. The relationship is defined by model coefficients derived from input provided by ground motion experts.

- Aleatory Model—The aleatory variability in ground motion is defined by a lognormal distribution whose parameters are a median (of 1.0) and a logarithmic standard deviation (denoted as σ).

- Epistemic Model—This model consists of two parts. The first part defines the epistemic uncertainty in the parameters of the median ground motion and aleatory model. Uncertainty in the model parameters, μ and σ, is defined by lognormal distributions for each. The second part of the epistemic model is the uncertainty that arises from the alternative ground motion models as derived from the input provided by each of the ground motion experts.

In aggregate, the probabilistic ground motion model is intended to provide a measure of the state of knowledge with respect to the assessment of ground motions at Yucca Mountain. To be valid, expert judgments in an expert elicitation must be traceable and technically defensible (NRC, 1996, 1997).

Spectral Decay (Kappa)

During review of the probabilistic seismic hazard analysis, specific issues were raised regarding the definition of the shallow crustal velocity near the free surface and the value of crustal kappa used for ground motion estimation at Yucca Mountain. These issues were raised because of the differences between the site condition at Yucca Mountain and the representations of the empirical strong motion database used (mainly California). There is a great difference in shear wave velocities, deep crustal damping [Q(f)], and shallow crustal {top 1–2 km [0.62–1.24 mi]} damping value (kappa) between California and Yucca Mountain. Kappa, defined as the spectral decay, is primarily caused by subsurface geological structures near the site. It is a smaller value for hard rock sites than for soft rock sites. The value of kappa estimated by Su, et al. (1996) for the southwestern part of the Nevada Test Site ranged from 0.005 to 0.024 seconds. In the probabilistic seismic hazard analysis, a value of 0.0186 second was used. DOE agreed[14] that if new studies find the median value of kappa for material with shear wave velocity below 1,900 m/s [6,234 ft/s] is different from 0.0186 second, median attenuation model will be adjusted. Potential adjustment of the median attentuation model will be addressed by DOE in Topical Report #3.

Vibratory Ground Motion Hazard Results

Median and fractile ground acceleration and aleatory and epistemic uncertainties for various earthquake magnitudes, sources-to-site distances, and different fault styles were estimated by the experts. Uncertainties in seismic source characterization and ground motion attenuation

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

relations were quantified by considering inputs from six seismic source fault displacement expert teams and seven ground motion experts. Each team and each expert provided their own assessment of uncertainty. The moment magnitude, $\mathbf{M}_w$, used in the probabilistic seismic hazard analysis ranged from 5.0 to 8.0 for normal and strike-slip faulting, and the distances examined were from 1 to 160 km [0.62 to 99 mi].

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

The vibratory ground motion hazard calculations were performed for each expert proposed attenuation equation and seismic source parameters. In general, the most ground motion contributors to uncertainty in the hazard were σ_μ and σ_σ, within expert uncertainties, rather than expert-to-expert uncertainties. The total uncertainty caused by ground motion is larger than the uncertainty caused by the seismic source characterization. Combining the experts' hazard curves, giving each expert equal weight, a set of integrated hazard curves were produced. The integrated results, based on input from the six expert teams and the seven ground motion expert represent the seismic hazard and its associated uncertainty at Yucca Mountain. The separation between the 15th- and 85th-percentile curves conveys the effects of the epistemic uncertainty on the calculated hazards. It should be noted these hazard curves were estimated at a reference rock outcrop on the surface, on a reference site at the same elevation as the repository.

Seismic Hazard Analysis

An evaluation of the seismic and ground motion characterization of CRWMS M&O (1998) and Stepp, et al. (2001) concluded that the seismic source characterization is adequate, and sufficient information exists for staff to review this aspect of the probabilistic seismic hazard analysis for a potential license application.

The ground motion characterization component of the Yucca Mountain seismic hazard analysis cannot be closed, however, until additional information is provided by DOE. Specifically, DOE agreed[15] to provide information to address staff concerns regarding (i) the ground motion expert elicitation process (see the discussion in Section 5.4, Expert Elicitation); (ii) site specific seismic

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

data, including input to the site response model (to be documented in the Seismic Design Inputs Report and Seismic Topical Report #3); (iii) Assumption 5.5 of CRWMS M&O (2000c), which assumes the median fault displacement values, rather than the mean values, are more accurate predictors of faulting for annual probabilities less than 10^{-6} per year (see earlier discussion of faulting); and (iv) incorporation of seismicity into cladding failure scenarios (see discussion in Section 3.3.4.4.3). Staff review of the Yucca Mountain ground motion models raises questions about the scientific basis for several of the expert ground motion assessments and the completeness elicitation feedback process. In particular, examination of several expert ground motion models illustrates that large differences exist between the experts, regarding predicted ground motions and epistemic and aleatory uncertainties. In some cases, staff noted wide diversity between experts and large variability within individual expert models. For instance, the 5-and 95-percent confidence limits pertaining to the estimate of the median ground motion for an earthquake of a given magnitude-distance and style of faulting for two cases of the expert models are shown in Table 3.3.2-1.

Table 3.3.2-1. Epistemic Uncertainty in the Median Peak Ground Acceleration Ground Motion—One Expert Model

				Fractiles Based on Epistemic Uncertainty—Median (g)			
Case	**Magnitude**	**Distance (km) [mi]**	**Style of Faulting**	**0.05**	**0.50***	**0.95**	**Ratio (95/5)**
1	6.5	1 [0.62]	Normal	0.11	0.50	2.33	21.2
2	6.5	10 [.62]	Normal	0.11	0.28	0.73	6.64

*Median scaled from attenuation model plots in CRWMS M&O, "Probabilistic Seismic Hazard Analyses for Fault Displacement and Vibratory Ground Motion at Yucca Mountain, Nevada, Final Report." WBS Number 1.2.3.2.8.3.6. Las Vegas, Nevada: CRWMS M&O. 1998.

The results provided in Table 3.3.2-1 suggest a large uncertainty in the estimate on the median ground motion. For instance, in Case 1, the expert suggests there is a 5-percent chance the true estimate of the median ground motion at a site 1 km [0.62 mi] from the **M**6.5 event is greater than 2.33g. In other words, the entire attenuation relationship shifts upward to this ground motion level. A similar conclusion can be derived for the lower estimate of the median ground motions. That is, there is a 5-percent chance the true estimate of the median ground motion at a site 1 km [0.62 mi] from the **M**6.5 event is less than 0.11g. For this expert, this observation is particularly interesting because his median estimate for the cases considered in the table is also the highest among the seven experts. In addition, the epistemic uncertainty provided by this expert is significantly larger than the variation in the range of median values predicted by the other experts.

As a measure of the technical integrity of the expert elicitation process and the scientific evaluation of individual expert assessments, and in light of these observations about the variability of their results, staff examined the bases for the ground motion models and results as documented in available reports (e.g., CRWMS M&O, 1998). The review raised a series of questions about the feedback-documentation part of the probabilistic seismic hazard analysis expert elicitation process

- Did the process involve training the experts, and were measures taken to demonstrate the experts understood, with reasonable assurance, the applicable probabilistic concepts and their implementations in the ground motion model?

- What was the process (i.e., technical evaluations) the experts undertook individually and within the context of workshops to affirm their understanding and concurrence with the probabilistic ground motion model derived from their input, and was the process adequate? For instance, did the facilitation teams provide the experts with an accurate awareness of the 5–95 fractile estimates of the median ground motion?

- In the example just given, where is the specific documentation of the scientific basis for the experts' agreement with the results? Although such information may exist, it is not available in CRWMS M&O (1998).

At the Structural Deformation and Seismicity Technical Exchange Meeting (October 2000), DOE provided a brief summary of the approach to expert elicitation used in the ground motion part of the probabilistic seismic hazard analysis. As part of the agreements made at that technical exchange, DOE agreed[16] to provide additional information about the ground motion elicitation process.

The information provided by many of the experts at the April 1997 workshop (mentioned previously) is a description of the procedure they followed to generate their inputs rather than providing the scientific basis for their assessments. In CRWMS M&O (1998), the individual ground motion expert reports contain statements the experts accepted the models derived by the facilitation team from their input. There was, however, no information provided as part of CRWMS M&O (1998) or later submissions that indicated the experts evaluated or reviewed the acceptability of the probabilistic ground motion models developed from their ground motion input parameters. In the absence of the necessary documentation, two questions remain unanswered:

- Were the experts aware the 5-and 95-percent confidence limits predicted by their models led to high estimates of median ground motion?

- Did the experts make an attempt to critically examine the distribution on their median ground motion (for a given ground motion measure) such that they were aware of the range and meaning of the epistemic uncertainty? The information presented in CRWMS M&O (1998) neither demonstrates the experts' understanding of the probabilistic ground motion model derived from their input nor describes the

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

methodology each expert used to assess the probabilistic estimates of ground motions made by their model.

In summary, to address the aforementioned concerns, DOE agreed[17] to provide the appropriate technical bases and document the process used to provide feedback to experts following the elicitation process.

3.3.2.4.3.2 Data Are Sufficient for Model Justification

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.2.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess mechanical disruption of engineered barriers (i.e., seismicity) with respect to sufficient data for model justification.

The seismic sources identified by the experts in the probabilistic seismic hazard analysis adequately characterize the potential sources of seismicity that will contribute to the anticipated peak and spectral ground motions at Yucca Mountain resulting from future earthquakes in the Yucca Mountain region based on the following observations:

- The seismic source characterization adequately incorporated the geologic and tectonic settings of the region into the probabilistic seismic hazard analysis. The range of tectonic models and the implications of those models to the probabilistic seismic hazard analysis are geologically consistent and entirely compatible with the current understanding of the Yucca Mountain tectonic framework and with the Basin and Range.

- Fault and areal sources were adequately identified by DOE. For example, comparison of Type I faults (McKague, et al., 1996) with the DOE lists of relevant faults (U.S. Geological Survey, 1996) shows general agreement, especially on the most important sources to the overall seismic hazard. DOE (U.S. Geological Survey, 1996) uses the terms relevant and potentially relevant in describing faults. At this time, staff consider all known candidate Type I faults in the Yucca Mountain region have been evaluated adequately. Staff found differences between DOE and NRC classifications of particular faults rooted in three parameters: fault trace length, attenuation function, and use of median or 84th percentile groundmotion: for identification of those faults that will exceed the 0.1g cutoff criterion. These differences lead to only minor differences in predicted ground motions (<0.1g) and are not considered significant to overall estimates of repository performance.

- The earthquake historical data and paleoseismicity were adequately characterized by DOE on the site and in the region. That record included approximately 30,000 earthquakes from historical earthquake catalogs used by the experts in the

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

probabilistic seismic hazard analysis. The earthquake magnitudes used in the analysis were corrected to a common moment magnitude ($\mathbf{M}_w$) and ranged from $\mathbf{M}_w$5.0 to $\mathbf{M}_w$8.0. Information on earthquakes from nuclear testing was removed based on compilations of all known nuclear tests. Foreshocks and aftershocks information was removed using standard declustering methods (Youngs, et al., 1987; Veneziano and van Dyck, 1985). The declustering techniques were tested for effectiveness by analysis of the Little Skull Mountain sequence, which had independently known foreshock, main shock, and aftershock sequences. Staff consider maximum magnitudes are reasonable for the fault sources based on established and published scaling relationships of rupture dimensions of the source. For example, empirical relationships between magnitude versus rupture length, rupture area, and maximum surface displacement (e.g., Wells and Coppersmith, 1994) were appropriately used to estimate maximum magnitude. Estimates of the rupture area and average slip on the fault were used by the experts to calculate the maximum magnitude event (Anderson, et al., 1996). For areal sources, the maximum magnitude earthquake was based on the maximum earthquake to occur within the area. The magnitude ranges used by the experts were based on moment magnitude (i.e., $\mathbf{M}_w$).

- Activity and fault slip rates were reasonably estimated by DOE. For example, recurrence and slip rates were primarily derived from paleoseismic data obtained by the U.S. Geological Survey detailed investigations of faulting in the Yucca Mountain region (CRWMS M&O, 1998). Additional constraints were derived from geologic data that estimated longer-term slip rates (e.g., Stamatakos, et al., 1997).

- Clustered events were adequately considered by DOE. For example, multiple rupture scenarios were derived (U.S. Geological Survey, 1996) and incorporated by the experts in the probabilistic seismic hazard analysis (CRWMS M&O, 1998).

In contrast, additional information pertaining to ground motion modeling is needed before staff can consider this acceptance criterion closed for seismicity (see the discussion in Section 3.3.2.4.3.1). To address this concern, DOE agreed[18] to provide the needed information.

3.3.2.4.3.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.2.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess mechanical disruption of engineered barriers (i.e., seismicity) with respect to the characterization and propagation of data uncertainty through the model abstraction.

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

DOE has not provided information to justify the probability distributions and bounding assumptions of ground motion or to account reasonably for the associated uncertainties and variabilities. Similar to faulting, DOE developed models for seismicity and ground motion based on a probabilistic seismic hazard analysis (CRWMS M&O, 1998; Stepp, et al., 2001). In those models, values for ground motion probabilities less than 10^{-6} annual exceedance per year are based on the median rather than the mean values from the probabilistic seismic hazard analysis curves (CRWMS M&O, 2000c, Assumption 5.5). As discussed in Section 3.3.2.4.3.1, the adequacy of the characterization and propagation of uncertainty associated with the use of the median rather than the mean values is not supported by sufficient technical basis (also see Section 3.2.2).

In addition, staff review of the probabilistic seismic hazard analysis noted insufficient technical bases with regard to the ground motion expert elicitation (see Section 3.3.2.4.3.1). To address this concern, DOE agreed[19] to provide this information.

3.3.2.4.3.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

At the time this report was prepared, effects of seismicity were excluded from the total system performance assessment model abstraction for disruptive events. DOE agreed to address the NRC concerns, as discussed in Sections 3.3.2.4.3.1, 3.3.2.4.3.2, and 3.3.2.4.3.3. Depending on the resolution of these concerns, the effects of seismicity will be included or excluded from the total system performance assessment model abstraction for disruptive events.

3.3.2.4.3.5 Model Abstraction Output Is Supported by Objective Comparisons

At the time this report was prepared, effects of seismicity were excluded from the total system performance assessment model abstraction for disruptive events. DOE agreed to address the NRC concerns, as discussed in Sections 3.3.2.4.3.1, 3.3.2.4.3.2, and 3.3.2.4.3.3. Depending on the resolution of these concerns, the effects of seismicity will be included or excluded from the total system performance assessment model abstraction for disruptive events.

3.3.2.4.4 Rockfall and Drift Collapse

3.3.2.4.4.1 System Description and Model Integration Are Adequate

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.2.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess mechanical disruption of engineered barriers (i.e., rockfall and drift collapse) with respect to system description and model integration.

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

Repository Safety After Permanent Closures

According to CRWMS M&O (2000c; 2001a,b), the consequences of rockfall and drift collapse are not being considered in the mechanical disruption of engineered barriers model abstraction for the DOE Total System Performance Assessment Code. The technical bases for this screening decision are provided in an analysis and model report (CRWMS M&O, 2000k) and CRWMS M&O (1999, 2000l) calculation reports. The detailed discussion that follows conveys the results of the staff review of these documents and the rationale for their findings. In summary, the staff review determined DOE

- Underestimated the probability, size, and areal coverage of potential discrete rock blocks that may be dislodged from the drift wall during earthquakes or from natural degradation of the drift wall rock mass

- Underestimated the probability, magnitude, and areal coverage of potential drift collapse

- Did not consider, in an acceptable manner, the potential consequences of rockfall and drift collapse on the engineered barrier subsystem

The effects of rockfall and drift collapse on repository performance will be manifested through changes in seepage characteristics and engineered barrier subsystem component temperatures, seismic response characteristics, near-field chemistry, corrosion rates, and functional capabilities (e.g., water infiltration pathways through breached drip shields).

<u>Occurrence of Rockfall and Drift Collapse</u>

The current DOE position on the occurrence of rockfall and drift collapse (CRWMS M&O, 2000m) is summarized as follows:

> Assuming complete degradation of the ground-support system at closure, time-dependent reduction in joint cohesion, thermal stresses, and seismic events combined will generate rockfall in less than 2.5 percent of the total length of emplacement drifts within 10,000 years after closure.

The DOE position contrasts with the following opinion of a DOE expert panel on drift stability:

> All drifts are likely to collapse in the fullness of time because of the severity of the THM [thermal-hydrological-mechanical]-driving gradients after emplacement ... (Brekke, et al., 1999, p. 3-16).

The DOE position was based on analyses documented in the CRWMS M&O (2000k) analysis and model report, which concluded that the emplacement drifts would experience only negligible rockfall and would essentially retain their as-built shape and size through the 10,000-year period of regulatory concern. The analyses were conducted using a computer code based on the key-block model in which a rock mass intersected by an opening is modeled as a network of rigid blocks and block-bounding fractures. In the model, a block may slide along its bounding fractures being influenced by gravitational force if sliding of the block is kinematically possible. The blocks exposed at the intersection with the opening and geometrically constrained in such a way that their sliding into the opening is kinematically possible are referred to as the key blocks. The sliding of all other blocks is kinematically impossible because of being restrained directly or indirectly by the key blocks. Therefore, the stability of the opening can be assured by preventing failure (i.e., sliding and eventual detachment from the network) of the key blocks. The driving force that may cause failure arises from gravity, and the resistance to failure is provided by the shear strength of the block-bounding fracture surfaces.

The key-block model does not have a mechanism to include a system of internal forces, such as may arise from a temperature distribution (thermal loading), earthquake (seismic loading), or other kinds of stress-generating processes. Furthermore, because blocks are treated as rigid in the mathematical formulation of the key-block model, the potential fracturing of blocks, which can have a significant effect on failure modes in a highly stressed rock mass, and the internal deformation of blocks, which has significant effects on fracture-surface stress, are not accounted for in key-block analysis. DOE indicated that some shortcomings of the key-block model were overcome in the drift degradation analysis (CRWMS M&O, 2000k) through the following procedures.

- The value of the cohesion parameter for fracture surfaces (i.e., the shear-strength intercept parameter of the Mohr-Coulomb strength criterion) was reduced from 0.86 Mpa [125 psi] to 0.01 MPa [1.45 psi] to represent thermal loading and time-dependent degradation of fracture surfaces.

- The value of friction angle for fracture surfaces was reduced by 8.0, 16.7, and 23.3 degrees, to represent seismic ground motions with 0.14, 0.30, and 0.43g peak ground accelerations. The value of friction-angle reduction in each case was calculated as the arc tangent of the respective peak ground accelerations. The peak ground acceleration values of 0.14, 0.30, and 0.43g are intended to represent the 1,000-, 5,000-, and 10,000-year earthquakes.

The rationale for the DOE approach is that the additional shear stress induced on fracture surfaces from a temperature distribution (thermal loading) or earthquake (seismic loading) and the weakening of fracture surfaces by time-dependent degradation can all be represented by the specified reduction of the cohesion and friction-angle parameters. DOE did not present a satisfactory mathematical basis to relate the cohesion reduction to the temperature distribution or the friction-angle reduction to the seismic loading to support an argument that the applied fracture-strength reductions appropriately represent the thermal and seismic loadings for the proposed repository.

Although it is theoretically possible to represent the effect of thermally induced shear stress on a fracture surface through a reduction of the fracture-surface strength, there are important requirements imposed by basic solid-mechanics principles that must be satisfied to apply the procedure satisfactorily. Because thermal stress is a tensor variable, the scalar parameter used to replace its effect must be mathematically tied to the components of the tensor, which, in turn, are dependent on the temperature, temperature gradient, mechanical boundary conditions, and mechanical properties. For this reason, the magnitude of the applied strength reduction would be expected to vary with the thermal load, time, location relative to the heated drift, fracture orientation, and rock-mass mechanical properties. As discussed in

Section 2.1.7.3 of this report, thermally induced rock failure at the proposed repository would likely be dominated by slip on subhorizontal fractures in the roof and floor areas of the drifts and in the pillars and slip on vertical fractures in the sidewall areas. The mechanisms of potential failure are controlled by the emplacement geometry, however, the actual occurrence of thermally induced rock failure would be determined by the strength and stiffness of the intact rock and fractures. None of these characteristics of thermally induced failure can be simulated correctly by representing thermal load as a constant cohesion reduction applied uniformly in a key-block model.

A similar argument can be made regarding the representation of seismic loading using a constant friction-angle reduction applied uniformly in the model. The appropriate friction-angle reduction would vary with the fracture orientation and with several characteristics of seismic ground motion that cannot be represented with peak ground acceleration only (e.g., frequency, duration, and direction of the associated particle motion).

The DOE expert panel on drift stability also noted the limitations of key-block modeling. Having identified rock raveling of small pieces of rock around the boundary of the drifts as a potentially important failure mechanism, the panel noted (referring to a set of illustrative numerical analyses conducted by the panel)

> These analyses do not support the application of key-block modeling to evaluate potential excavation degradation. The key-block approach does not examine subsequent behavior of a system of blocks or redistribution of loads. The raveling degradation may progress as a consequence of stress and/or temperature changes and other factors, which cannot be directly represented in a key-block model (Brekke, et al., 1999, p. 3–18).

Because of these shortcomings, the CRWMS M&O (2000k) analysis and model report does not provide the technical bases to support the current assessment of the effects of thermal loading, seismic loading, or time-dependent degradation of rock on the behavior of underground openings at Yucca Mountain. Further, the current assessment of drift stability is not consistent with the current state of knowledge on the behavior of underground openings in fractured rock [i.e., that the majority of the drifts are likely to collapse within a relatively short time (compared to the 10,000-year period of regulatory concern) after the cessation of maintenance]. This interpretation of the current state of knowledge is consistent with the DOE expert panel conclusion on drift stability (Brekke, et al., 1999, p. 3–16) and is supported by recent analyses of the behavior of unsupported drifts in fractured rock during seismic loading from an earthquake (Hsiung and Shi, 2001).

There are also concerns with the seismic and fracture data used for the drift degradation analysis. The seismic data used for the drift degradation analysis were the design basis seismic ground motions for both Categories 1 and 2 events. These seismic ground motion parameters are appropriate for preclosure-related design and analysis but are not proper for any postclosure considerations. DOE agreed[20] to address this concern in Seismic Topical Report #3. Development of the fracture data is documented in the fracture geometry analysis and model report (CRWMS M&O, 2000n), which, as previously noted,[21] contains the following implicit or explicit assumptions requiring technical justification:

- Volume sample from full periphery maps eliminates directional bias in the fracture distributions
- Fractures in the Exploratory Studies Facility and cross drift are representative of fracturing throughout the proposed emplacement volume at Yucca Mountain
- Lithology is the sole influence on fracture set characteristics
- Consideration of only fractures more than 1 m [3.3 ft] in length is representative or perhaps conservative with respect to rockfall and drift collapse
- Orientation variation within fracture sets is not important to drift stability
- Curvilinear trace length measured along the tunnel walls is representative of fracture size
- Strike and dip direction of shallowly dipping (<30 degrees) fractures is not important to drift stability
- The number of samples analyzed gives statistically significant results

To address the NRC concerns related to the occurrence of rockfall and drift collapse, as outlined in this section, DOE agreed[22] to

- Provide revised drift degradation analyses using an appropriate range of mechanical and strength properties for rock joints and account for their long-term degradation
- Provide an analysis of block sizes based on the full distribution of joint trace length data from the fracture geometry analysis and model report (CRWMS M&O, 2000n), including small joints trace lengths

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

[21]Ferrill D., W. Dunne, S. Hsiung, and A. Morris. Review of Analysis and Model Report entitled "Fracture Geometry Analysis for the Stratigraphic Units of the Repository Host Horizon." Letter Report to NRC (December 27). San Antonio, TX: CNWRA. 2000.

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

- Verify the results of the revised drift degradation analyses using (i) appropriate boundary conditions for thermal and seismic loading, (ii) critical fracture patterns from the fracture-network simulations used for the drift degradation analyses (at least two patterns for each rock unit), (iii) consistent thermal and mechanical properties for rock blocks and joints, (iv) long-term degradation of rock block and joint strength parameters, and (v) site-specific ground motion time histories appropriate for the postclosure period

- Provide the technical basis for the effective maximum rock size, including consideration of the effect of variation of the joint dip angle, to be used in assessing the response of the drip shield to rock block impacts

- Provide a detailed documentation of the analysis results

- Evaluate the uncertainties related to the rockfall and drift-collapse analyses and the importance of the outcome of the analyses to the performance of the repository

Staff reviewed DOE documentation of the fracture geometry parameters relevant to rockfall analyses of the repository host horizon rock units (CRWMS M&O, 2000n). Results of this review were documented in an NRC letter dated August 3, 2001,[23] and are summarized as follows.

- Directional Bias: Provide a technical basis for the conclusion that fracture geometry parameter values for the repository host horizon are correct; provide a set of data corrected for these sampling biases, along with a description of the methodology used for sampling bias correction; or risk inform the results to demonstrate that bias does not impact performance of the repository.

- Representativeness of Fracture Parameters: Provide a technical basis or rationale to support the extrapolation of fracture parameters to the repository footprint area. This extrapolation needs to account for heterogeneities in the repository host horizon and uncertainties in the fracture characteristics and their distribution. This technical basis is required to support the models and calculations used to select the new emplacement drift alignment and for the key-block analyses. Similarly, adequate technical rationales should be developed to support the use of the active fracture model and calculations that import or abstract fracture spacing data from the repository host horizon fracture analysis and model report (CRWMS M&O, 2000n).

- Misrepresentation of Aggregated Fracture Characteristics: Provide an adequate technical basis and rationale for the selection of fracture sets (i.e., sets based on orientation and lithology, rather than on origin) and provide statistics that represent the parameter distributions within each fracture set, or risk inform the aggregated characteristics.

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[23]Reamer, C.W. "Structural Deformation and Seismicity Key Technical Issue Agreements: Additional Information Needed." Letter (August 3, 2001) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

- Fractures More Than One Meter [3.3 ft] in Length: Provide an adequate technical basis for the fracture-length database used in rockfall analyses and other calculations, especially for the one-meter [3.3 ft] truncation. This technical basis should be adequate to support DOE key-block analyses for the Topopah Spring Tuff crystal-poor lower lithophysal unit. Alternatively, DOE could risk inform the fracture-length database.

- Orientation Variation Within Fracture Sets: Describe the procedure for defining fracture sets, explain the use of single-values to represent mean fracture set orientations, provide statistics that represent the range or variation in mean fracture orientations distribution of within each fracture set, or risk inform the fracture-orientation variation database.

- Fracture Trace Length and Fracture Shape: Provide an adequate technical basis for the method used to measure fracture lengths in tunnels and drifts and the potential fracture shapes and the significance, if any, to performance. Alternatively provide a risk-informed analysis of fracture trace length and fracture shape data and assumptions.

- Strikes of Shallowly Dipping Fractures: Provide a technically defensible distribution of fracture orientations and related population statistics for subhorizontal fractures used or assumed for tunnel stability analysis or risk inform the current uses or assumptions.

- Statistical Significance of Fracture Populations in the Exploratory Studies Facility and Enhanced Characterization of the Repository Block: Provide a population statistical analysis, unit by unit and set by set, of the fracture data and results and provide the character statistics, or risk inform the current assumptions.

Alternatively, DOE could explain the currently unsupported assumptions using a risk-informed approach. For example, with the absence of complete and persuasive evidence supporting the DOE assumptions of a uniform distribution of fracture characteristics throughout the repository, DOE could develop viable fracture models and use those models to develop a range of representative fracture characteristics most important to repository performance.

<u>Effect of Rockfall and Drift Collapse</u>

<u>Finite Element Modeling Methodology:</u> The process-level models used to approximate the response of the drip shield and waste package to various disruptive events are based on the finite element method. The finite element method is ideally suited to perform these analyses because it can readily account for the combined effects of nonlinear material behavior, nonlinear boundary conditions, and nonlinear geometry (i.e., large strains and large displacements). An important aspect of constructing finite element models, however, is the level of mesh discretization needed to achieve the requisite resolution of the results. To date, DOE has not provided any studies that demonstrate the finite element models used to simulate the functionality of the waste package and drip shield are sufficient to capture highly localized phenomena. For example, complex deformations of the waste package outer barrier in the immediate region of the waste package pallet support are expected. As a result, the finite element discretization will have to be sufficiently refined to capture adequately the localized

stress states. Reasonable approximations of the stress are needed to assess the susceptibility of the various engineered barrier subsystem components to stress corrosion cracking.

Drip Shield: The finite element analysis models used by DOE to assess the structural integrity of the drip shield when subjected to rock block impacts (CRWMS M&O, 2000l) do not employ (i) appropriate boundary conditions, (ii) material properties corresponding to the expected emplacement drift environment and the effects of various material degradation processes, or (iii) acceptable criteria for assessing material failure and susceptibility to stress corrosion cracking.

Even though the drip shield is intended to be a free-standing structure, the DOE finite element model uses fixed displacement boundary conditions at its base. In addition, the finite element model did not account for (i) the potential interaction between the drip shield and gantry rails, (ii) the effect of the invert floor moving vertically upward as a result of the seismic excitation that may occur concurrently with rockfall, or (iii) the degradation of the carbon steel structural framework of the invert. These boundary conditions have a significant influence on the overall structural behavior of the drip shield when subjected to rock block impacts. As a result, the location and magnitude of the maximum stresses experienced by the drip shield when subjected to rockfall have not been adequately determined. DOE also assumed in these models that the contact area between the impacting rock block and drip shield will encompass at least 3 m [9.9-ft] length of the drip shield. Distributing the impact load over a relatively large surface area of the drip shield significantly reduces the magnitude of stress that would be experienced by the drip shield if the initial contact area was consistent with localized, point-type impacts.

DOE indicated the drip shield will be fabricated using Titanium Grades 7 and 24. The constitutive relationships used for these two materials within the finite element models simulating the drip shield and rock block impacts were derived from empirical data obtained at room temperature {i.e., approximately 20 °C [68 °F]}. The mechanical material properties for Titanium Grade 7 (American Society of Mechanical Engineers, 1995, 2001), however, are strongly dependent on temperature. The temperature-dependent values for the yield stress, ultimate tensile strength, and Young's modulus of Titanium Grades 5 or 24 are not provided in the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. Note that the compositions of Titanium Grades 5 and 24 are the same except Grade 24 contains 0.04–0.08-percent palladium. As a result, it is expected these two grades will exhibit similar mechanical behavior (i.e., mechanical properties). The U.S. Department of Defense (1998) and ASM International (1994) provide extensive material data for Titanium Grade 5. The Titanium Grade 5 values for the yield stress, ultimate tensile strength, and Young's modulus extracted from graphical data provided in U.S. Department of Defense (1998) are also strongly dependent on temperature. Even though Titanium Grade 5 exhibits much higher strengths than Titanium Grade 7, the relative effects of temperature are still significant and must be considered when assessing the ability of the drip shield to withstand rock block impacts.

In addition to temperature effects, DOE has not adequately addressed the influence of (i) welding flaws and defects, (ii) hydrogen entry into metal, and (iii) fluoride on the corrosion rate of titanium when assessing the ability of the drip shield to perform its intended functions after rockfall and seismic events. Enhanced susceptibility of the titanium drip shield to cracking may occur through hydrogen generated from the galvanic coupling of titanium with degraded carbon steel ground support materials such as rock bolts, steel mesh, or steel sets (CRWMS M&O, 2000o), or the gantry rail. The subsequent uptake of hydrogen into the titanium drip shield materials may reduce the ductility of the titanium drip shield. In addition, corrosion rates of titanium alloys are strongly dependent on fluoride concentration. Groundwater compositions in the emplacement drifts may have elevated fluoride concentrations as a result of evaporation (CRWMS M&O, 2000p). Elevated fluoride concentrations can result in accelerated corrosion of the titanium drip shield and increased hydrogen uptake that, in turn, may increase the susceptibility of the titanium drip shield to either mechanical failure or hydrogen-induced cracking.

No discussion was provided in the CRWMS M&O (2000l) report detailing which components or types of strain measure were used to conclude that "... no crack develops in the drip shield due to the dynamic impact of a rock on the drip shield for any of the rock sizes ... ." For generalized three-dimensional stress states, failure criteria for metals are typically based on maximum shear stress, octahedral shear stress, Tresca stress, Von Mises stress, or strain-energy density. These measures are used because they can be readily employed to discern failure when complex stress states exist using data derived from simple tension tests.

The finite element analysis results obtained from the drip shield and rock block impact simulations were also used to assess the potential for the initiation of stress corrosion cracking in the drip shield. The results indicated that the drip shield residual stresses developed as a consequence of the rock block impact may be sufficient to cause stress corrosion cracking. No discussion was provided in the report detailing which components or types of stress were used in making this assessment. For example, no information was provided that addresses the recommended procedure for how generalized three-dimensional stress states obtained from engineering analyses should be interpreted to determine whether the initiation stress threshold for stress corrosion cracking has been exceeded. In addition, given the significant reduction in yield stress for Titanium Grades 7 and 24 at emplacement drift temperatures relative to the corresponding values at room temperature, the assumed initial stress threshold for the stress corrosion cracking criterion does not appear to be conservative.

The potential effects of dead loads on the drip shield caused by rockfall and drift collapse have not been adequately considered by DOE when assessing the performance capabilities of the drip shield. These effects include, but may not be limited to, changes to the dynamic response of the drip shield when subjected to seismic excitation, buckling, and creep.

It can be reasonably assumed that the effective mass of the drip shield will increase without appreciably changing its structural stiffness when supporting dead loads. The natural frequencies of the drip shield, therefore, will be reduced. Reduction in the drip shield natural frequencies is a concern because earthquake loads typically resonate structures with natural frequencies below 33 Hz. As a consequence, the drip shield may respond to seismic excitation by oscillating with displacements large enough to cause repeated impacts with a waste package, resulting in damage presently not accounted for.

Under static conditions, dead loads may also cause the drip shield to buckle or experience large plastic deformations, potentially transferring the dead loads from the drip shield directly to a waste package.

Because the reductions in yield stress and ultimate tensile strength for Titanium Grades 7 and 24 resulting from elevated emplacement drift temperatures are significant, there is some concern by the staff that these materials will also be susceptible to creep-related failures arising from the support of dead loads (e.g., fallen rock blocks or drift collapse). This concern is further substantiated by information provided in a U.S. Department of Defense handbook which states

> Below about 149 °C [300 °F], as well as above about 371 °C [700 °F], creep deformation of titanium alloys can be expected at stresses below the yield strength. Available data indicate that room-temperature creep of unalloyed titanium may be significant (exceed 0.2-percent creep-strain in 1,000 hours) at stresses that exceed approximately 50 percent F_{ty} [tensile yield stress], ... (U.S. Department of Defense, 1998, p. 5-2).

Moreover,

> The alpha-beta alloys [Titanium Grade 24] have good strength at room temperature and for short times at elevated temperature. They are not noted for long-time creep strength. (U.S. Department of Defense, 1998, p. 5-51).

Room-temperature creep has been investigated for a variety of alpha or near-alpha (hexagonal closed packed) and alpha-beta (hexagonal closed packed-body centered cubic) titanium alloys. Significant room-temperature creep can occur in alpha or near-alpha titanium alloys, whereas, alpha-beta titanium alloys are not as susceptible to this degradation mechanism. Chu (1970) reported considerable creep strains for a near-alpha T1-6Al-2Cb-1Ta-0.8 Mo alloy at room temperature when the applied stress was above 80 percent of the yield strength. In contrast, the creep strains observed for alpha-beta Ti-6Al-4V at 90 percent of the yield strength are low (Odegard and Thompson, 1974) but dependent on the microstructure of the alloy (Imam and Gilmore, 1979). Tests conducted on as-welded Ti-6Al-4V showed similar behavior to the base alloy with the exception of a decrease in the yield strength for the as-welded material (Odegard and Thompson, 1974).

DOE has neither referenced specific creep data for Titanium Grades 7 and 24 nor provided adequate analyses demonstrating that dead loads caused by fallen rock blocks and drift collapse will not occur. Creeping of the drip shields subjected to dead loads can reduce the clearance between the drip shield bulkhead and the waste package. Given time, the dead loads may ultimately be supported by the waste package directly, or during a seismic event, the clearance may have been sufficiently reduced to the point that the drip shield will repeatedly impact the waste package, resulting in damage presently not accounted for.

DOE proposed an evaluation of the drip shield static loading (CRWMS M&O, 2000q) using a procedure based on Rankine's theory of earth pressure (e.g., Terzaghi, et al., 1996). The proposed approach, however, is inappropriate because it does not account for the dead weight of fallen rock that may rest directly on the drip shield, and it does not adequately represent the lateral loads arising from naturally occurring or engineered backfill.

To address NRC concerns related to the effect of rockfall and drift collapse on the drip shield, as outlined in this section, DOE agreed[24,25,26] to

- Perform drip shield seismic evaluations that include the effects of static loads from fallen rock
- Perform drip shield rockfall evaluations that include the effects of (i) wall thinning caused by corrosion, (ii) hydrogen embrittlement, and (iii) multiple rock blocks falling simultaneously
- Provide (i) the justification for not including the rockfall effect and drift collapse loads on stress corrosion cracking of the drip shield and (ii) the documentation for the point loading rockfall analyses
- Demonstrate how the Tresca Failure criterion bounds a fracture mechanics approach to calculating the mechanical failure of the drip shield. Provide a technical basis for a stress measure that can be used as the equivalent uniaxial stress for assessing the susceptibility of titanium to stress corrosion cracking. The proposed equivalent uniaxial stress measure must be consistent and compatible with the methods proposed by DOE to assess stress corrosion cracking of the containers in WAPDEG. A detailed discussion of how the equivalent uniaxial stress measure will be used to determine nucleation of stress corrosion cracks in the calculations performed to evaluate the stress corrosion cracking criterion for the drip shield should be included
- Clarify why the effects of seismicity and large block rockfall are not considered in the Total System Performance Assessment Code (features, events, and processes numbers 1.2.03.02.00 and 2.1.07.01.00) [when providing this clarification, DOE should include analyses of the drip shield subjected to rock block impacts and seismic loads using boundary conditions that (i) represent the drip shield as a free-standing structure, (ii) account for the potential interactions between the drip shield and gantry rails (and any other relevant structures, systems, or components), and (iii) include the effects of seismic ground motion at the invert floor and take into account welding flaws and defects and the reduced mechanical strength of titanium commensurate with anticipated temperatures]

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

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

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

- Provide technical basis for the screening argument pertaining to creeping of metallic materials in the engineered barrier subsystem (features, events, and processes number 2.1.07.05.00)

Waste Package: The finite element analysis models used by DOE to assess the structural integrity of the waste package when subjected to rock block impacts (CRWMS M&O, 1999) do not employ (i) boundary conditions between the inner and outer barriers of the waste package consistent with the current waste package design, (ii) material properties corresponding to the expected emplacement drift environment and the effects of various material degradation processes, or (iii) acceptable criteria for assessing material failure and susceptibility to stress corrosion cracking.

Furthermore, DOE has not performed an assessment of the stresses generated in the waste package outer barrier near the pallet support caused by rock block impacts and seismic excitation. Specific aspects of the new waste package design and analyses of concern to the NRC staff are (i) the assumption that the inner and outer barriers can be treated as a single composite component in the DOE finite element models, (ii) the potential loss of material ductility in the immediate area of the closure lid welds, (iii) the design provisions that do not properly account for the difference in thermal expansion between the inner and outer barriers of the waste package, and (iv) the failure criteria used to assess the structural integrity of the waste package.

DOE has not adequately addressed the effects of welding flaws and defects and waste package degradation processes such as uniform corrosion, localized corrosion, stress corrosion cracking, and the possible decreased ductility as a result of container fabrication or long-term thermal aging that may reduce the ability of the waste package to withstand rockfall or seismic events. Penetration of the waste package outer barrier by localized corrosion or stress corrosion cracking will result in the exposure and subsequent degradation of the inner stainless steel container. In addition, the effects of container fabrication, thermal aging, or an increase in the exposure temperature as a result of volcanic activity may result in the formation of brittle phases that reduce the ductility of the waste package materials.

To address the NRC concerns related to the effects of rockfall and drift collapse on the waste package, as outlined in this section, DOE agreed[27,28,29,30] to

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

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

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

[30]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

- Perform waste package rockfall evaluations that include the effects of (i) potential waste package closure weld material embrittlement after stress annealing and (ii) multiple rock blocks falling simultaneously

- Provide the documentation for the waste package point loading rockfall analyses

- Demonstrate how the Tresca Failure criterion bounds a fracture mechanics approach to calculating the mechanical failure of the waste package. Provide a technical basis for a stress measure that can be used as the equivalent uniaxial stress for assessing the susceptibility of Alloy 22 to stress corrosion cracking. The proposed stress measure must be consistent and compatible with the methods proposed by DOE to assess stress corrosion cracking of the containers in WAPDEG. A detailed discussion of how the stress measure will be used to determine nucleation of stress corrosion cracks in the calculations performed to evaluate the stress corrosion cracking criterion for the waste package should be included).

- Clarify why the effects of seismicity and large block rockfall are not considered in the Total System Performance Assessment Code (features, events, and processes numbers 1.2.03.02.00 and 2.1.07.01.00) [when providing this clarification, DOE should include analyses of the waste package that consider the effects of (i) temperature-dependent material properties, (ii) uniform and localized corrosion, (iii) welding flaws and defects, (iv) differential thermal expansion effects, and (v) susceptibility of the outer barrier to stress corrosion cracking where potential interactions with the drip shield may have occurred and in the immediate contact region with the pallet support]

- Clarify the description of the primary features, events, and processes (number 1.2.03.02.00, seismic vibration causes container failure)

- Provide the technical basis for the screening argument pertaining to the differing thermal expansion of repository components (features, events, and processes number 2.1.11.05.00)

3.3.2.4.4.2 Data Are Sufficient for Model Justification

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.2.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess mechanical disruption of engineered barriers (i.e., rockfall and drift collapse) with respect to sufficient data for model justification.

The fracture contact stiffness and strength properties used to support the drift degradation analysis (CRWMS M&O, 2000k) are not sufficient. These properties were determined based on 12 laboratory shear tests of fractures from the Topopah Spring densely welded devitrified lithophysal-poor Tuff. No distinction was made on the fracture properties among the three subunits of the Topopah Spring densely welded devitrified lithophysal-poor Tuff thermal-mechanical unit (CRWMS M&O, 2000k,r). Furthermore, the fracture shear stiffness (an important parameter for the verification studies) is not available and was assumed in the analysis (CRWMS M&O, 2000r). DOE agreed[31] to address these concerns.

See Sections 2.1.7.3, 3.3.1.4.1.2, and 3.3.1.4.2.2 of this report for comments related to data being sufficiently characterized and propagated for model justification for this topic area.

3.3.2.4.4.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

At the time this report was prepared, the effects of rockfall and drift collapse were excluded from the total system performance assessment model abstraction for disruptive events. DOE agreed to address the NRC concerns, as discussed in Sections 3.3.2.4.4.1, 3.3.2.4.4.2, and 3.3.2.4.4.3. Depending on the resolution of these concerns, the effects of rockfall and drift collapse will be included or excluded from the total system performance assessment model abstraction for disruptive events.

3.3.2.4.4.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

At the time this report was prepared, the effects of rockfall and drift collapse were excluded from the total system performance assessment model abstraction for disruptive events. DOE agreed to address the NRC concerns, as discussed in Sections 3.3.2.4.4.1, 3.3.2.4.4.2, and 3.3.2.4.4.3. Depending on the resolution of these concerns, the effects of rockfall and drift collapse will be included or excluded from the total system performance assessment model abstraction for disruptive events.

3.3.2.4.4.5 Verification of Model Abstraction

At the time this report was prepared, the effects of rockfall and drift collapse were excluded from the total system performance assessment model abstraction for disruptive events. DOE agreed to address the NRC concerns, as discussed in Sections 3.3.2.4.4.1, 3.3.2.4.4.2, and 3.3.2.4.2.3. Depending on the resolution of these concerns, the effects of rockfall and drift collapse will be included or excluded from the total system performance assessment model abstraction for disruptive events.

3.3.2.4.5 Criticality

3.3.2.4.5.1 System Description and Model Integration Are Adequate

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.2.5), is sufficient to conclude that the necessary information will be available at the

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

time of a potential license application to assess mechanical disruption of engineered barriers (i.e., criticality) with respect to system description and model integration.

DOE screened the occurrence of nuclear criticality for commercial spent nuclear fuel from consideration in the Total System Performance Assessment Code based on no waste package breach or failure and a low probability of critical configuration formation at any time during the postclosure period (CRWMS M&O, 2000s,t). DOE recently indicated (Bechtel SAIC Company, LLC, 2001a,b), however, there would be waste package failures prior to 10,000 years caused by improper heat treatment during fabrication. In addition, DOE has yet to demonstrate adequately the waste packages can satisfactorily maintain confinement from either direct or indirect effects that can be attributed to mechanically disruptive events or various corrosion processes (see Section 3.3.1). As a result, DOE agreed[32] to reexamine the screening arguments for postclosure criticality.

For criticality induced by seismic loading, the methodology for estimating the probability of a criticality event (DOE, 2000) will first identify and evaluate the waste package configurations that could become critical or supercritical as a result of being subjected to seismic loads. These configurations are called seismic predecessor configurations. To determine the probability of a criticality event initiated by seismic loads, the probability of any given seismic predecessor configuration will be multiplied by the probability of a seismic event that has a magnitude capable of taking such a configuration to criticality.

The methodology for estimating the probability of an igneous-induced criticality begins by identifying the potential critical configurations that can be created by an igneous event. The criticality potentials of these configurations are then evaluated according to the process described in the topical report.

DOE used the methodology for estimating the probability of criticality induced by an igneous event for waste packages containing pressurized water reactor spent nuclear fuel (CRWMS M&O, 2000t). To obtain this probability estimate, DOE evaluated the criticality configuration potential pertaining to the complete destruction of the seven waste packages located in Zone 1 (CRWMS M&O, 2000e). The result indicated that the system would be subcritical for the range of pellet spacings and fuel and magma volumes considered in the analyses. The analysis did not include any other waste package types containing high-enriched fuel (e.g., U.S. Navy and DOE-owned spent nuclear fuel). As for the Zone 2-type damages (CRWMS M&O, 2000e) (i.e., partial damage of the remaining waste packages in any drift intersected by an igneous intrusion), DOE calculated the probability for criticality to be 1.8×10^{-7} over 10,000 years, which is smaller than 1×10^{-4} over 10,000 years (screening criteria per 10 CFR 63.113). Similar to the Zone 1 analysis, DOE only evaluated waste packages containing pressurized water reactor spent nuclear fuel. Staff do not believe DOE can screen out igneous-induced criticality by evaluating only one waste package and fuel type. Therefore, the approach should include the probability and configurations for all potential waste

[32]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Range of Operating Temperatures (September 18–19, 2001)." Letter (October 2) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

package and fuel types. In the Range of Thermal Operating Modes Technical Exchange, DOE agreed[33] to update the probability estimates for criticality by performing analyses that include different waste package and fuel types.

Because of the large uncertainty associated with calculating criticality probabilities, DOE also agreed[34] to perform a what-if criticality consequence analysis using a revised methodology (DOE, 2000), which is presently being reviewed by NRC, to determine the potential effects of criticality on meeting repository performance requirements.

3.3.2.4.5.2 Data Are Sufficient for Model Justification

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.2.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess mechanical disruption of engineered barriers (i.e., criticality) with respect to sufficient data for model justification.

DOE indicated relevant data pertaining to seismicity, faulting, volcanism, and rockfall used in criticality models will be consistent with data used in other areas of the total system performance assessment, where appropriate (DOE, 2000). Other significant data will be contained in the validation reports for the inventory, neutronics, and geochemistry computer codes that will be used in the criticality modeling. DOE agreed[35] to provide these validation reports to NRC prior to submission of any license application for the Yucca Mountain repository.

3.3.2.4.5.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.2.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess mechanical disruption of engineered barriers (i.e. criticality) with respect to the characterization and propagation of data uncertainty through the model abstraction.

DOE indicated that uncertainty distributions of parameters associated with seismicity, faulting, volcanism, and rockfall used in criticality models will be consistent with other areas of the total system performance assessment where appropriate (DOE, 2000). The validation reports for the inventory, neutronics, and geochemistry computer codes will quantify the effect of data

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[33]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Range of Operating Temperatures (September 18–19, 2001)." Letter (October 2) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[34]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Criticality (October 23–24, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

[35]Ibid.

uncertainty on the results of these computer codes. DOE agreed[36] to provide these validation reports to NRC prior to submission of any license application for the Yucca Mountain repository.

3.3.2.4.5.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

At the time this report was prepared, the effects of criticality were excluded from the total system performance assessment model abstraction for disruptive events. DOE agreed to address the NRC concerns, as discussed in Sections 3.3.2.4.5.1, 3.3.2.4.5.2, and 3.3.2.4.5.3. Depending on the resolution of these concerns, the effects of criticality will be included or excluded from the total system performance assessment model abstraction for disruptive events.

3.3.2.4.5.5 Model Abstraction Output Is Supported by Objective Comparisons

At the time this report was prepared, the effects of criticality were excluded from the total system performance assessment model abstraction for disruptive events. DOE agreed to address the NRC concerns, as discussed in Sections 3.3.2.4.5.1, 3.3.2.4.5.2, and 3.3.2.4.5.3. Depending on the resolution of these concerns, the effects of criticality will be included or excluded from the total system performance assessment model abstraction for disruptive events.

3.3.2.5 Status and Path Forward

Table 3.3.2-2 provides the status of all key technical issue subissues, referenced in Section 3.3.2.2, for the Mechanical Disruption of Engineered Barriers Integrated Subissue. The table also provides the related DOE and NRC agreements pertaining to the Mechanical Disruption of Engineered Barriers Integrated Subissue. The agreements listed in the table are associated with one or all five generic acceptance criteria discussed in Section 3.3.2.4. Note that the status and the detailed agreements (or path forward) pertaining to all the key technical issue subissues are provided in Table 1.1-3 and Appendix A.

[36]Schlueter, J.R. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Criticality (October 23–24, 2000)." Letter (October 27) to S. Brocoum, DOE. Washington, DC: NRC. 2000.

Table 3.3.2-2. Related Key Technical Issue Subissues and Agreements			
Key Technical Issue	**Subissue**	**Status**	**Related Agreements***
Container Life and Source Term	Subissue 1—Effects of Corrosion Processes on the Lifetime of the Containers	Closed-Pending	CLST.1.13 CLST.1.14 CLST.1.16 CLST.1.17
	Subissue 2—Effects of Phase Instability of Materials and Initial Defects on the Mechanical Failure and Lifetime of the Containers	Closed-Pending	CLST.2.01 through CLST.2.09
	Subissue 5—Effect of In-Package Criticality on Waste Package and Engineered Barrier Subsystem Performance	Closed-Pending	CLST.5.01 CLST.5.03 CLST.5.06 CLST.5.07
	Subissue 6—Effect of Alternate of Engineered Barrier Subsystem Design Features on Container Lifetime and Radionuclide Release from the Engineered Barrier Subsystem	Closed-Pending	None
Igneous Activity	Subissue 1—Probability of Igneous Activity	Closed-Pending	None
	Subissue 2—Consequences of Igneous Activity	Closed-Pending	IA.2.10 IA.2.18 IA.2.19 IA.2.20
Repository Design and Thermal-Mechanical Effects	Subissue 1—Design Control Process	Closed	None
	Subissue 2—Seismic Design Methodology	Closed-Pending	RDTME.2.01 RDTME.2.02
	Subissue 3—Thermal-Mechanical Effects	Closed-Pending	RDTME.3.03 RDTME.3.15 To RDTME.3.19
Structural Deformation and Seismicity	Subissue 1—Faulting	Closed-Pending	SDS.1.02
	Subissue 2—Seismicity	Closed-Pending	SDS.2.01 SDS.2.03 SDS.2.04
	Subissue 3—Fracturing and Structural Framework of the Geologic Setting	Closed-Pending	SDS.3.04

Table 3.3.2-2. Related Key Technical Issue Subissues and Agreements (continued)			
Key Technical Issue	**Subissue**	**Status**	**Related Agreements***
Structural Deformation and Seismicity	Subissue 4—Tectonic Framework of the Geologic Setting	Closed	None
Total System Performance Assessment and Integration	Subissue 1—System Description and Demonstration of Multiple Barriers	Closed-Pending	None
	Subissue 2—Scenario Analysis and Event Probability	Closed-Pending	TSPAI.2.02 TSPAI.2.04
	Subissue 3—Model Abstraction	Closed-Pending	TSPAI.3.06
	Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards	Closed-Pending	None
*Related DOE and NRC agreements are associated with one or all five generic acceptance criteria. Note: Key Technical Issue Agreement GEN.1.01 pertains to multiple integrated subissues, as well as some specific issues related to this integrated subissue.			

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (through specific data collection, testing, and analyses), acceptably addresses the NRC questions so that no information beyond that provided, or agreed to, will likely be required at the time of a potential license application.

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3.3.3 Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms

3.3.3.1 Description of Issue

The Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms Integrated Subissue addresses features, events, and processes in the engineered barrier subsystem that may alter the chemical composition or volume of water present on the drip shield and waste package surfaces. To facilitate issue resolution, hydrologic processes affecting seepage rates are treated in the Flow Paths in the Unsaturated Zone Integrated Subissue, and quantity and chemistry of water inside breached waste packages are addressed by the Radionuclide Release Rates and Solubility Limits Integrated Subissue. Relationship of this integrated subissue to other subissues is depicted in Figure 3.3.3-1. The figure shows the relationship between this integrated subissue and the flow paths in the unsaturated zone (Section 3.3.6), mechanical disruption of engineered barriers (Section 3.3.2), radionuclide transport in the unsaturated zone (Section 3.3.7), degradation of engineered barriers (Section 3.3.1), and radionuclide release and solubility limits (Section 3.3.4) subissues. The overall organization and identification of all the integrated subissues are depicted in Figure 1.2-2.

3.3.3.2 Relationship to Key Technical Issue Subissues

The Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms Integrated Subissue incorporates subject matter previously captured in the following key technical issue subissues:

- Evolution of the Near-Field Environment: Subissue 1—Effects of Coupled Thermal-Hydrological-Chemical Processes on Seepage and Flow (NRC, 2000a)
- Evolution of the Near-Field Environment: Subissue 2—Effects of Coupled Thermal-Hydrological-Chemical Processes on the Waste Package Chemical Environment (NRC, 2000a)
- Evolution of the Near-Field Environment: Subissue 3—Effects of Coupled Thermal-Hydrological-Chemical Processes on the Chemical Environment for Radionuclide Release (NRC, 2000a)
- Evolution of the Near-Field Environment: Subissue 4—Effects of Thermal-Hydrological-Chemical Processes on Radionuclide Transport through Engineered and Natural Barriers (NRC, 2000a)
- Evolution of the Near-Field Environment: Subissue 5—Effects of Coupled Thermal-Hydrological-Chemical Processes on Potential Nuclear Criticality in the Near-Field (NRC, 2000a)
- Radionuclide Transport: Subissue 4—Nuclear Criticality in the Far Field (NRC, 2000b)

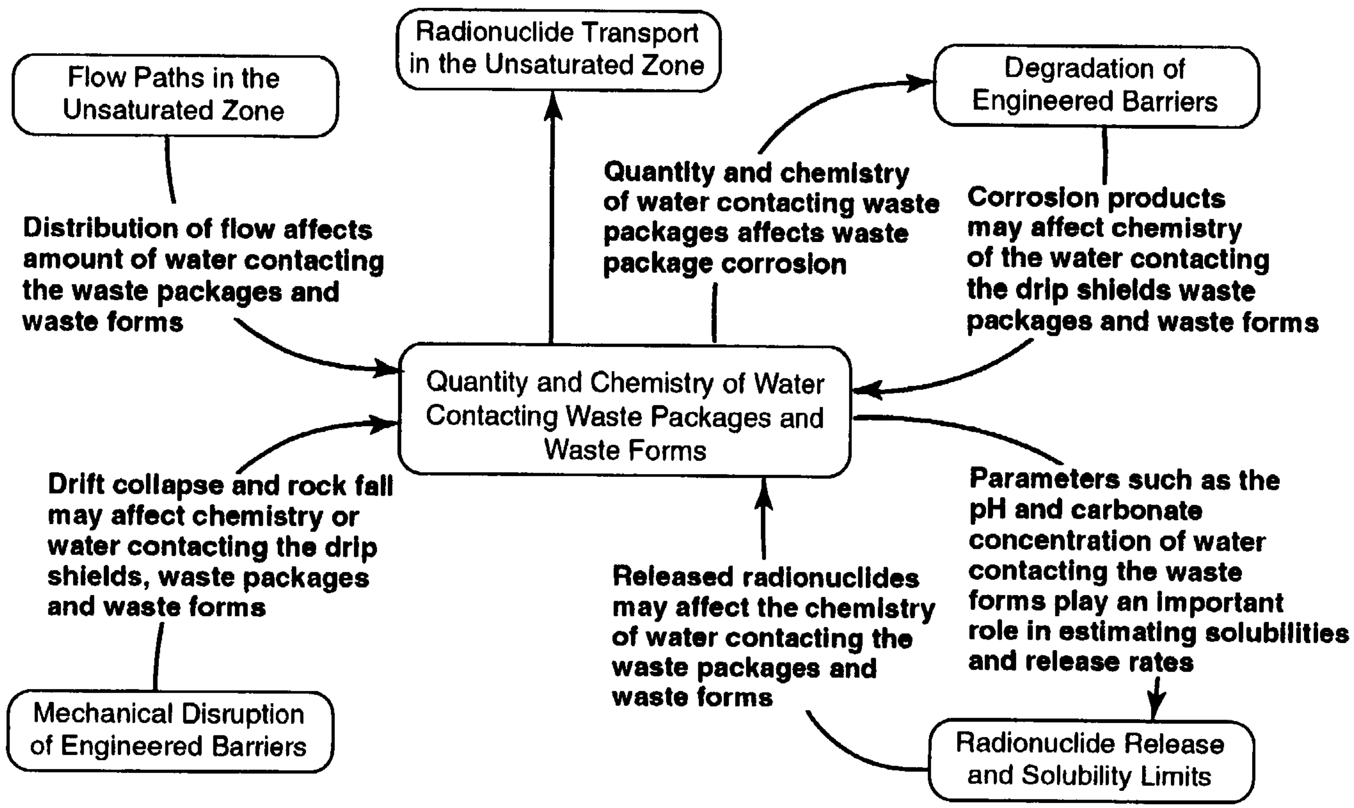

Figure 3.3.3-1. Diagram Illustrating the Relationship Between the Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms Integrated Subissue and Other Integrated Subissues

- Thermal Effects on Flow: Subissue 1—Features, Events, and Processes Related to Thermal Effects on Flow (NRC, 2000c)
- Thermal Effects on Flow: Subissue 2—Thermal Effects on Temperature, Humidity, Saturation, and Flux (NRC, 2000c)
- Container Life and Source Term: Subissue 1—The Effects of Corrosion Processes on the Lifetime of the Containers (NRC, 2000d)
- Container Life and Source Term: Subissue 3—The Rate at Which Radionuclides in Spent Nuclear Fuel are Released from the Engineered Barrier Subsystem through the Oxidation and Dissolution of Spent Nuclear Fuel (NRC, 2000d)
- Container Life and Source Term: Subissue 4—The Rate at Which Radionuclides in High-Level Waste Glass are Released from the Engineered Barrier Subsystem (NRC, 2000d)
- Container Life and Source Term: Subissue 5—The Effect of In-Package Criticality on Waste Package and Engineered Barrier Subsystem Performance (NRC, 2000d)
- Container Life and Source Term: Subissue 6—The Effects of Alternate Engineered Barrier Subsystem Design Features on Container Lifetime and Radionuclide Release from the Engineered Barrier Subsystem (NRC, 2000d)
- Repository Design and Thermal-Mechanical Effects: Subissue 3—Thermal-Mechanical Effects on Underground Facility Design and Performance (NRC, 2000e)
- Unsaturated and Saturated Flow Under Isothermal Conditions: Subissue 4—Deep Percolation (NRC, 2000f)
- Structural Deformation and Seismicity: Subissue 3—Fracturing and Structural Framework of the Geological Setting (NRC, 2000g)
- Total System Performance Assessment Integration: Subissue 1—System Description and Demonstration of Multiple Barriers (NRC, 2000g)
- Total System Performance Assessment Integration: Subissue 2—Scenario Analysis and Event Probability (NRC, 2000g)
- Total System Performance Assessment Integration: Subissue 3—Model Abstraction (NRC, 2000g)
- Total System Performance Assessment Integration: Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards (NRC, 2000g)

The key technical issue subissues formed the bases for the previous versions of the issue resolution status reports and also were the bases for technical exchanges with DOE, where agreements were reached on what additional information DOE needed to provide to resolve the subissue. The resolution status of this integrated subissue is based on the resolution status of each of the contributing key technical issue subissues. The subsequent sections incorporate applicable portions of these key technical issues subissue, however, no effort was made to explicitly identify each subissue.

3.3.3.3 Importance to Postclosure Performance

One aspect of risk informing the NRC review was to determine how this integrated subissue is related to the DOE repository safety strategy. DOE recognizes the importance of infiltration to repository performance at Yucca Mountain in the repository safety strategy for the postclosure safety case (CRWMS M&O, 2000a). Five of the DOE eight principal factors in the repository safety strategy can be related to the Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms Integrated Subissue. These principal factors are (i) seepage into emplacement drift, because this describes the quantity of water initially available to drip onto the drip shields and waste packages; (ii) performance of the drip shield/drift invert system, because performance depends on the quantity and chemistry of water contacting these materials; (iii) performance of the waste package, because performance depends on the quantity and chemistry of water contacting the waste package; (iv) radionuclide concentration limits in water, because radionuclide concentration limits in pure water may differ from the limits in the more complex water compositions expected to occur in an emplacement drift setting; and (v) radionuclide delay through the unsaturated zone, because the quantity and chemistry of water shed off the drip shield onto the inverts could influence the mobility of radionuclides by controlling precipitation and sorption processes.

3.3.3.4 Technical Basis

NRC developed a plan (2002) consistent with the acceptance criteria and review methods found in previous issue resolution status reports. A review of DOE approaches for including quantity and chemistry of water contacting the waste packages and waste forms in total system performance assessment abstractions is provided in the following subsections. The review is organized according to the five acceptance criteria identified in Section 1.5: (i) System Description and Model Integration Are Adequate, (ii) Data Are Sufficient for Model Justification, (iii) Data Uncertainty Is Characterized and Propagated Through the Model Abstraction, (iv) Model Uncertainty Is Characterized and Propagated Through the Model Abstraction, and (v) Model Abstraction Output Is Supported by Objective Comparisons.

3.3.3.4.1 System Description and Model Integration Are Adequate

Overall, the current information, along with agreements reached between DOE and NRC (Section 3.3.3.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess quantity and chemistry of water contacting waste packages and waste forms with respect to system description and model integration.

The DOE technical bases for including or excluding the features, events, and processes related to the Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms Integrated Subissue are provided in the three analysis and model reports (CRWMS M&O, 2000b,c,d). Staff questions with the technical bases provided by DOE for several of these features, events, and processes. Staff comments on FEP 2.1.08.07.00 (Pathways for Unsaturated Flow and Transport in the Waste and Engineered Barrier Subsystem), address a key model integration/model abstraction concern and are most appropriately discussed in this section. The following paragraphs also provide review comments on the conceptual and modeling approach developed by DOE to integrate features, events, and processes affecting the quantity and chemistry of water in Total System Performance Assessment–Site Recommendation abstractions.

To develop the Total System Performance Assessment–Site Recommendation (CRWMS M&O, 2000e), site and design information were fed into detailed process-level models, the process-level models were abstracted for use in the Total System Performance Assessment–Site Recommendation and the inputs and outputs from the various model abstractions were integrated for internal consistency. Two of the nine groups of process-level model abstractions used in the Total System Performance Assessment–Site Recommendation directly relate to the Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms Integrated Subissue: (i) Unsaturated Zone Flow and (ii) Engineered Barrier Subsystem Environments.

The Unsaturated Zone Flow abstraction, presented in CRWMS M&O (2000f), outputs a seepage flux into the drift for the time the drift wall temperature is below boiling and, thus, provides the time-dependent quantity of seepage water that enters the emplacement drift for the majority of the 10,000-year compliance period. During the boiling period, seepage fluxes are calculated using two analysis and model reports (CRWMS M&O, 2000g,h) that evaluate the possibility that coupled effects on flow would significantly alter flow pathway, and conclude that secondary phases precipitate in volumes too small to alter rock permeabilities. Hence, seepage fluxes under both ambient and thermally perturbed conditions are taken directly from thermal-hydrological models without chemistry. Staff find this approach reasonable, but are concerned that current DOE models may not address all important features, events, and processes in models calculating seepage flux into the proposed emplacement drifts. Discussion of those concerns and associated DOE agreements follow.

DOE neglect of mineral precipitation in the vicinity of the emplacement drifts is based on the results of simulations described in the analysis and model report (CRWMS M&O, 2000h). These multiphase reactive transport simulations require special handling of mass transport and mineral reactions near computational cells that have dried completely because of vigorous heating. Some approaches to handling dry computational cells in reactive transport simulations artificially inhibit mineral precipitation at the position of the boiling front. CRWMS M&O (2000h) did not provide enough detail to determine if the simulations adequately represent mineral precipitation at the boiling front. DOE agreed[1] to provide additional documentation on the simulations pertaining to quantity of unreacted solute mass trapped in the dryout zone in TOUGHREACT simulations as well as on how this mass would affect precipitation and the resulting change in hydrologic properties.

The present DOE multiscale thermal-hydrological model approach does not adequately represent what may be called the cold-trap effect (i.e., mass movement along the length of drift, resulting from thermal gradients, causing condensation in cooler regions). This process may have occurred in the enhanced characterization of the repository block drift when it was isolated from the ventilation system by a bulkhead to allow re-equilibration to unventilated conditions. Dripping has been observed {e.g., ~10 to 30-cm [4- to 12-in]} diameter puddles, wet drip cloths, and corroded metal) in the sealed portion of the enhanced characterization of the repository block. This dripping may result from vapor-phase mobilization of water and condensation on surfaces such as rock bolts, ventilation ducts, and utility conduits under small thermal gradients. In an unventilated near-field environment where waste-canister heat causes spatial temperature variability, this process could result in significant localized dripping. It is likely that condensate would react with metal and grout at elevated but below-boiling temperatures. Alternatively, dripping in the enhanced characterization of the repository block may have resulted from seepage into the drift. DOE data at present are insufficient to distinguish what processes are primarily responsible for the observed dripping. Dripping from condensation may be masking observation of dripping from seepage. Current DOE testing in the Enhanced Characterization of the Repository Block is directed toward distinguishing the processes.

DOE has not provided an adequate evaluation of the potential cold-trap effect, but has provided a reasonable approach to do so by the time of license application, based on DOE agreements to provide additional documentation.[2] As agreed, DOE will represent the cold-trap effect in the appropriate models or provide the technical basis for exclusion of it in the various scale models (mountain, drift, and such) considering thermal effects on flow and other abstractions/models (e.g., chemistry). DOE will represent the cold-trap effect in the analysis and model report (CRWMS M&O, 2000g). This report will provide technical support for inclusion or exclusion of the cold-trap effect in the various scale models. The analysis will consider thermal effects on flow and the in-drift geochemical environment abstractions. In addition, DOE should assess the processes responsible for the observed evidence of dripping in the enhanced characterization of repository block (i.e., vapor transport and condensation or seepage) and incorporate those processes into model abstractions, if appropriate. Because the compositions of seepage and condensation water are likely to differ significantly, the additional documentation to be provided by DOE is expected to evaluate the impact of the cold-trap effect on water and gas compositions in the emplacement drifts. DOE agreed to provide a technical basis for representation of or the neglect of dripping from rockbolts in the Enhanced Characterization of

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[1]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[2]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Thermal Effects on Flow (January 8–9, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

the Repository Block in performance assessment, including the impacts on hydrology, chemistry, and other impacted models at the Total System Performance Assessment and Integration Technical Exchange.[3]

In the Total System Performance Assessment–Site Recommendation (CRWMS M&O, 2000e), the composition of seepage waters is allowed to evolve in the engineered barrier subsystem environment through evaporation and salt formation processes and by variations in flow pathways within the engineered barrier subsystem. The analysis and model report (CRWMS M&O, 2001a) describes evaporation and salt formation processes in the engineered barrier subsystem by integrating two submodels, a high relative humidity model and a low relative humidity model. The high relative humidity model is represented by EQ3/6 Pitzer calculations. As these calculations are only verifiable up to an ionic strength of 10 molal, the high relative humidity model can only be used for relative humidities above 85 percent. At relative humidities lower than 85 percent, DOE employed a low relative humidity model, based on a mass balance approach. During the time relative humidity rises from 50 to 85 percent, the low relativity model simulates brine generation. This period is divided into equal time increments. For each time interval, DOE assumes that half the dissolved amount in the previous interval flows out of the local system or reactor. Currently, staff have no specific concerns related to system description/model integration for evaporation and salt formation. The staff, however, have general integration concerns related to the near-field environment, which are discussed later in this section.

DOE has not adequately defined the near-field geochemical environment that may be important to drip shield and waste package performances. Without a complete inventory of material that would be left in and surrounding the emplacement drifts after closure (i.e., a complete design), DOE predictions of the environment pertinent to drip shield and waste package performances are not adequate. Although current abstractions attempt to address some material that would be left in the emplacement drifts, elemental composition information for these materials is limited to major components and elements. DOE has not provided information on trace elements that may be important to the performances of the drip shield and waste package. In addition, because flow paths and the reaction pathways for fluid interaction are defined by local conditions, global and batch calculations do not capture the range of potential fluid compositions possible or the impact on repository performance. DOE agreed[4] to provide the technical basis for bounding the trace elements, including fluoride, for the geochemical environment affecting the drip shield and waste package, including the impact of engineered materials. DOE will document the concentrations of trace elements and fluoride in waters that could contact the drip shield and waste package in a revision to the analysis and model report (CRWMS M&O, 2000i). In addition, trace elements and fluoride concentrations in introduced materials in the engineered barrier subsystem (including cement grout, structural steels, and

[3]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[4]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

other materials, as appropriate) will be addressed in a revision to the analysis and model report (CRWMS M&O, 2000j).

The technical basis for selecting, including, and excluding specific coupling relationships from the Total System Performance Assessment–Site Recommendation (CRWMS M&O, 2000e) is not transparent and traceable in all cases. One of the major assumptions of the DOE modeling approach for the total system performance assessment, for example, is that coupled thermal-hydrological-chemical processes can be decoupled, evaluated separately, and then recoupled, without adversely affecting predictions of repository performance. DOE has not yet provided a transparent list of the criteria used to distinguish between included and excluded couplings or an adequate technical basis for modeling decisions based on those criteria. DOE agreed[5] to identify specific coupling relationships included and excluded from total system performance assessment, including Onsager couples, and give technical bases for their inclusion or exclusion. The information will be documented in a revision to the process model report (CRWMS M&O, 2000k).

DOE has not yet provided a complete characterization of the dust expected to settle on engineered materials in the proposed repository drift environment or an analysis of how dust could affect the chemistry of water contacting the waste packages and drip shields. DOE agreed[6] to provide documentation regarding the deposition of dust and its impact on the salt analysis. DOE will document dust sampling in the Exploratory Studies Facility, analyze the dust, and evaluate its impact on the chemical environment on the surface of the drip shield and waste package in a revision to the analysis and model report (CRWMS M&O, 2000j).

Staff view integration between process model abstractions in the Total System Performance Assessment–Site Recommendation as well as between supporting process-level model analysis and model reports as a key factor in the robustness of the DOE safety case. Staff will, therefore, continue to evaluate the architecture of the Total System Performance Assessment–Site Recommendation (CRWMS M&O, 2000e) as new information becomes available.

Several integration concerns related to near-field environment models and data were expressed at the DOE and NRC technical exchange.[7] NRC was concerned with both integration between models and analyses and integration within models and analyses. The staff expressed concerns that the corrosion testing to define the potential (or lack thereof) for localized corrosion and the magnitude and variability in general corrosion was not sufficiently integrated with projections of potential in-drift environmental conditions. DOE and NRC reached

[5]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[6]Ibid.

[7]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

agreement that DOE would complete corrosion testing in the predicted chemical environments or provide a technical basis as to why it is not needed. In addition, DOE would provide, in future documentation, a comparison of the environments predicted to those used in corrosion testing.

An area of concern for model integration and model abstraction is the screening status that DOE provided for FEP 2.1.08.07.00 (Pathways for Unsaturated Flow and Transport in the Waste and Engineered Barrier Subsystem). This feature is listed as included with regard to pathways for unsaturated flow and transport in the waste and engineered barrier subsystem in the DOE features, events, and processes database (CRWMS M&O, 2000c). This item evaluates unsaturated flow and radionuclide transport that may occur along preferential pathways in the waste form and engineered barrier subsystem. The technical basis DOE gives for the status of this item is that preferential pathways are already included via a series of linked one-dimensional flowpaths and mixing cells representing chemical evolution of the engineered barrier subsystem (CRWMS M&O, 2000c). Staff are concerned that preferred pathways in the engineered barrier subsystem are not being evaluated at the appropriate scale and, therefore, that potentially important aspects of this feature have not been included in the Total System Performance Assessment–Site Recommendation (CRWMS M&O, 2000e). Water has been observed to drip preferentially from grouted rock bolts in the enhanced characterization of the repository block (e.g., demonstrating that the introduced structures and materials themselves can influence the location of preferred flow pathways). Interactions with engineered materials, such as cementitious and metallic components, can have a significant effect on evolved water and gas compositions. DOE agreed[8] to address NRC concerns about screening arguments for features, events, and processes.

Also, in the DOE features, events, and processes database (CRWMS M&O, 2001b), the description for FEP 2.1.08.07.00 (Pathways for Unsaturated Flow and Transport in the Waste and Engineered Barrier Subsystem) includes the statement that physical and chemical properties of the engineered barrier subsystem and waste form, in both intact and degraded states, should be considered in evaluating (preferential) pathways. Hence, staff expect the screening arguments to be based on an evaluation of these topics (Evolution of the Near-Field Environment Issue Resolution Status Report Revision 03). DOE should explicitly include the possibility of localized flow pathways in the engineered barrier subsystem in total system performance assessment calculations, including the influence of introduced materials on these pathways, or provide adequate technical bases for not including this feature, event, and process. DOE agreed[9] to address NRC concerns about screening arguments for features, events, and processes.

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[8]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[9]Ibid.

Features, events, or processes that DOE is evaluating according to DOE and NRC technical exchange agreements[10] for key technical issues other than evolution of the near-field environment, could, depending on the nature of the process-level model results, significantly alter water compositions in the evolution of the near-field environment. DOE should take a broad approach toward integrating key features, events, and processes between integrated subissues. The list of key technical issue subissues in Section 3.3.3.2 provides a useful resource for considering the appropriate extent of integration between the Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms Integrated Subissue and other areas of research. For clarity, a few examples of areas that may require integration efforts with the Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms Integrated Subissue are identified next.

If the drifts were to collapse onto the drip shields/waste packages, the lifetime of the drip shields/waste packages could be altered by local variations in relative humidity and chemical compositions of water and gas developed in voids between the collapsed rocks. Also, temperature versus time profiles for the drip shield/waste package surfaces may differ significantly for scenarios where the drifts do and do not collapse, resulting in different estimates of water chemistry. NRC staff review of the analysis and model report (NRC, 2000g) concluded that DOE has not provided a satisfactory basis for screening out drift collapse because thermal and seismic loadings were not represented satisfactorily in the documented analyses. DOE has agreed[11] to provide an adequate path forward for the analysis of thermal-mechanical effects, but does not relate those issues to the quantity and chemistry of water contacting the waste packages and waste forms. If future DOE analyses indicate that drift collapse is likely, the impact of drift collapse on water and gas chemistries in the engineered barrier subsystem would be evaluated. DOE agreed[12] to evaluate spatial heterogeneity on unsaturated zone flow, seepage into drifts, and transport for both ambient and drift collapse conditions.

Another concern is the thermal-mechanical effects on hydrological properties (see Section 3.3.2 of this report). DOE proposed an evaluation of thermal-mechanical effects on hydrological properties based on analyses of localized thermally induced rock response near a heated drift (CRWMS M&O, 2000l; DOE, 2001). An important case of fracture-aperture changes in the pillar between two heated drifts was not considered in the DOE analyses, however. An increase in the aperture of subhorizontal fractures in the pillar from thermal-mechanical effects is possible and would be important to cross-repository water flow because of the potential diversion of water flux from the pillar to one of the adjacent drifts, thereby focusing flux toward

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[10]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[11]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Thermal Effects on Flow (January 8–9, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[12]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

the drift.[13] If future DOE evaluations of thermal-mechanical effects on hydrological properties indicate significant focused flow toward the drift, DOE would also evaluate the impact of the focused flow on repository performance with respect to the quantity and chemistry of water in the engineered barrier subsystem. DOE agreed[14] to consider this particular scenario.

In summary, system description and model integration for quantity and chemistry of water contacting waste packages and waste forms are not yet adequate. DOE agreed[15] to address these concerns in future documents.

3.3.3.4.2 Data Are Sufficient for Model Justification

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.3.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess quantity and chemistry of water contacting waste packages and waste forms with respect to data being sufficient for model justification.

In the Total System Performance Assessment–Site Recommendation (CRWMS M&O, 2000e), response surfaces describing temperature, humidity, liquid saturation, pH, total carbonate, ionic strength, and seepage flux are evaluated by the Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms Integrated Subissue. The process-level and conceptual models used to define these response surfaces depend on a wide range of information and data including waste form properties, engineered barrier subsystem material properties, drip shield and waste package design properties, repository design properties, site geohydrology, and site geochemistry. Because of the inherent interconnectedness between the Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms Integrated Subissue and other key technical issues and integrated subissues (see Section 3.3.3.2), DOE should evaluate staff comments raised in other sections of this report for applicability to the Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms Integrated Subissue.

Currently, insufficient data are being used to constrain the chemistry of brine solutions under low relative humidity conditions in the analysis and model report (CRWMS M&O, 2001a). Drip shield and waste package degradation are expected to be most active when the repository is still hot and the deliquescent humidity has been reached. The characterization of high ionic strength solution chemistries (e.g., greater than 10 molal) in complex natural environments exceeds the limitations of the DOE modeling approach and may be best characterized by experimental data. Interpolation techniques DOE used in the low relative humidity model are

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[13]Ofoegbu, G.I., S. Painter, R. Chen, R.W. Fedors, and D.A. Ferrill. "Geomechanical and Thermal Effects on Moisture flow at the Proposed Yucca Mountain Nuclear Waste Repository." *Nuclear Technology*. Vol. 134. In press. June 2001.

[14]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[15]Ibid.

insufficient to constrain the models. Modeling results, such as predicted concentrations of major anionic and cationic species, are inadequately described.

Also, current DOE technical bases are not adequate to justify the assumption that pure salts will define the deliquescent humidity. The minimum deliquescent humidity of a salt mixture is typically lower than the individual deliquescent humidity. Although 50 percent is the lowest deliquescent humidity for all the pure salts considered in CRWMS M&O (2001a), the deliquescent humidity of the salt mixture in the Yucca Mountain waters may be even lower.

To address the concerns in the preceding paragraphs, DOE agreed[16] to provide a revision of the analysis and model report (CRWMS M&O, 2001a) that includes (i) the major anionic (e.g., fluoride or chloride) and cationic species and (ii) additional technical bases for the low relative humidity model. The data should provide the technical basis why the assumption of the presence of sodium nitrate is conservative, when modeling and experimental results indicate the presence of other mineral phases for which the deliquescence points are unknown. DOE will provide additional information to constrain the low relative humidity salts model. The information will include the deliquescent behavior of mineral assemblages derived from alternative starting water compositions (including bulk water compositions and local variations associated with cement leaching or the presence of corrosion products) representing the range of potential water compositions in the emplacement drifts.

In view of safety insights achieved since the DOE and NRC technical exchange[17] staff reassessed the analysis and model report (CRWMS M&O, 2001a) and identified several repository performance concerns. Although DOE considers evaporation and salt formation processes in the engineered barrier subsystem throughout the 10,000-year compliance period, NRC staff continue to focus its review of these models on the initial deliquescent period, when brine solutions are likely to be most corrosive. Staff concerns are described in the following two paragraphs.

The analysis and model report (CRWMS M&O, 2001a) assumes that during discrete time intervals, half the dissolved amount in the previous time interval would flow out of the reactor as soon as a minimum relative humidity is reached. For soluble species such as nitrates, the fraction, $f_{i,(k-1)/2}$, in Eq. 3 (CRWMS M&O, 2001a) is unity, and all the accumulated nitrate salts would be dissolved in the liquid. Hence, DOE would predict that most accumulated salt will flow out of the reactor in a few time intervals. Staff are concerned that this may be an unrealistic artifact of the modeling approach. The following scenario was predicted by staff evaluations, and has raised concerns with the DOE models. In this scenario, only a small amount of concentrated liquid is present after the deliquescent humidity is reached. The concentrated liquid stays in the pores of the deposits until the liquid volume reaches a point where the solid deposit could no longer hold the liquid. This scenario does not agree with the rapid depletion of accumulated salts predicted by DOE models following the onset of deliquescence, and

[16]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[17]Ibid.

suggests DOE may be underestimating the amount of nitrate salts on the waste package. DOE should provide stronger technical bases for the approach used in the low relative humidity model during the initial stages of deliquescence, when the potential for corrosion is highest. As part of the DOE and NRC technical exchange,[18] DOE agreed to provide technical basis for the simplifications used when developing model abstractions.

In the analysis and model report (CRWMS M&O, 2001a), DOE assumes all accumulated nitrate salts are dissolved in the liquid as soon as a minimum relative humidity is reached. The total calculated volume of water is large at this time, causing concentrations for other important species such as Cl^- to be extremely low. Staff are concerned the DOE model may significantly underestimate concentrations of these aqueous species following deliquescence. DOE should provide additional technical bases explaining why these species concentrations are not limited by solubilities of the salts. DOE agreed[19] to provide additional technical bases for the low relative humidity salts model, including the major anionic and catronic species.

The data DOE used to calibrate and validate several process-level models providing input into total system performance assessment are not adequate, and the technical reliability and representativeness of these data have not been adequately evaluated. A significant amount of experimental data was collected for use in the analysis and model report (CRWMS M&O, 2000h). Insufficient analyses, however, were performed to interpret the data and to establish that parameter values are bounded. In addition, the criterion used to include and exclude individual water and gas measurements for use in these models has not been clearly documented. Similar concerns exist about the reliability data used to validate and calibrate the analysis and model report (CRWMS M&O, 2001a). Finally, validation efforts can only be as robust as the data they are being validated against. Thus, DOE should fully scrutinize the reliability of data used to validate this model and provide this information to NRC staff for review. DOE did not make a transparent distinction between calibration and validation efforts in either of these analysis and model reports. Data should be used to either calibrate or validate, but not to simultaneously calibrate and validate.

To address the previous concerns, DOE agreed[20] to provide additional documentation on the data used to calibrate models and to support model predictions. In addition, the DOE agreed to assess data uncertainty (e.g., sampling and analytical), including critical analyses of variables that affect the data measurements and their interpretations (e.g., drift-scale thermal and evaporation tests). DOE will provide documentation of data used to calibrate models and of data to support model predictions, together with an assessment of data uncertainties (e.g., sampling and analytical) in the areas of water and gas chemistries, from the drift-scale

[18]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[19]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[20]Ibid.

thermal tests and evaporation tests. This documentation will be provided in revisions to the analysis and model report (CRWMS M&O, 2000i) or in another future document.

DOE has not demonstrated that water and gas chemistry analyses used as initial conditions in process-level models supporting total system performance assessment calculations are appropriately bounding. The level of detailed information DOE provided on the full water chemistry, including trace metals potentially important to drip shield and waste package performance, is insufficient.

To address the previous concern, DOE agreed[21] to provide additional information about the range of water composition that could contact the drip shield or waste package, including whether such waters are of the bicarbonate or chloride-sulfate type. DOE will describe the range of bulk composition for waters that could affect corrosion of the drip shield or waste package outer barrier in a revision to the analysis and model report (CRWMS M&O, 2000i).

DOE has not yet provided sufficient data to support models of coupled thermal-hydrological-chemical processes on the waste package environment. Silica mobility may play an important role in models predicting the quantity and chemistry of water contacting the waste packages, but kinetic parameters for the silicate phases present at Yucca Mountain are poorly understood. DOE has not sufficiently constrained coupled thermal-hydrological-chemical models of Yucca Mountain with site-specific experimental data.

To address the previous concern, DOE agreed[22] to provide documentation of the results obtained from the crushed tuff hydrothermal column experiment and of posttest analysis in new reports specific to the column test.

DOE has not adequately considered changes in local water and gas chemistries because of interactions with engineered materials, such as grouted rock bolts, along preferential flow pathways. The current total system performance assessment approach weights the volumetric contribution made by local variations in water and gas chemistries against bulk engineered barrier subsystem water and gas chemistries to evaluate the potential impact of local chemistry on repository performance. Staff are concerned this approach does not adequately address the potential impact that preferential pathways could have on the chemistry of water contacting the drip shield and waste packages, because it does not allow the composition of water moving along these pathways to deviate from the bulk engineered barrier subsystem composition.

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[21]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[22]Ibid.

To address this concern, DOE agreed[23] to provide the analyses of laboratory solutions that have interacted with introduced materials. DOE will provide additional information about laboratory solutions that have interacted with introduced materials in a revision to the analysis and model report (CRWMS M&O, 2000i). DOE will also reevaluate the impact of these water compositions in the context of preferential flow pathways in the total system performance assessment and repository performance.

NRC staff expressed concern that unmeasured loss of mass from the heated drift complicates analysis of the Drift Scale Test results and may ultimately compromise utility of the Drift Scale Test for evaluating refluxing during the thermal phase of the proposed repository design. DOE maintained that "more accurate characterization of the heat loss through the bulkhead" is difficult, problematic, and unnecessary (CRWMS M&O, 1999). Because of concerns regarding these uncertainties, however, DOE decided to take a dual approach to quantifying mass and energy losses through the bulkhead (CRWMS M&O, 2000m). First, a proposal by the University of Nevada to measure losses in a manner requiring sealing of the cable bundles and other leakage through the bulkhead would be pursued. Second, the DOE thermal test team would deploy a series of humidity and temperature sensors along the drift immediately outside the bulkhead. Muffin fans would be used to ensure proper air movement and to prevent condensation. Both approaches would be implemented in fiscal year 2001 if funding were approved (CRWMS M&O, 2000m). DOE reversed this position at the DOE and NRC Thermal Effects on Flow Technical Exchange,[24] however, stating that measuring mass and energy losses through the bulkhead of the Drift Scale Test is not necessary for the intended use of the Drift Scale Test results.

To address these concerns, DOE agreed[25] to provide additional documentation to address mass and energy losses through the Drift Scale Test bulkhead. In addition, DOE will provide NRC with a White Paper on the technical basis for DOE understanding of heat and mass losses through the bulkhead. This White Paper will address uncertainty in the fate of thermally mobilized water in the Drift Scale Test and also the effect this uncertainty has on conclusions drawn from the Drift Scale Test results. NRC will provide comments on this white paper. DOE will analyze the effects of this uncertainty on the uses of the Drift Scale Test in response to NRC comments.

DOE data from ventilation testing are not sufficient to support the ventilation model. The design objective of maintaining pillar temperatures below boiling to allow for condensate drainage between emplacement drifts depends on the efficacy of the ventilation system. The analysis and model report (CRWMS M&O, 2000n) shows 70 percent heat removal by ventilation flow

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[23]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[24]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Thermal Effects on Flow (January 8–9, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[25]Ibid.

rates between 10 and 15 m^3/s [350 and 530 ft^3/s]. This model involves simplifying assumptions, however, and is not supported by experimental data. Plans have been developed for a quarter-scale ventilation test to be conducted at the Engineered Barrier Subsystem Test Facility in North Las Vegas, Nevada (CRWMS M&O, 2000n). This test needs to be completed to provide data for support and verification of the ventilation model. To address these concerns, DOE will provide[26] the detailed test plan for Phase III of the ventilation test. NRC comments on the test plan will be considered by DOE before initiation. DOE will provide the analysis and model report (CRWMS M&O, 2000n) and CRWMS M&O (2000o). Test results will be provided in an update to CRWMS M&O (2000n).

In summary, data sufficiency and model justification for quantity and chemistry of water contacting waste packages and waste forms are not yet adequate, but DOE agreed[27] to address staff concerns in future documents.

3.3.3.4.3 Data Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.3.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess quantity and chemistry of water contacting waste packages and waste forms with respect to data uncertainty being characterized and propagated through the model abstraction.

Uncertainties in data used to constrain individual in-drift geochemical submodels have yet to be adequately evaluated and documented, and the impact of these uncertainties on the predicted quantity and chemistry of water contacting the waste packages and waste forms has not yet been propagated through total system performance assessment calculations. DOE agreed[28] to evaluate data and model uncertainties for specific in-drift geochemical environment submodels used in total system performance assessment calculations and propagate those uncertainties. DOE will document the evaluation in an update to the analysis and model report (CRWMS M&O, 2000j) (or in another future document). DOE also agreed to address the various sources of uncertainty [e.g., model implementation, conceptual model, and data uncertainty (hydrologic, thermal, and geochemical)] in the thermal-hydrological-chemical model. DOE will evaluate the various sources of uncertainty in the thermal-hydrological-chemical process model, including details on how the propagation of various sources of uncertainty is calculated in a systematic uncertainty analysis, and document those sources in a revision to the analysis and model report (CRWMS M&O, 2000h) (or in another future document).

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[26]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[27]Ibid.

[28]Ibid.

Several concerns related to propagation of uncertainty in near-field environment models and data were expressed at the DOE and NRC technical exchange.[29] The DOE models on the near-field environment propagated a limited amount of uncertainty from upstream sources. Staff expressed concern that the uncertainty in the environmental conditions generated by the DOE models did not adequately propagate all significant sources of uncertainty, therefore, leading to an underprediction in the range of expected environmental conditions. DOE agreed[30] to address the NRC concerns in several agreements.

DOE thermal-hydrological calculations used to support seepage fluxes do not currently account for measurement error, bias, and scale dependence in the saturation, water potential, and pneumatic pressure data. Standard deviation of saturation data from cores was used to estimate weights for the weighted least-squares inverse algorithm (CRWMS M&O, 2000p), however, the effect of measurement errors on the resulting calibrated properties was not evaluated. Three types of data (matrix saturation from cores, water potential from boreholes, and pneumatic pressures) were measured on different scales ranging from a few centimeters for cores to several tens of meters or more for pneumatic pressures. Matrix saturations from core data were upscaled by arithmetic averaging, a process that tends to smooth out variability; but it is not clear how the scale dependence of the water potentials and pneumatic pressure data were treated. Pneumatic pressure data are known to be scale dependent because fracture permeabilities from barometric pumping responses tend to be about two orders of magnitude greater than fracture permeabilities determined from air-injection testing (CRWMS M&O, 2000p). This information is important because property sets developed in the calibrated properties analysis and model report are used in the unsaturated zone flow models (and multiscale thermohydrological model) essentially deterministically. That is, a single property set for each high-, median-, and low-infiltration condition is assumed to capture all the variability and uncertainty in the model. Propagation of uncertainty from unsaturated zone and multiscale thermohydrologic process models to model abstractions necessitates incorporating all sources of uncertainty.

The nonlinear least-squares maximum likelihood inverse method implemented in ITOUGH2 accounts for uncertainty through measurement error. Thus, the measurement error must be generalized to include other sources of uncertainty, such as scale dependence and modeling errors, because there is no other way to account for uncertainty in the least-squares inverse approach (McLaughlin and Townley, 1996).

DOE presented a discussion of the conceptual model used to develop the calibrated property sets used in the analysis and model report (CRWMS M&O, 2000o) stating that [h]eterogeneity of hydrologic properties is predominantly a function of geological layering, and therefore, each geological layer in the model is treated as homogeneous. The resulting average layer-calibrated, layer-averaged, drift-scale property sets for the basecase show fracture permeability in the Topopah Spring (Tsw34) unit to be 2.76E–13 m^2 [2.97E–12 ft^2] and in the Tsw35 unit to

[29]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[30]Ibid.

be 1.29E–12 m^2 [1.39E–11 ft^2]. For the upper-bound infiltration map, these change to 4.63E–13 m^2 [4.98 E–12 ft^2] and 5.09E–12 m^2 [5.48E–12 ft^2] and for the lower-bound, to 4.99E–13 m^2 [2.97E–12 ft^2] and 1.82E–12 m^2 [1.96E–11 ft^2] for the Tsw34 and Tsw35 units, respectively. Thus, all the variability and uncertainty in model layer fracture permeability for these two units ranges within approximately one order of magnitude. A statistical analysis of air-injection data collected from the niches in the Exploratory Studies Facility, however, found fracture permeabilities ranging from 1.53E–15 m^2 to 7.15E–10 m^2 [1.65E–14 to 7.70 E–9 ft^2]. These data, collected in the Tsw34 unit, indicate that heterogeneity of fracture permeability can range at least four orders of magnitude within a single geological layer. The DOE use of homogeneous layer properties in a model, with variability ranging only one order of magnitude, does not adequately represent variability and uncertainty that may range several orders of magnitude within a single geological layer.

To address these concerns, DOE agreed[31] to provide additional documentation to address data uncertainty that will be provided as part of the issue resolution process and, if provided by DOE by the time of any license application, should afford sufficient information for NRC to conduct its licensing review. DOE will provide documentation of analyses of spatially heterogeneous fracture permeability using refinement of the grid for the heterogeneous fields in three dimensions and will evaluate the effect of high-permeability features (e.g., faults) crossing the drifts. DOE will consider the NRC suggestion to compare the numerical model results with the Phillips (1996) analytical solution.

In summary, data uncertainty being characterized and propagated through the model abstraction with regard to quantity and chemistry of water contacting waste packages and waste forms is not yet adequate, but DOE agreed[32,33] to address all staff concerns in future documents.

3.3.3.4.4 Model Uncertainty Is Characterized and Propagated Through the Model Abstraction

Overall, the current information, along with agreements reached between the DOE and NRC (Section 3.3.3.5), is sufficient to ensure that the information will be available at the time of a potential license application to assess quantity and chemistry of water contacting waste packages and waste forms with respect to model uncertainty being characterized and propagated through the model abstraction.

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[31]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Thermal Effects on Flow (January 8–9, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[32]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[33]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Thermal Effects on Flow (January 8–9, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

DOE has not yet documented how different flow pathways impact total system performance assessment predictions of the Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms Integrated Subissue. Even for ambient conditions (Browning, et al., 2000), water and gas compositions will vary, depending on the types of materials encountered along a particular flow pathway and the duration of those interactions. The Total-system Performance Assessment code evaluates several different flow pathways in the engineered barrier subsystem, but does not adequately consider local changes in water and gas chemistries that may result from interactions with engineered materials, corrosion products, or both (such as cement-grouted rock bolts) located above the drip shield. Staff are also concerned that water, affected by these interactions may impact performance.

To address these concerns, DOE agreed[34] to evaluate the impact of the range of local chemistry (e.g., dripping of equilibrated evaporated cement leachate and corrosion products) conditions at the drip shield and waste package, considering the chemical divide phenomena that may propagate small uncertainties into large effects. DOE should also evaluate the range of local chemical conditions at the drip shield and waste package (e.g., local variations in water composition associated with cement leaching or the presence of corrosion products), considering potential evaporative concentration. This evaluation will be documented in a revision to the analysis and model report (CRWMS M&O, 2000j). DOE should determine whether calculated water compositions for various flow pathways in the engineered barrier subsystem are significant, given the uncertainties in the data and models.

Inadequate technical bases have been provided for the DOE major assumption that all reactions proceed to equilibrium. The suppression of mineral precipitation in process-level models supporting total system performance assessment is an acknowledgment of the role of kinetics, but DOE has not yet documented the criteria used to identify which mineral reactions are suppressed and the conditions under which the suppression of these reactions is applicable. To address this concern, DOE agreed[35] to provide stronger technical bases for the suppression of individual mineral reactions predicted by equilibrium models in a revision to the analysis and model report (CRWMS M&O, 2000j). DOE also agreed[36] to provide the technical basis for current treatment of the kinetics of chemical processes in the in-drift geochemical models in a revision to the analysis and model report (CRWMS M&O, 2000j). The technical basis will include reaction progress simulation for laboratory evaporative concentration tests and appropriate treatment of time as related to the residence times associated with the abstractions used to represent in-drift processes in total system performance assessment.

Two different models were presented in the analysis and model report (CRWMS M&O, 2000h) that provide some insight into model uncertainty. The two models differ mainly in the number of minerals and dissolved elemental components considered, and the model results show that the

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[34]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[35]Ibid.

[36]Ibid.

limited suite mineral model provides a closer match to Drift Scale Test data. DOE has not yet provided sufficient technical bases demonstrating the output from these two cases bound the quantity and chemistry of water contacting the waste packages and waste forms. In contrast to the DOE claim that infiltration rates are the major uncertainty in thermal-hydrological and thermal-hydrological-chemical models (CRWMS M&O, 2000e), NRC staff believe that uncertainties associated with the representation of complex chemical interactions are equally significant. Stronger technical bases are needed for the DOE exclusion of chemistry-related uncertainties in total system performance assessment abstractions of seepage compositions. Additional technical bases are also needed for the lack of spatial variability in Total System Performance Assessment–Site Recommendation Abstraction for Seepage Water and Gas Chemistries. To address these concerns, DOE agreed[37] to provide a revision of the analysis and model report (CRWMS M&O, 2000h) that includes information supporting both the limited suite mineral model and the more complete extended model. In addition, DOE should provide sufficient technical bases demonstrating that the output from these two cases bound predictions of the quantity and chemistry of water contacting the waste packages and waste forms, given that both models most closely approximate conditions near the center of the potential repository.

CRWMS M&O (2000g) uses only the drift-scale property sets to calculate thermohydrologic variables. It is not clear how this captures the variability and uncertainty seen in predictions using other property sets or the uncertainty in comparisons to actual test results. Note that, to date, all thermal tests at Yucca Mountain have been conducted in the middle nonlithophysal unit of the Tsw34, so all conclusions of the analysis and model report (CRWMS M&O, 2000q) apply only to that unit. Thus, it seems reasonable that if the analyses were performed on the remaining geological units, the predicted variability and uncertainty would be greater. Further, the analysis and model report (CRWMS M&O, 2000p) recommends that future studies should consider the use of Monte Carlo simulations to evaluate the appropriateness of using the prior information uncertainty for the calibrated properties. Such exercises would be useful for evaluating the propagation of uncertainty through the least-squares inverse approach, as discussed previously. This approach would not address the uncertainty inherent in spatial heterogeneity nor would it adequately address the uncertainty in the equally valid but significantly different models and property sets of CRWMS M&O (2000q). Additional studies applying generally accepted methods of stochastic subsurface hydrology, sensitivity, and bounding analyses would be required to address the data and model uncertainties.

To address this concern, DOE agreed[38] to provide additional documentation to address model uncertainty. DOE will represent the full variability/uncertainty in the results of the thermal effects on flow simulations in the abstraction of thermodynamic variables to other models or provide technical bases that a reduced representation is appropriate (considering risk

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[37]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001

[38]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Thermal Effects on Flow (January 8–9, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

significance). DOE will provide an updated calibrated properties analysis and model report (CRWMS M&O, 2000p) that incorporates uncertainty from all significant sources. DOE will consider model uncertainty, including (i) types of model uncertainty, (ii) flow conceptualization for ambient conditions, (iii) flow conceptualization for thermal conditions, (iv) fracture flow for ambient and thermal conditions, (v) fracture matrix interaction model evolution, (vi) discrete fracture description, and (vii) model uncertainty reduction.

In summary, characterization and propagation of model uncertainty through the model abstraction, as applied to quantity and chemistry of water contacting waste packages and waste forms, are not yet adequate. DOE agreed[39] to address staff concerns in future documents.

3.3.3.4.5 Model Abstraction Output Is Supported by Objective Comparisons

Overall, the current information along, with agreements reached between the DOE and NRC (Section 3.3.3.5), is sufficient to conclude that the necessary information will be available at the time of a potential license application to assess quantity and chemistry of water contacting waste packages and waste forms with respect to model abstraction output being supported by objective comparisons.

Although this integrated subissue deals with water in the drift, the composition of seepage water is likely influenced by the phases in the unsaturated fractured rock with which it reacts. Geochemical modeling has been used to predict the composition of the water seeping into the drifts. The predictions resulting from geochemical modeling are uncertain. Sources of uncertainty include the modeler decisions on components to include or exclude in the system studied, kinetics of reactions, surface areas of minerals and fractures and activity coefficients of species in the aqueous and solid phases. DOE agreed[40] to provide physical evidence to support the model of fracture/matrix interaction by overcoring in the Single Heater Test and side-wall sampling mineralogy/petrology of the Drift Scale Test. Comparison of pre and posttest mineral assemblages, looking for evidence of alteration, and redistribution can be used to support predictive models.

In addition to seepage water, increased attention is currently being paid to condensation. Evidence suggests that condensation is occurring behind the bulkhead of the Enhanced Characterization of the Repository Block, where conditions are unventilated, and relative humidity is high. DOE is conducting experiments to address concerns related to condensation. If the experiments suggest that a significant portion of the water that could contact the waste

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[39]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Evolution of the Near-Field Environment (January 9–12, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[40]Ibid.

packages and waste form is condensate, DOE agreed[41] to represent this process in appropriate models, including the thermal effects on flow and the in-drift geochemical environment.

DOE should provide model support by predicting thermohydrologic results of the Cross Drift Thermal Test to verify that the thermohydrologic model abstraction adequately represents the potential thermohydrologic conditions expected in the proposed repository. DOE should identify and implement a useful approach toward verifying total system performance assessment predictions of engineered barrier subsystem environments in the proposed repository setting. Numerical simulations are used to predict the occurrence (or lack) of mineral precipitation around the emplacement drifts. Conditions of above-boiling temperatures may persist for hundreds to thousands of years. Resulting from the numerous sources of uncertainty, strong model support is needed for the numerical result of no significant alteration around the emplacement drifts. Staff have discussed this concern with DOE. DOE stated that significant alteration has not been observed for the Drift Scale Test and that this provides sufficient support for the modeling result of limited mineral alteration around the emplacement drifts. The difficulty with using this piece of information as the primary support for the modeling result is the temporal scales associated with the processes. For instance, if the minerals in the Drift Scale Test were being altered at a rate of less than 1 percent per year, most of the characterization performed to date, or the observation of thermodynamic variables, would be unable to resolve the magnitude of the alteration. This amount of alteration during hundreds to thousands of years, however, could have significant impacts on the quantity and chemistry of water contacting the waste packages and waste forms. Additional sources of information (e.g., simulation of laboratory experiments on silica precipitation) should be used to provide additional model support. DOE agreed to address concerns related to model support in future documentation.[42]

In summary, DOE has not provided sufficient evidence, either through field tests or natural analogs, that modeling results of quantity and chemistry of water contacting waste packages and waste forms are sufficient for inclusion in license application. DOE agreed to address the concerns described previously with the results from field tests in the enhanced Characterization of the Repository Block, and from laboratory experiments.

3.3.3.5 Status and Path Forward

Table 3.3.3-1 provides the status of all key technical issue subissues, referenced in Section 3.3.3.2, for the Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms Integrated Subissue. The table also provides the related DOE and NRC agreements pertaining to the Quantity and Chemistry of Water Contacting Waste Packages and Waste Forms Integrated Subissue. The agreements listed in the table are associated

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[41]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Thermal Effects on Flow (January 8–9, 2001)." Letter (January 26) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

[42]Reamer, C.W. "U.S. Nuclear Regulatory Commission/U.S. Department of Energy Technical Exchange and Management Meeting on Total System Performance Assessment and Integration (August 6–10, 2001)." Letter (August 23) to S. Brocoum, DOE. Washington, DC: NRC. 2001.

with one or all five generic acceptance criteria discussed in Section 3.3.3.4. Note that the status and the detailed agreements (or path forward) pertaining to all the key technical issue subissues are provided in Table 1.1-3 and Appendix A.

The DOE-proposed approach, together with the DOE agreements to provide NRC with additional information (through specified testing, analyses, and the like), acceptably addresses the NRC questions so that no information beyond that provided, or agreed to, will likely be required at the time of a potential license application.

Table 3.3.3-1. Related Key Technical Issue Subissues and Agreements

Key Technical Issue	Subissue	Status	Related Agreements*
Evolution of the Near-Field Environment	Subissue 1—Effects of Coupled Thermal-Hydrological-Chemical Processes on Seepage and Flow	Closed-Pending	ENFE.1.01 ENFE.1.03 through ENFE.1.07
	Subissue 2—Effects of Coupled Thermal-Hydrological-Chemical Processes on the Waste Package Chemical Environment	Closed-Pending	ENFE.2.01 ENFE.2.03 through ENFE 2.18
	Subissue 3—Effects of Coupled Thermal-Hydrological-Chemical Processes on the Chemical Environment for Radionuclide Release	Closed-Pending	ENFE.3.01 ENFE.3.02 ENFE.3.03 ENFE.3.05
	Subissue 4—Effects of Coupled Thermal-Hydrological-Chemical Processes on Radionuclide Transport through Engineered and Natural Barriers	Closed-Pending	ENFE.4.01 ENFE.4.02 ENFE.4.03 ENFE.4.04
	Subissue 5—Effects of Coupled Thermal-Hydrological-Chemical Processes on Potential Nuclear Criticality in the Near Field	Closed-Pending	ENFE.5.01
Thermal Effects on Flow	Subissue 1—Features, Events, and Processes Related to Thermal Effects on Flow	Closed-Pending	TEF.1.01
	Subissue 2—Thermal Effects on Temperature, Humidity, Saturation, and Flux	Closed-Pending	TEF.2.01 TEF.2.02 TEF.2.05 through TEF.2.08 TEF.2.10 TEF.2.11

Table 3.3.3-1. Related Key Technical Issue Subissues and Agreements (continued)

Key Technical Issue	Subissue	Status	Related Agreements*
Container Life and Source Term	Subissue 1—The Effects of Corrosion Processes on the Lifetime of the Containers	Closed-Pending	None
	Subissue 3—The Rate at Which Radionuclides in Spent Nuclear Fuel Are Released from the Engineered Barrier Subsystem through the Oxidation and Dissolution of Spent Nuclear Fuel	Closed-Pending	CLST.3.02 CLST.3.04
	Subissue 4—The Rate at Which Radionuclides in High-level Waste Glass Are Released from the Engineer Barrier Subsystem	Closed-Pending	CLST.4.02 CLST.4.04
	Subissue 5—The Effects of In-Package Criticality on Waste Package and Engineered Barrier Subsystem Performance	Closed-Pending	CLST.5.01 CLST.5.05
	Subissue 6—The Effects of Alternate Engineered Barrier Subsystem Design Features on Container Lifetime and Radionuclide Release from the Engineered Barrier Subsystem	Closed-Pending	None
Radionuclide Transport	Subissue 4—Nuclear Criticality in the Far Field	Closed-Pending	RT.4.03
Repository Design and Thermal-Mechanical Effects	Subissue 3—Thermal-Mechanical Effects on Underground Facility Design and Performance	Closed-Pending	RDTME.3.20 RDTME.3.21
Unsaturated and Saturated Flow Under Isothermal Conditions	Subissue 4—Deep Percolation	Closed-Pending	None
Structural Deformation and Seismicity	Subissue 3—Fracturing and Structural Framework of the Geological Setting	Closed-Pending	SDS.3.03 SDS.3.04

Table 3.3.3-1. Related Key Technical Issue Subissues and Agreements (continued)			
Key Technical Issue	**Subissue**	**Status**	**Related Agreements***
Total System Performance Assessment and Integration	Subissue 1—System Description and Demonstration of Multiple Barriers	Closed-Pending	None
	Subissue 2—Scenario Analysis and Event Probability	Closed-Pending	TSPAI.2.01 TSPAI.2.02
	Subissue 3—Model Abstraction	Closed-Pending	TSPAI.3.07 through TSPAI.3.13
	Subissue 4—Demonstration of Compliance with the Postclosure Public Health and Environmental Standards	Closed-Pending	None

*Related DOE and NRC agreements are associated with one or all five generic acceptance criteria.

Note: Key Technical Issue Agreement GEN.1.01 pertains to multiple integrated subissues, as well as some specific issues related to this integrated subissue.

3.3.3.6 References

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