**Department of Energy**
Washington, DC 20585

MAY 21 1992

Mr. Joseph J. Holonich, Director
Repository Licensing and Quality
Assurance Project Directorate
Division of High-Level Waste Management
Office of Nuclear Material Safety
and Safeguards
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Holonich:

As requested by Mr. Kenneth Hooks at the QA Bi-Monthly technical exchange on April 30, 1992, the enclosed consolidation and transition plans are being forwarded for your information. In addition, technical direction letters initiating the plans are also enclosed.

Should you require additional information, please contact Chris Einberg at (202) 586-8869.

Sincerely,

John P. Roberts

John P. Roberts
Acting Associate Director for
Systems and Compliance
Office of Civilian Radioactive
Waste Management

6 Enclosures: on the shelf

Configuration Management Transition Plan, Revision 0, April 1992

Plans and Procedures Organization Transition Plan, Revision 0, March 1992

Yucca Mountain Project, Las Vegas, Local Record Center's Consolidation Plan, Revision 0, February 1992



Ltr. Encl.

ADD: Ken Hooks

102.8
WM-11
NH03

Technical Direction Letter (TDL) - Transition Plan for Change Control Board (CCB) Support

Technical Direction Letter (TDL) - Transition Plan for Plans and Procedures Division (PPD) Support

Technical Direction Letter (TDL) - Transition Plan for Local Records Center (LRC)

cc w/Enclosures:
C. Gertz, YMPO
K. Hooks, NRC
R. Carlson, NRC
R. Loux, State of Nevada
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received with letter dtd 4/5/91

YMP/CM-0011

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U.S. DEPARTMENT OF ENERGY

OCRWM

YUCCA MOUNTAIN

# YUCCA MOUNTAIN SITE CHARACTERIZATION PROJECT

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# SITE CHARACTERIZATION PROGRAM BASELINE

## REVISION 1

## VOLUME 4 OF 5

102.8

MARCH 1991

UNITED STATES DEPARTMENT OF ENERGY

U.S. DEPARTMENT OF ENERGY

OCRWM

# YUCCA MOUNTAIN SITE CHARACTERIZATION PROJECT

Document No. YMP/CM-0011
Revision 1
CI No. CI.11.0000/CI.13.0000
Date 4/5/91
WBS No. 1.2.3
QA Level Yes

## PROJECT BASELINE DOCUMENT

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# YUCCA MOUNTAIN SITE CHARACTERIZATION PROGRAM BASELINE (SCPB) VOLUME 4

*CHANGES TO THIS DOCUMENT REQUIRE PREPARATION AND APPROVAL OF A CHANGE REQUEST IN ACCORDANCE WITH PROJECT AP-3.3Q*

DEPARTMENT OF ENERGY · UNITED STATES OF AMERICA

UNITED STATES DEPARTMENT OF ENERGY
YUCCA MOUNTAIN SITE CHARACTERIZATION PROJECT OFFICE

Y-AD-057
9/90

# YUCCA MOUNTAIN PROJECT
# CHANGE DIRECTIVE (CD)

1 CR No. 91/052
Page 1 of 2

## SECTION I. IDENTIFICATION

Title of Change:

Submittal of the "Site Characterization Program Baseline, Rev. 1," for CCB Control

3 Change Classification:
☐ Class 1 ☐ Class 3
☒ Class 2

## SECTION II. DISPOSITION

4 CR Disposition:
☐ Approved ☐ Disapproved
☒ Approved with Conditions

5 Conditions: *(if applicable)*

The next revision of this document should incorporate the following items:

1. The term "Exploratory Shaft Facility" should be replaced by "Exploratory Studies Facility" throughout the document.

2. The term "repository" should be replaced by "potential repository" throughout the document.

(See Change Documentation Continuation Page ___ )

Implementation Direction: *(if applicable)*

1. This Change Request (CR) is approved for CCB Baselining as the "Site Characterization Program Baseline, Revision 1," and is assigned Controlled Document number YMP/CM-0011.

2. The Director, Regulatory and Site Evaluation Division is responsible for ensuring the above listed conditions are incorporated into the next revision of Document YMP/CM-0011.

(See Change Documentation Continuation Page 2 )

## SECTION III. CONCURRENCE

7 Quality Assurance Organization Concurrence

Name: D. G. Horton *(print)* Org.: PQA *(print)*
Signature: N Valdura for DG Horton Date: 3/28/91

8 Disposition Authority

Name: M. B. Blanchard *(print)* Title: CCB Chrprsn *(print)*
Signature: Robert V [illegible] For Maxwell Blanchard Date: 3/28/91

9 Effective Date: 3/28/91

Implementation Direction (continued)

3. The CCB Secretary shall ensure that the Cover Page and the Title Page for Document YMP/CM-0011, Revision 1, are prepared.

4. The Document Originator shall provide a Print Ready Copy of YMP/CM-0011, Revision 1, to the CCB Secretary. The Document Number and Revision Number will be identified on each page of the Publication Ready Document, YMP/CM-0011.

5. The CCB Secretary shall ensure that YMP/CM-0011, Revision 1, is prepared in accordance with this Change Directive (CD). The CCB Secretary shall ensure the Document Change Notice (DCN), indicating changes made in the document, is prepared. The DCN will be attached to the front of the Print Ready Copy of the document. The CCB Secretary shall also prepare a Controlled Document Issuance Authorization (CDIA) to transmit this CD, the DCN, and YMP/CM-0011, Revision 1, to the Project Document Control Center (DCC) in accordance with AP-1.5Q.

6. Per AP-3.3Q, each TPO and Project Office Division Director will complete an Affected Document Notice (ADN) as notification of completion of implementation planning for this CD.

7. The CCB Secretary shall ensure that the Configuration Information System (CIS) and the CCB Register are updated to reflect Revision 1 to YMP/CM-0011.

8. Any changes to document YMP/CM-0011, Revision 1, will require submittal of a CR to the Project CCB.

9. Upon release of YMP/CM-0011, Revision 1, all Project Participants will be required to use YMP/CM-0011, Revision 1, in performing duties applicable to this document.

Y-AD-059
9/90

# YUCCA MOUNTAIN PROJECT
# DOCUMENT CHANGE NOTICE (DCN) RECORD



1 Document Title:

Site Characterization Program Baseline

2 Document Number:
YMP/CM-0011

The document identified in Blocks 1 and 2 has been changed. The changed pages attached to this DCN are identified in Block 7 opposite the latest DCN number in Block 3. The original issue of this document as modified by all applicable DCN's constitutes the current version of the document identified in Blocks 1 and 2.

| 3 DCN NO. | 4 CR NO. | 5 DOCUMENT Rev./ICN # | 6 CR TITLE | 7 AFFECTED PAGES | CHANGE | ADD | DELETE | 8 DATE |
|---|---|---|---|---|---|---|---|---|
| 001 | 91/052 | Rev. 1 | Submit SCPB, Rev. 1 for CCB Control (complete revision of information related to ESF design) | All | X | | | 4/5/91 |

Department of Energy
Yucca Mountain Site Characterization
Project Office
P. O. Box 98608
Las Vegas, NV 89193-8608

WBS 1.2.9
QA: N/A

MAR 20 1991

Distribution

RENAMING OF EXPLORATORY SHAFT EFFORT

As a consequence of the instructions from Dr. John W. Bartlett, Director of the Office of Civilian Radioactive Waste Management, on February 12, 1991, about the redirection of Yucca Mountain Site Characterization Project efforts associated with the Exploratory Shaft Facility design effort, it has become apparent that retaining the name of Exploratory Shaft would be somewhat misleading when the current design studies are focusing upon ramps, and a shaft is only being considered as a possible backup.

Therefore, after considerable discussion with many parties about selecting a new name, I have concluded that the most appropriate approach for now is to change the name of Exploratory Shaft Facility (ESF) to Exploratory Studies Facility (ESF). As you can observe, the acronym remains the same but "Shaft" becomes "Studies."

For all future communication, I request that you use this new name for this very important facility. We do not plan on modifying any completed documents or sending out errata sheets. I do request that all new communications within the U.S. Department of Energy's program now refer to this facility as the Exploratory Studies Facility. I thank you for your cooperation.

Carl P. Gertz
Project Manager

YMP:MBB-2814

YMP-5

MAR 20 1991

Distribution—Memorandum dated

John W. Bartlett, HQ (RW-1) FORS
Franklin G. Peters, HQ (RW-2) FORS
Robert Clark, HQ (RW-3) FORS
Thomas H. Isaacs, HQ (RW-4) FORS
Jerome D. Saltzman, HQ (RW-5) FORS
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Vincent F. Iorii, YMP, NV
A. C. Robison, YMP, NV
Robert V. Barton, YMP, NV

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The numbering scheme used in this table of contents reflects that the numbering of the Site Characterization Plan has been preserved to maintain consistency among related documents. Sections 8.5 and 8.6 have been intentionally excluded.

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## 8.3.5.2 Issue resolution strategy for Issue 2.4: Can the repository be designed, constructed, operated, closed, and decommissioned so that the option of waste retrieval will be preserved as required by 10 CFR 60.111?

This issue is concerned with the ability to retrieve emplaced waste as required by 10 CFR 60.111(b). As a result of this requirement, the repository must be designed, constructed, operated, and maintained to ensure that the emplaced waste can be retrieved. This leads to impacts on the design of the repository and upon the duration of many of the planned operations. As will be discussed in more detail later, numerous design decisions are based to a large degree on retrieval-related considerations; for example, the selection of the waste emplacement mode, the selection of materials for rock-support systems, and the maintenance requirements for the ramps, shafts, and drifts. Furthermore, the operations directly related to waste retrieval must be recognized as more complex than the emplacement operations, primarily because of the more difficult environment related to retrieval (e.g., increased heat).

There are three points that should be identified relative to the discussion of retrieval presented here. First of all, this issue (Issue 2.4, waste retrievability) is a performance issue. The importance of this issue and the numerous design constraints created to ensure retrievability lead to strong ties to the principal design issue (Issue 4.4, preclosure design and technical feasibility, Section 8.3.2.5). Issue 4.4 is responsible for the reference repository design, supporting analyses, and demonstrations required by this and other design or performance issues. This relationship between design and performance issues is shown in Figure 8.3.2.1-1 (Section 8.3.2.1). Because the performance goals for retrieval are integrated in Issue 4.4 with other related goals, the site data needed to implement and evaluate the goals are identified in the discussions under Issue 4.4. Secondly, the act of retrieval is considered complete in these discussions when the waste is brought to the surface. Temporary waste storage at the surface and offsite transport after retrieval are not addressed in the SCP because it is unlikely that these activities would require any site data that are not already being requested. Finally, the decision to retrieve will be made as a result of the performance confirmation program or by the DOE for recovery of resources. The discussions of retrieval are therefore limited to activities intended to maintain the retrieval option and to retrieve the waste.

In the discussion that follows in this section, the regulatory basis for addressing waste retrieval is presented, the approach to resolving this issue is described, and the interrelationships among the information needs related to retrievability are discussed.

### Regulatory basis for the issue

The regulations concerning the retrieval of high-level radioactive waste from geologic repositories are contained in the Nuclear Waste Policy Act (NWPA, 1983) and the NRC regulation 10 CFR Part 60. The DOE requirement for reasonably available technology is contained in 10 CFR Part 960.

The principal NWPA reference to retrieval is contained in Section 122 (NWPA, 1983):

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> Notwithstanding any other provision of this subtitle, any repository constructed on a site approved under this subtitle shall be designed and constructed to permit the retrieval of any spent nuclear fuel placed in such repository, during an appropriate period of operation of the facility, for any reason pertaining to the public health and safety, or the environment, or for the purpose of permitting the recovery of the economically valuable contents of such spent fuel. The Secretary shall specify the appropriate period of retrievability with respect to any repository at the time of design of such repository, and such aspect of such repository shall be subject to approval or disapproval by the Commission as part of the construction authorization process under subsections (b) through (d) of Section 114.

The principal NRC reference to retrievability is in Section 60.111(b) of 10 CFR Part 60.

> Retrievability of Waste. (1) The geologic repository operations area shall be designed to preserve the option of waste retrieval throughout the period during which wastes are being emplaced and, thereafter, until the completion of a performance confirmation program and Commission review of the information obtained from such a program. To satisfy this objective, the geologic repository operations area shall be designed so that any or all of the emplaced waste could be retrieved on a reasonable schedule starting at any time up to 50 yr after waste emplacement operations are initiated, unless a different time period is approved or specified by the Commission. This different time period may be established on a case-by-case basis consistent with the emplacement schedule and planned performance confirmation program. (2) This requirement shall not preclude decisions by the Commission to allow backfilling part or all, or permanent closure of, the geologic repository operations area prior to the end of the period of design for retrievability. (3) For purposes of this paragraph, a reasonable schedule for retrieval is one that would permit retrieval in about the same time as that devoted to construction of the geologic repository operations area and the emplacement of wastes.

In addition, minor references to retrieval and retrievability are included in 10 CFR Part 60, Sections 21(c)(12), 46(a)(1), 102(d), 133(c), 133(e), and 135(b)(3). These sections address the content requirements for the license application, design changes that affect retrievability, stages in the licensing process, design criteria for the surface and underground facilities, design criteria for underground openings, and design criteria for waste packages.

The DOE requirement for reasonably available technology is contained in 10 CFR 960.5-1(a)(3):

> Ease and cost of siting, construction, operation, and closure. Repository siting, construction, operation, and closure shall be demonstrated to be technically feasible on the basis of reasonably

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available technology, and the associated costs shall be demonstrated to be reasonable relative to other available and comparable options.

A retrieval requirement is presented in 40 CFR 191.14(f). However, in the introductory text to Part 191.14, the EPA authors clearly indicate that this particular section does "not apply to facilities regulated by the [Nuclear Regulatory] Commission. (See 10 CFR Part 60)."

In compliance with the regulations, the Yucca Mountain repository is being designed with the option to initiate retrieval of emplaced waste at any time up to 50 yr after waste emplacement operations are initiated and to use reasonably available technology for the retrieval operations.

Approach to resolving the issue

The basic approach to resolving Issue 2.4 (waste retrievability) is depicted in the logic diagram provided as Figures 8.3.5.2-1a and 1b. The essence of the logic for resolving retrievability concerns is to

1. Evaluate regulatory requirements and existing site data, designs and analyses to determine what functions and processes must be performed to not preclude retrieval.

2. Establish performance measures and goals (design criteria) for the processes that contribute to performing those functions.

3. Identify normal and credible abnormal conditions for retrieval-related operations and identify input items needed from Issue 4.4 (preclosure design and technical feasibility).

4. Identify and request site parameters necessary to meet the goals of related issues for common system elements or develop the reference preclosure repository design, operations plans, supporting analyses and demonstrations requested to support resolution of all related issues.

5. Conduct a compliance analysis to critically evaluate whether the appropriate retrieval conditions have been considered, whether the input items provided by Issue 4.4 are complete and sufficient, and whether the performance goals are met.

Steps 1 and 2 above represent the performance allocation process being used in the SCP to communicate the development of preliminary performance measures and associated goals and needed confidence for resolving the design and performance issues. The remainder of this section on the approach to resolving this issue documents the current preliminary results of the performance allocation process for retrieval. The future work associated with steps 3 to 5 is described in the retrieval information needs discussions (Sections 8.3.5.2.1 through 8.3.5.2.6 for this issue) or in the discussions of future work for Issue 4.4 (preclosure design and technical feasibility). These steps indicate an important relationship between the retrieval issue and Issue 4.4. Figure 8.3.5.2-2 shows what the waste retrievability issue

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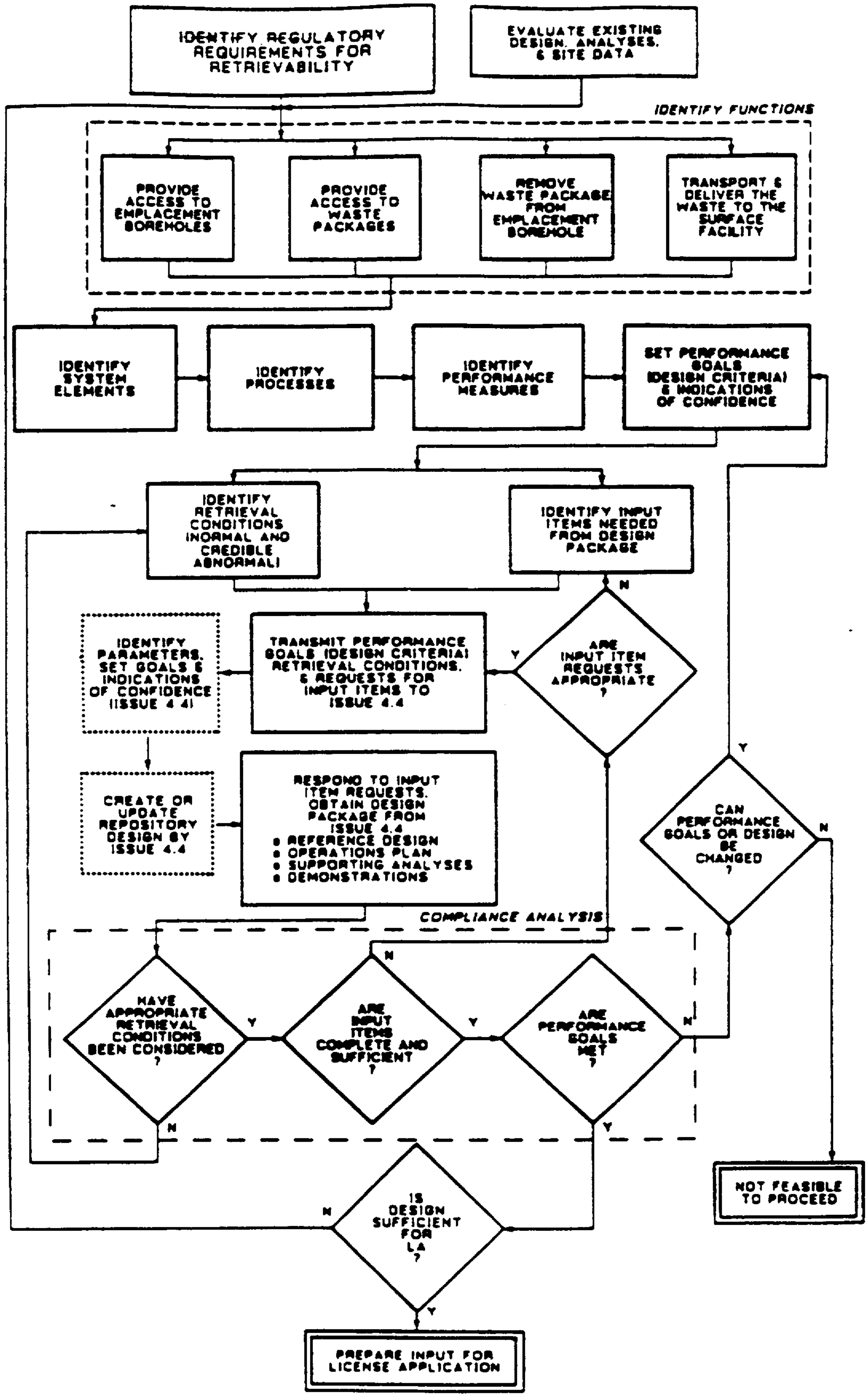


**Figure 8.3.5.2-1a.** Logic diagram for Issue 2.4 (waste retrievability). See Figure 8.3.5.2-1b for legend. Section 8.3.2.1 describes the relationships and interfaces between design and performance issues.

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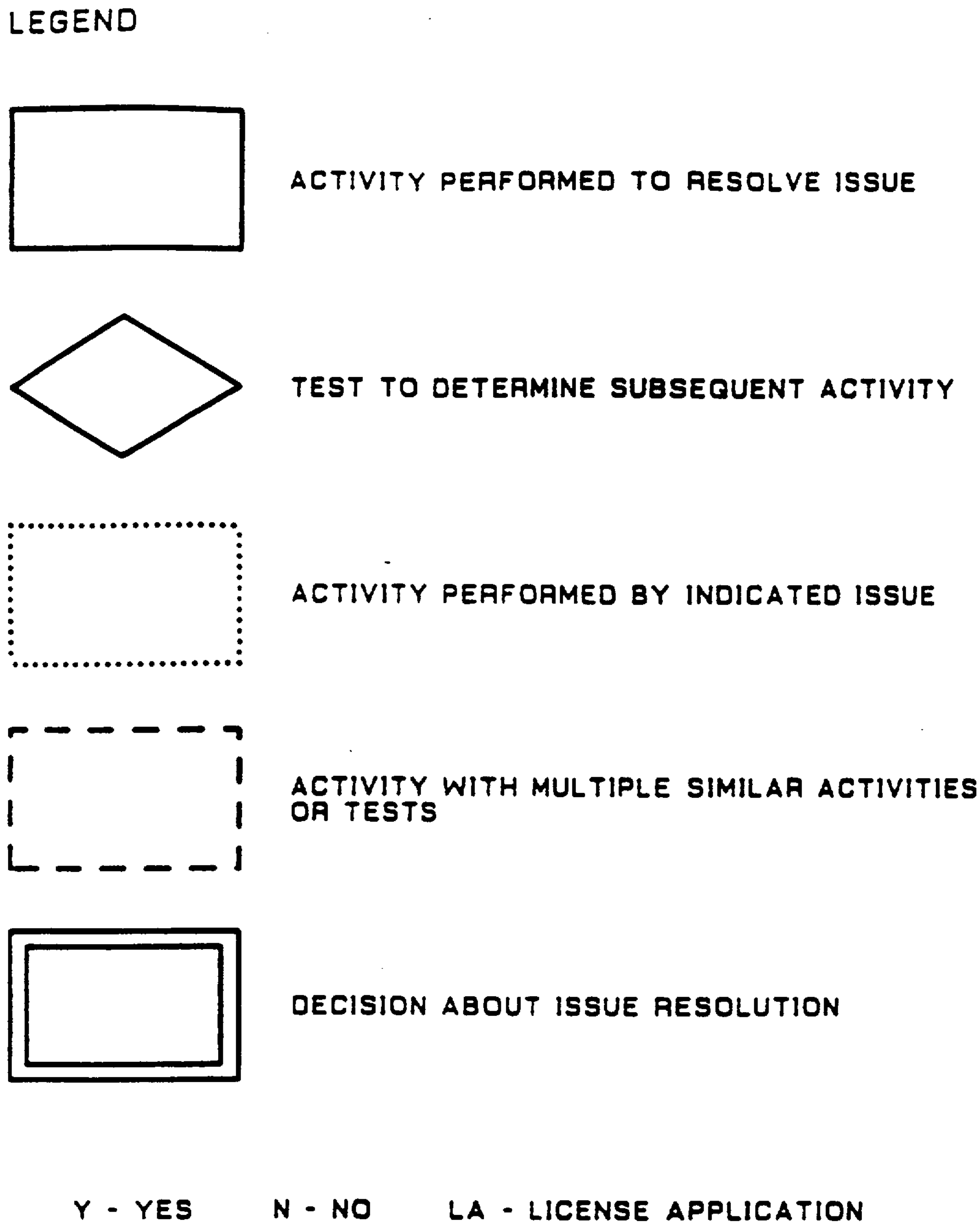


**Figure 8.3.5.2-1b.** Legend for Figure 8.3.5.2-1a.

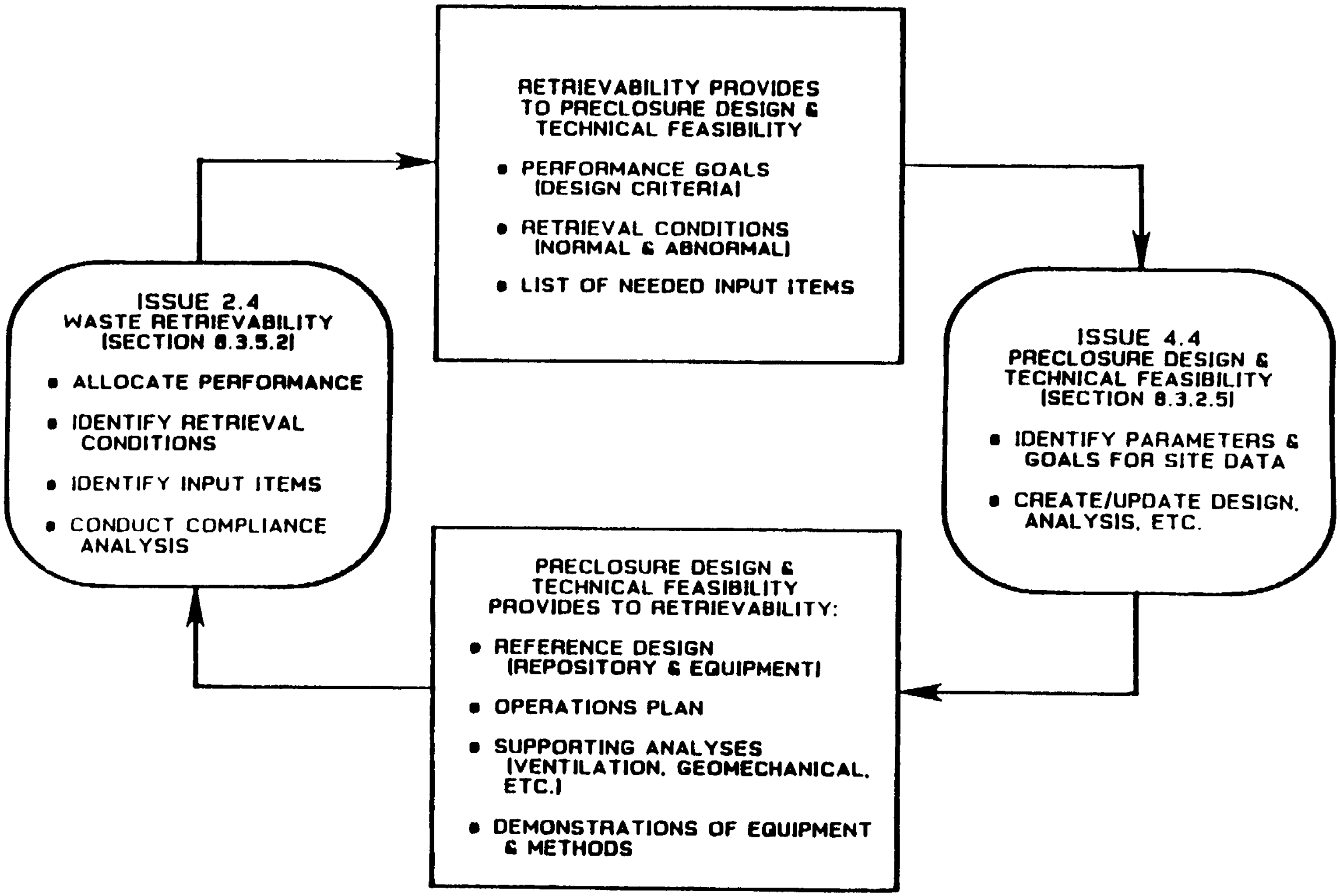


**Figure 8.3.5.2-2.** Information exchanged between Waste Retrievability Issue and Preclosure Design and Technical Feasibility Issue



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provides to the repository design issue, as well as what the design issue provides for use in the retrieval compliance analysis. The results or products produced by the repository design issue include reports that document the reference designs for the repository and equipment as well as reports that document operations plans, analyses, and equipment demonstrations. Not all of these products (for example, muck haulage analyses) are needed to evaluate retrieval-related concerns. The products developed in Issue 4.4 that are requested by the retrievability issue are called "input items" in the discussions on retrieval to distinguish them from products (retrieval conditions, compliance analyses, etc.) developed by the retrieval issue. Section 8.3.2.1 explains this relationship between input items and products in more detail.

One concept considered throughout the retrieval discussions is that of identifying both normal conditions and credible abnormal conditions that might be expected to exist during retrieval-related operations. Normal conditions are the state or conditions (temperature, air quality, opening stability, etc.) expected to be present most of the time. The term, normal conditions, is generally used to indicate conditions expected about 90 percent of the time. Standard equipment and procedures would be expected to be used for retrieval operations when normal conditions exist. Credible abnormal conditions are the state or conditions expected to have a reasonable potential for occurring infrequently during the life of a repository. This term is generally used to identify those conditions that need to be considered in developing contingency plans for related retrieval operations. Such operations may require special equipment or procedures and may require substantial time to complete.

The starting point for the performance allocation process for retrievability is consideration of the regulatory requirements (discussed earlier in this section) and an evaluation of the existing design, analyses, and site data. Retrieval-related concerns are woven throughout numerous sections of the current design discussions in Chapter 6 of the SCP and, likewise, in the Site Characterization Plan-Conceptual Design Report (SCP-CDR) (SNL, 1987). Rather than presenting the details of those discussions here, a directory of these discussions related to retrieval is provided in Table 8.3.5.2-1. From the directory, it is evident that the subject of retrievability has received consideration in numerous areas, particularly in the design requirements, the ventilation system evaluation, operations planning, analyses of both thermomechanical effects and liner stresses, and equipment design. Additionally, a specific evaluation (Appendix L of the SCP-CDR; SNL, 1987) was made to determine the relative importance of various items to maintaining the option to retrieve the waste in a timely manner; it is this evaluation that forms the basis for the preliminary list of potential abnormal conditions that might exist during retrieval.

Using the regulatory requirements and the current design and analyses, the functions that must be performed for retrieval have been identified. These four functions are

1. Provide access to the emplacement boreholes.
2. Provide access to the waste packages.
3. Remove waste package from the emplacement borehole.
4. Transport and deliver the waste packages to the surface facilities.

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Table 8.3.5.2-1. Directory of discussions related to retrieval

| Topic | SCP section | SCP-CDR[a] section |
|---|---|---|
| Waste retrieval schedule | 6.1.1.6.4 | 3.0 |
| Retrievability-related design criteria | 6.1.1.7 | 2.4.4.3 |
| Waste retrieval and shipping operations | 6.2.3.2 | 3.2.2. and Appendix J |
| Retrieval requirements and planning-basis time periods | 6.2.9.1. | 2.4.4.1 |
| Retrieval conditions | 6.2.9.2 | 6.3.1 and Appendix J |
| Equipment development | 6.2.9.3 | Appendix J |
| Issue 2.4 waste retrievability (current status) | 6.4.8 | 8.3.5 |
| Issue 2.4 waste retrievability (issue resolution strategy and future work) | 8.3.5. | NA[b] |
| Retrieval philosophy | NA | 2.4.4.2 and 3.2.1 |
| Drift ventilation conditions for maintenance and retrieval | NA | 3.4.2.2 |
| Waste removal operations for performance confirmation | NA | 4.5.4 |
| Retrieval demonstrations | NA | 6.3.2 |
| Full repository retrieval | NA | 6.3.3 |
| Expected temperature for borehole walls and drifts after spent fuel emplacement | NA | Appendix A |
| Air-cooling requirements--vertical | NA | 3.4.2.3 |
| Air-cooling requirements--horizontal | NA | 3.4.3.3 |
| Preliminary liner stress analysis | NA | Appendix B |
| Ventilation and cooling analyses | NA | Appendix C |
| Equipment for retrieval | NA | Appendix D |

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Table 8.3.5.2-1. Directory of discussions related to retrieval (continued)

| Topic | SCP section | SCP-CDR[a] section |
|---|---|---|
| An assessment of the feasibility of disposing of nuclear waste in the horizontal configuration | NA | Appendix E |
| Waste retrieval | NA | Appendix J |
| Items important to retrievability at the Yucca Mountain Repository | NA | Appendix L-2 |
| Thermomechanical analyses | NA | Appendix N |

[a]SCP-CDR = Site Characterization Plan-Conceptual Design Report (SNL, 1987).
[b]NA = not applicable. Topic discussed only in SCP-CDR or in SCP.

These four functions provide the organizing framework upon which the retrieval discussions, information needs, and plans for future evaluations are based. Specific information that was used in identifying the functions are the requirements documents (DOE, 1986b; Appendix P of SNL, 1987), operations reports (Dennis et al., 1984a,b; Stinebaugh and Frostenson, 1986; Stinebaugh et al., 1986), the Project report on a strategy for retrieval-related compliance demonstrations (Flores, 1986), and the applicable portions of 10 CFR Part 60 and 10 CFR Part 960.

For each of the four functions, the system elements and processes that relate to performing the functions were identified. The system elements involved in the performance of the general functions were identified by (1) reviewing the requirements contained in the system requirements (SR) and the subsystems design requirements (SDR) (SNL, 1987, Appendix P) and (2) analyzing the defined systems definitions with respect to the general function to be performed. A figure containing the system elements defined for the Yucca Mountain Project is presented in Section 8.2.1. The processes were identified using the previously mentioned operations reports and the Project report on retrieval-related compliance demonstration (Flores, 1986).

Next, the performance measures for each of the processes were established. These measures were developed using reference design information and engineering judgment. Performance goals and levels of confidence were defined for each of the performance measures. In instances where the goal is quantifiable, specific values are presented. For performance measures that do not require site data, specific goals are not presented in the SCP. These goals will be presented in the repository design plan. In

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many instances the goals are based on specific details and assumptions in the current design. The goals may change as the design and design assumptions are refined. The paragraphs and tables that follow will document the performance allocation process for each of the four functions.

Function 1: Provide access to the emplacement boreholes

To provide a safe and reliable access from the surface facilities to the emplacement boreholes, the underground openings must be usable and the environment within them must be acceptable under normal and credible abnormal conditions. The processes, performance measures, and performance goals (design criteria) involved in providing this ability are presented in Table 8.3.5.2-2. The output of this performance allocation process, shown in the table, are performance goals (design criteria).

With respect to access and drift usability, the performance goal is usability for a time period of at least 84 yr. As shown in Figure 8.3.5.2-3, this time period is generated by adding the design-basis period of retrievability (50 yr) and the actual retrieval period of 34 yr (Flores, 1986). For purposes of design, the actual retrieval period is assumed to be the time for construction of the repository (6 yr) and the emplacement of waste (28 yr), a total of 34 yr. This time period is a significant and potentially severe restriction that will impact the design, construction, and operation of the repository. For example, the materials selected for the rock support system, the necessity for a continual, long-term monitoring and maintenance program for the underground openings, the timing for backfilling operations, and the selection of an acceptable emplacement mode (vertical, short horizontal, or long horizontal boreholes) are all significantly impacted by the 84-yr duration of potential activities (Figure 8.3.5.2-3).

Subsection (2) of 10 CFR 60.111(b) allows for the use of backfill before the end of the retrievability period. Since the access and drifts will be designed to be usable throughout the retrievability period, the option to backfill will be maintained through decommissioning. The Yucca Mountain Project design basis does not include the use of backfill during the period of retrievability; hence no performance goals relative to retrieval are established for backfilling operations. Descriptions of the postclosure-related goals for backfill are provided in discussions related to sealing (Section 8.3.3.2) and to the postclosure design of the repository (Section 8.3.2.2).

To ensure that the environment in the nonoperational areas (areas that were closed off after waste was emplaced) would not be so severe as to cause reentry to be impractical, the following goals were established for the nonoperational areas:

1. For vertical emplacement, the access drift wall temperatures will not exceed 50°C for 50 yr after waste emplacement is initiated.

2. For horizontal emplacement, the emplacement drift wall temperatures will not exceed 50°C for 50 yr after waste emplacement is initiated.

These goals are referred to as the 50/50 goals. The 50°C limit was selected such that it would not be impractical to modify the environment

Table 8.3.5.2-2. Performance measures, goals, and needed confidence for processes or activities involved in providing access to the emplacement borehole for retrieval (retrieval function 1) (page 1 of 2)

| Process or activity | Performance measures | Tentative goals[a] | Needed confidence |
|---|---|---|---|
| Design and construct the accesses and drifts to be usable throughout the retrievability period for normal and credible abnormal conditions | Time during which the drifts and accesses will remain usable | Time ≥84 yr | High |
| Develop rock support concepts that ensure maintainability | Amount of spall | Spall averages less than 3 tons per 1,000 ft of drift per year | High |
| | Opening displacement | Opening displacement <6 in. | High |
| | Frequency of maintenance | Frequency of needed maintenance in underground openings >5 yr average | Low |
| Develop backfill removal concepts (if needed) | Time and level of effort for backfill removal | None--the current design basis allows for backfilling during repository closure (i.e, after the period of retrievability) | NA[b] |
| Monitor drifts and accesses to determine maintenance needs | Localized rock and rock support displacement | Monitor displacements >1 in. | High |



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Table 8.3.5.2-2. Performance measures, goals, and needed confidence for processes or activities involved in providing access to the emplacement borehole for retrieval (retrieval function 1) (page 2 of 2)

| Process or activity | Performance measures | Tentative goals[a] | Needed confidence |
|---|---|---|---|
| Design for a specific temperature and air quality environment within the accesses and drifts | Drift temperature | Temperature less than 50°C (for 50 yr - emplacement drift (H)[c] or access drift (V)[c]) | Low |
| | Air quality | Air quality standards met (work areas) | High |
| Verify environment for maintenance and retrieval operations | Air quality | Air quality measurements adequate for retrieval operations to meet standards | High |
| Modify environment (as necessary) | Time required to modify the environment for retrieval | Air quality standards met within 8 weeks (unprotected) | Medium |

[a]These goals are integrated with goals from other issues in the discussion of Issue 4.4, preclosure design and technical feasibility (Section 8.3.2.5, Tables 8.3.2.5-1 through 12). Site characterization related design or performance parameters, their goals, and their confidences are also established in the Issue 4.4 discussions.

[b]NA = not applicable for SCP.

[c]H = horizontal emplacement; V = vertical emplacement.



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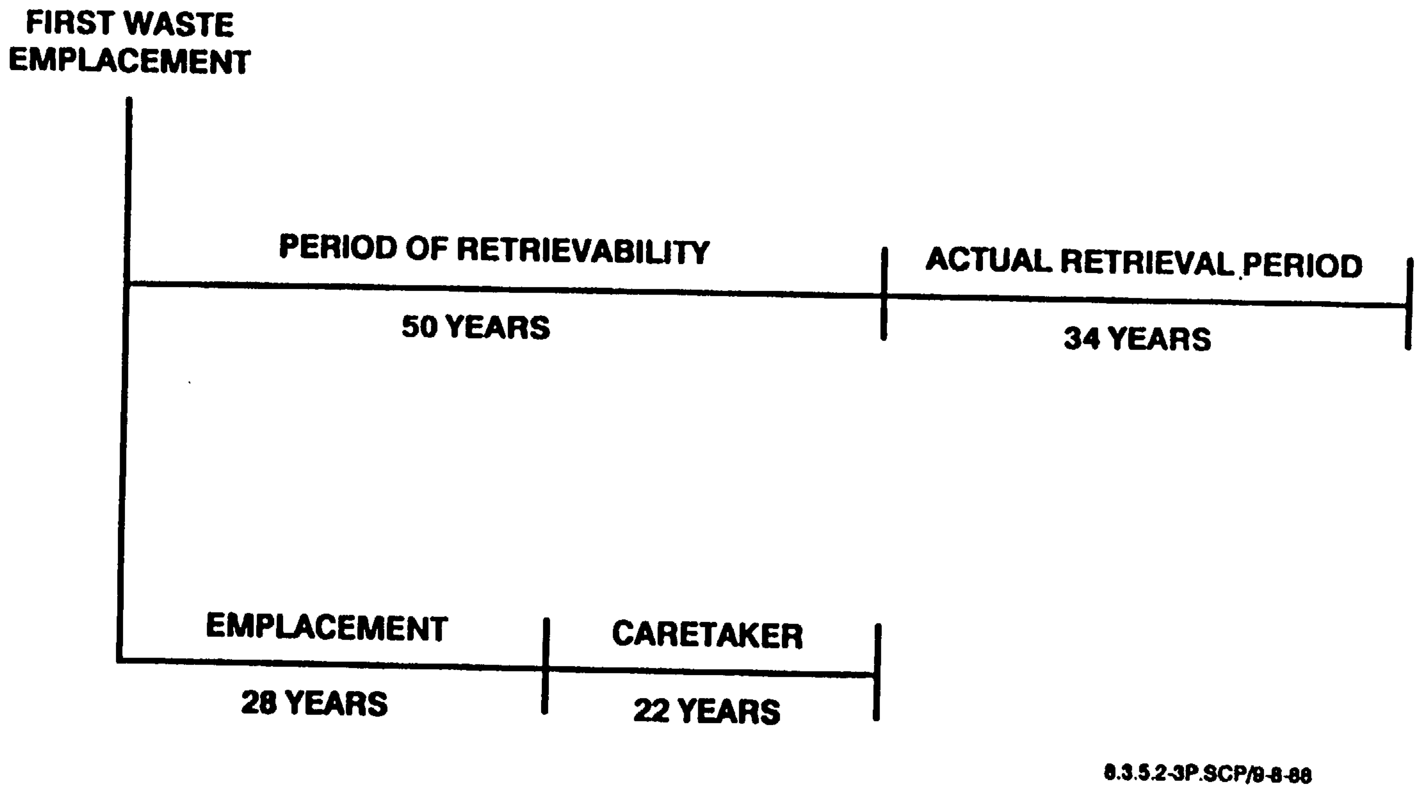


**Figure 8.3.5.2-3.** Retrieval time frame for design purposes.



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within the drifts for unprotected workers during the 50-yr period of retrievability. In addition, protected workers could reenter for inspection purposes with minimal need for environment modification.

For the working areas, the ventilation system must be capable of maintaining the environment within specified limits on a continuous basis throughout the period of retrievability and the actual retrieval period. For nonoperational areas (areas that were closed off after waste was emplaced), the goal is for the system to be capable of providing a safe environment within a reasonable period of time selected to be eight weeks after initiation of "cooldown" activities.

Function 2: Provide access to the waste packages

To provide a safe and reliable access from the emplacement drifts to the waste packages, the waste emplacement envelope (borehole, liner, shield plug, and shielding collar) must be designed to allow for removal of emplaced waste under normal and credible abnormal conditions. The processes, performance measures, and performance goals (design criteria) involved in providing this ability are presented in Table 8.3.5.2-3.

The primary concern with respect to waste package access is to ensure that the waste package does not become lodged inside the emplacement borehole. As a result, the tentative goals for liner displacement and radius of curvature were established and tentative goals for borehole rockfall and displacement were identified. For vertical emplacement, the performance goal for maximum deflection of the liner or borehole is 2 in. (5 cm) assuming a partially lined hole. For horizontal emplacement, the goal is for a maximum liner deflection of 3 in. (7.6 cm). The larger allowable deflection for horizontal emplacement is a result of the larger diameter (hence, more clearance) of the emplacement boreholes. To ensure that the waste package does not bind against the liner for horizontal emplacement, the radius of curvature for the borehole and liner should be 110 ft (33.5 m) or greater. For both emplacement methods, the liner lifetime will be 84 yr or greater. The rationale for the 84-yr period is provided under Function 1.

The ability to perform this function will be one of the significant concerns in selecting the preferable emplacement mode. Particularly important will be a thorough and critical evaluation of the potential for excessive liner deflection in horizontal boreholes as a result of rockfall, seismic effects or excessive temperatures. If such abnormal conditions were found to be credible, relatively complicated retrieval operations would be required.

Function 3: Remove waste package from the emplacement borehole

To ensure that the waste package can be removed from the emplacement boreholes, the transporter and the waste package are being designed to allow for removal of the emplaced waste package under normal and credible abnormal conditions. In the horizontal case, a dolly also is included in the current design concepts. The processes, performance measures, and performance goals involved in providing this ability are presented in Table 8.3.5.2-4.

Table 8.3.5.2-3. Performance measures, goals, and needed confidence for processes or activities involved in providing access to the waste packages for retrieval (retrieval function 2) (page 1 of 2)

| Process or activity | Performance measures | Tentative goals[a] | Needed confidence |
|---|---|---|---|
| Design waste emplacement envelope to allow access to the waste package throughout the retrievability period for normal and credible abnormal conditions | Borehole usability | | |
| | Rockfall | Average rockfall <250 lb per foot of borehole | Medium |
| | Displacement of borehole wall | Rock displacement <2 in. | Medium |
| | Borehole liner lifetime | Liner lifetime ≥84 yr | High |
| | Borehole liner displacement | Liner displacement <2 in. (V)[b] | High |
| | | Liner displacement <3 in. (H)[b] | High |
| | Borehole liner curvature radius | Liner curvature radius >110 ft (H) | Medium |
| Assess the condition of the emplacement envelope and waste package prior to removal (as required) | Borehole liner displacement | Detect displacement >0.5 in. | Medium |




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Table 8.3.5.2-3. Performance measures, goals, and needed confidence for processes or activities involved in providing access to the waste packages for retrieval (retrieval function 2) (page 2 of 2)

| Process or activity | Performance measures | Tentative goals[a] | Needed confidence |
|---|---|---|---|
| Perform corrective actions (as required) | Time required to perform corrective actions | Average time <1 month per drift (normal conditions) | Medium |
| | | Timely manner considering site-specific credible abnormal conditions. For planning purposes, time <1 yr is assumed for each event. | Medium |

[a]These goals are integrated with goals from other issues in the discussion of Issue 4.4, preclosure design and technical feasibility (Section 8.3.2.5, Tables 8.3.2.5-1 through 12). Site characterization related design or performance parameters, their goals, and their confidences are also established in the Issue 4.4 discussions.

[b]V = vertical emplacement; H = horizontal emplacement.



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Of primary concern is the ability of the host rock and shielding collar to provide an acceptable level of shielding during waste removal. Consequently, the performance goal is to provide shielding such that radiation dose levels to the workers do not exceed the design limits that are established in Issue 2.7 (repository radiological design criteria (preclosure), Section 8.3.2.3). Shielding analyses and requirements for site data are addressed in Issue 4.4 (preclosure design and technical feasibility, Section 8.3.2.5). The second performance goal addresses the time allowed for removal of a waste package from an emplacement borehole. For purposes of initial design evaluations, the time allowed for the removal of a waste package (under normal conditions) has been selected to be less than twice the amount of time that was allowed for the emplacement of a waste package. The rest of the performance measures for function 3 do not involve site data not already being requested. As a result, the corresponding performance goals will be addressed in the repository design plan to be published prior to the advanced conceptual design.

The ability to perform this function for credible abnormal events could be among the most difficult repository operations. The operations are complicated by the high-temperature, radioactive environment, the need to handle containers some of which may have been emplaced for more than 50 yr, and the uncertainties regarding the condition of the boreholes and waste containers. Hence, to think of these operations as the reverse of emplacement would be an understatement of the potential operational difficulties. Selected operations to perform this function will probably require proof-of-principle demonstrations in accordance with DOE policy (DOE, 1986c). In-depth plans will be developed for these equipment demonstrations, however designs and further identification and evaluation of related credible abnormal conditions will be developed before demonstration tests can be planned in detail. Nevertheless, Section 8.3.5.2.4 describes the current list of equipment components that might need to be demonstrated.

Function 4: Transport and deliver the waste packages to the surface facilities

The transporter must be developed to allow for transport of the waste packages to the surface and unloading at the surface. The surface waste handling building must be designed and constructed to allow for unloading of waste. Transport and unloading must be performed under normal and credible abnormal conditions. As discussed in the introduction to this section, the surface storage of retrieved waste and offsite transport are not included in the retrieval discussions. The processes, performance measures, and performance goals for function 4 are presented in Table 8.3.5.2-5. The requirements for access and drift usability and for an acceptable environment are included under function 1, access to the boreholes (Table 8.3.5.2-2).

Hence, for function 4, it is assumed that the accesses and drifts are usable and that an acceptable environment exists, even if substantial maintenance had to be performed.

Numerous analyses of the performance and design of the transporter will be needed to evaluate its ability to safely and reliably transport the waste. Evaluations of accident conditions, reliability, and efficiency will be made.

Table 8.3.5.2-4. Performance measures, goals, and needed confidence for processes or activities involved in removing waste packages from emplacement boreholes (retrieval function 3) (page 1 of 2)

| Process or activity | Performance measures | Tentative goals[a] | Needed confidence |
|---|---|---|---|
| Design the waste package and transporter with the option to remove the waste for normal and credible abnormal conditions | Radiation protection | Worker dose less than allowable dose (see Issue 2.7 for specific goals and needed parameters) | High |
| | Time required to perform waste removal | Average time for removal less than twice the time for emplacement | Medium |
| | Removal latch and pull strength | These performance measures do not require site data and will be addressed in the repository design plan | NA[b] |
| | Structural strength of the waste package or dolly | These performance measures do not require site data and will be addressed in the repository design plan | NA |
| Verify conditions of equipment and waste package | Waste package structural failure detection | These performance measures do not require site data and will be addressed in the repository design plan | NA |
| | Removal equipment performance | These performance measures do not require site data and will be addressed in the repository design plan | NA |



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Table 8.3.5.2-4. Performance measures, goals, and needed confidence for processes or activities involved in removing waste packages from emplacement boreholes (retrieval function 3) (page 2 of 2)

| Process or activity | Performance measures | Tentative goals[a] | Needed confidence |
|---|---|---|---|
| Verify operator training | Operator competency certification | These performance measures do not require site data and will be addressed in the repository design plan | NA |

[a]These goals are integrated with goals from other issues in the discussion of Issue 4.4, preclosure design and technical feasibility (Section 8.3.2.5, Tables 8.3.2.5-1 through 12). Site characterization related design or performance parameters, their goals, and their confidences are also established in the Issue 4.4 discussions.

[b]NA - not applicable for SCP.



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Interrelationships of information needs

The content of Tables 8.3.5.2-2 through 8.3.5.2-5 and the accompanying text cover performance allocation steps in the issue resolution strategy presented in Figure 8.3.5.2-1. The balance of the steps in the issue resolution strategy will be discussed in terms of the following information needs.

| Information need | Subject |
|---|---|
| 2.4.1 | Site and design data required to support retrieval (Section 8.3.5.2.1) |
| 2.4.2 | Determination that access to the waste emplacement boreholes can be provided throughout the period of retrievability and the actual retrieval period for normal and credible abnormal conditions (Section 8.3.5.2.2) |
| 2.4.3 | Determination that access to the waste packages can be provided throughout the period of retrievability and the actual retrieval period for normal and credible abnormal conditions (Section 8.3.5.2.3) |
| 2.4.4 | Determination that the waste can be removed from the emplacement boreholes for normal and credible abnormal conditions (Section 8.3.5.2.4) |
| 2.4.5 | Determination that the waste can be transported to the surface and delivered to the waste-handling surface facilities for normal and credible abnormal conditions (Section 8.3.5.2.5) |
| 2.4.6 | Determination that the retrieval requirements set forth in 10 CFR 60.111(b) are met using reasonably available technology (Section 8.3.5.2.6) |

There is a direct relationship between the logic shown in Figure 8.3.5.2-1 for the resolution of the waste retrievability issue and its information needs because the information needs were derived from the work that must be performed to ensure that the requirements for retrievability are met. The information needs can be categorized as follows:

1. The first information need is a summary of the information that will be communicated to Issue 4.4. This communication is shown in Figure 8.3.5.2-1 in the box labeled "transmit performance goals (design criteria), retrieval conditions, and requests for input items to Issue 4.4."

Table 8.3.5.2-5. Performance measures, goals, and needed confidence for processes or activities involved in transporting and delivering the waste to the surface facilities (retrieval function 4)[a] (page 1 of 2)

| Process or activity | Performance measures | Tentative goals[b] | Needed confidence |
|---|---|---|---|
| Design the transporter with the ability to transport the waste to the surface for normal and credible abnormal conditions | Transporter design characteristics (braking ability, maximum speed, cornering ability, radiation protection) | Transporter must be able to operate with anticipated rockfall in accesses and drifts | High |
| | Time required to transport the waste to the surface | These performance measures do not require site data and will be discussed in the repository design plan | NA[c] |
| Design the surface waste-handling building and the transporter with the ability to unload waste at the surface facilities for normal and credible abnormal conditions | Time required to unload waste | These performance measures do not require site data and will be discussed in the repository design plan | NA |
| | Radiation protection | These performance measures do not require site data and will be discussed in the repository design plan | NA |
| | Transporter unloading capability | These performance measures do not require site data and will be discussed in the repository design plan | NA |
| Assess the ability to transport the waste to the surface facilities | Transporter drive system performance | These performance measures do not require site data and will be discussed in the repository design plan | NA |





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Table 8.3.5.2-5. Performance measures, goals, and needed confidence for processes or activities involved in transporting and delivering the waste to the surface facilities (retrieval function 4)[a] (page 2 of 2)

| Process or activity | Performance measures | Tentative goals[b] | Needed confidence |
|---|---|---|---|
| Assess the ability to transport the waste to the surface facility (continued) | Operator competency certification | These performance measures do not require site data and will be discussed in the repository design plan | NA |
| Assess the ability to unload the waste at the waste-handling building | Transporter unloading system | These performance measures do not require site data and will be discussed in the repository design plan | NA |
| | Surface facility unloading system performance | These performance measures do not require site data and will be discussed in the repository design plan | NA |
| | Operator competency certification waste to the surface | These performance measures do not require site data and will be discussed in the repository design plan | NA |

[a]Requirements for access and drift usability for transporter operation are included under function 1 (Table 8.3.5.2-1).

[b]These goals are integrated with goals from other issues in the discussion of Issue 4.4, preclosure design and technical feasibility (Section 8.3.2.5, Tables 8.3.2.5-1 through 12). Site characterization related design or performance parameters, their goals, and their confidences are also established in the Issue 4.4 discussions.

[c]NA = not applicable for SCP.





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2. The next four information needs correspond directly to the four retrieval functions and address what needs to be done to ensure that the option to retrieve is maintained. These information needs are responsible for the development of performance goals (design criteria), retrieval conditions, and requests for input items as shown in Figure 8.3.5.2-1.

3. The last information need ties the other information needs together and addresses the global requirements for retrieval to be completed on a "reasonable schedule" and for the use of "reasonably available technology." This work involves performing the compliance analysis indicated in Figure 8.3.5.2-1.

As shown in Figure 8.3.5.2-1, the logic for resolution of this issue involves an iterative process. As the repository and equipment designs are refined, work will be performed under this issue in the following areas:

1. The performance goals (design criteria), retrieval conditions, and input item requirements will be refined.

2. The strategy and planning documents will be refined.

3. Compliance analyses will be performed to verify that the design meets all of the requirements for retrievability.

### 8.3.5.2.1 Information Need 2.4.1: Site and design data required to support retrieval

Technical basis for addressing the information need

Issue 2.4 requires that compliance with the retrievability requirements be demonstrated using reasonably available technology. Information Need 2.4.1 requires that site and design data (input items) needed by this issue be identified. This identification is necessary to ensure the proper data are acquired during site characterization and to ensure all required design products developed by Issue 4.4 are provided to this issue. In addition, the design criteria (performance goals) and retrieval conditions established under this issue are communicated to Issue 4.4, preclosure design and technical feasibility, to ensure sufficient consideration for retrieval in the design process.

Link to the technical data chapters and applicable support documents

Chapter 6 presents the current design, and the status of this issue is summarized in Section 6.4.8. Retrieval-related performance goals (design criteria) that were considered in the development of the current design are presented in Section 6.1.1.7. The status on the development of retrieval conditions is presented in Section 6.4.8.2.2.

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Parameters

Because the retrieval-related design, support analyses and equipment tests and demonstrations are performed under Issue 4.4, site data needed to support these analyses and tests are specified by Issue 4.4. Requirements for products from Issue 4.4 are presented in the form of requests for input items. The current list of input items requested by this issue is shown in Table 8.3.5.2-6. More detailed information relative to the content of the input items is provided in later sections that discuss Information Needs 2.4.2 through 2.4.6 (Sections 8.3.5.2.2 through 8.3.5.2.6, respectively). The performance goals (design criteria) and retrieval conditions are presented in Tables 8.3.5.2-7 and 8.3.5.2-8, respectively. Generation of the actual performance goals was discussed in step 2 of the performance allocation process presented in the approach to resolving the issue section for this issue. Any refinement or updating of these performance goals will be addressed in design requirements documents in support of each phase of the repository design and will be reported in SCP progress reports.

As part of the resolution of this waste retrievability issue, Information Need 2.4.1 identifies the input items (products of Issue 4.4, preclosure design and feasibility) that are needed to evaluate whether performance goals of this issue are met and, in turn, to ensure compliance with the retrievability requirements. In addition, Information Need 2.4.1 facilitates the communication between this issue and Issue 4.4 (see Fig- ure 8.3.5.2-2) by transmitting the performance goals (design criteria) and retrieval conditions, generated by Issue 2.4, to Issue 4.4, and by requesting the input items from Issue 4.4. (Information Need 2.4.1 also receives the design products from Issue 4.4 and distributes them, as input items, to Information Needs 2.4.2 through 2.4.6 for use in performing the compliance analysis.)

Table 8.3.5.2-6. Retrieval-related input items (to be provided by Issue 4.4)

| Information need | Input item |
|---|---|
| 2.4.2 | Drift and access design and supporting evidence |
| - | Rock support system design and supporting analyses |
| 2.4.2 | Monitoring system (rock movement) and support analyses |
| | Drift and access maintenance program concepts and supporting evidence |
| | Ventilation system design and supporting analyses (for retrieval operations) |

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Table 8.3.5.2-6. Retrieval-related input items (to be provided by Issue 4.4) (continued)

| Information need | Input item |
|---|---|
| | Basis for ensuring air quality in operational areas and evaluating air quality in nonoperational areas |
| 2.4.3 | Waste emplacement envelope design and supporting analyses |
| | Waste emplacement envelope assessment |
| | Corrective actions (waste emplacement envelope) |
| 2.4.4 | Waste package removal system design and supporting analyses |
| | Concepts for borehole preparation for waste removal and supporting evidence |
| | Demonstrations of borehole preparation for waste removal and supporting evidence |
| 2.4.5 | Transporter design concepts and supporting analyses |
| | Unloading equipment design (surface facility) and supporting analyses |
| | Demonstrations for waste transport |
| | Demonstrations for waste unloading at the surface |
| 2.4.6 | Reference operations plans |
| | Basis for establishing the use of reasonably available technology for retrieval-related equipment |
| | Reference design and supporting analyses |

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Table 8.3.5.2-7. Retrieval-related design or performance goals (design criteria)

| Information need | Design or performance goal |
|---|---|
| 2.4.2 | The access and drifts will remain usable for at least 84 yr |
| | The average amount of spall in the drifts will be less than 3 tons per 1,000 ft of drift per year |
| | The rock displacement in the drifts will be less than 6 in. |
| | The monitoring system will detect rock displacements within the drifts that exceed 1 in. |
| | The frequency of maintenance within the underground openings will be greater than 5 yr |
| | For the vertical emplacement concept, the temperature within the access drifts will not exceed 50°C for 50 yr after waste emplacement |
| | For the horizontal emplacement concept, the temperature within the emplacement drifts will not exceed 50°C for 50 yr after waste emplacement |
| | For operational areas, all applicable air quality standards will be met |
| | The time required to modify the environment within closed drifts for unprotected workers will not exceed 8 wk |
| 2.4.3 | Rockfall within the emplacement boreholes will average less than 250 lb per foot of borehole |
| | Displacement of the borehole wall will be less than 2 in. |
| | The liner lifetime will be at least 84 yr |
| | The maximum liner deflection is 2 in. (5 cm) for the vertical emplacement concept and 3 in. (7.6 cm) for the horizontal concept |
| | For the horizontal emplacement concept, the minimum radius of curvature for the liner is 110 ft (33.5 m) |

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Table 8.3.5.2-7. Retrieval-related design or performance goals (design criteria) (continued)

| Information need | Design or performance goal |
|---|---|
| 2.4.4 | The time required per container for waste removal will not exceed twice the amount of time required for emplacement of a waste container |
| | Worker dose rate during removal operations will not exceed the allowable rate established in Issue 2.7, repository radiological design criteria (preclosure) |
| | The ability to perform borehole preparation tasks will be demonstrated |
| | The ability to remove the waste containers under normal and credible abnormal conditions will be demonstrated |
| 2.4.5 | None related to site characterization |
| 2.4.6 | The design basis for the actual retrieval period is 34 yr |
| | The ability to perform the retrieval operations using reasonably available technology is required |

Table 8.3.5.2-8. Potential abnormal conditions for retrieval

| Information need | Potential abnormal condition |
|---|---|
| 2.4.2 | Rockfall within the ramp due to a seismic event, faulting, variability in rock strength, a maintenance error, or corrosion-induced rockbolt failure |
| | Rockfall within a drift due to faulting, variability in rock strength, a maintenance error, corrosion-induced rockbolt failure, or human error resulting in excessive thermal loading |
| | Rockfall within a shaft due to faulting or variability in rock strength |

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Table 8.3.5.2-8. Potential abnormal conditions for retrieval (continued)

| Information need | Potential abnormal condition |
|---|---|
| | A ventilation system malfunction due to a seismic event, an equipment fabrication error, or a maintenance error |
| | Loss of offsite power due to a seismic event |
| 2.4.3 | Rockfall in the emplacement borehole (vertical only) due to a seismic event, faulting, variability in rock strength, or excessive thermal loading resulting from human error |
| | Axial movement of the waste container (horizontal only) due to a seismic event |
| | Waste container tilt (vertical only) due to a seismic event |
| | Shield plug jam due to a seismic event, or a fabrication error |
| | Excessive liner deflection (horizontal only) due to faulting, a fabrication error, or excessive corrosion resulting from radiolysis |
| | A collar malfunction due to a fabrication or maintenance error |
| | An auxiliary equipment malfunction due to a fabrication or maintenance error |
| 2.4.4 | A cask-collar bind due to a seismic event |
| | A dolly failure during removal (horizontal only) due to a fabrication error or excessive corrosion resulting from radiolysis |
| | A waste container pintle failure (vertical only) due to excessive corrosion resulting from radiolysis |
| | A malfunction of the transporter removal equipment due to a maintenance error |
| | Unspecified failures due to operator error including errors during alignment and waste removal |
| 2.4.5 | A transporter malfunction during transport or unloading due to a maintenance error |

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Table 8.3.5.2-8. Potential abnormal conditions for retrieval (continued)

| Information need | Potential abnormal condition |
|---|---|
| | A transporter collision with the ramp, a drift, auxiliary equipment, or another transporter due to human error |
| | Unspecified malfunctions due to operator error, including errors during alignment and waste unloading operations |

8.3.5.2.2 Information Need 2.4.2: Determination that access to the waste emplacement boreholes can be provided throughout the retrievability period for normal and credible abnormal conditions

This section describes the work that will be performed under Information Need 2.4.2 to ensure safe and reliable access to the emplacement boreholes throughout the period of retrievability and the actual retrieval period. Ensuring safe and reliable access to the emplacement boreholes consists of providing usable openings and providing an acceptable working environment for waste retrieval under both normal and credible abnormal conditions. Access to the emplacement boreholes is function 1 of the four functions discussed for this issue in the introductory material to this section.

The work performed under Information Need 2.4.2 focuses on (1) developing performance goals (design criteria) for retrieval-related aspects of the overall repository design to be developed under Issue 4.4; (2) defining the spectrum of retrieval conditions to be considered in the overall design; (3) identifying requirements for products from Issue 4.4 to be used as input items for subsequent compliance analyses; and (4) performing compliance analyses to ensure that the performance goals for function 1, access to the emplacement boreholes, are met.

Technical basis for addressing the information need

Link to the technical data chapters and applicable support documents

For Information Need 2.4.2, links to the technical data chapters fall into three categories: rock mechanics, ventilation systems, and retrieval conditions. The current drift designs are presented in Sections 6.2.6.1 through 6.2.6.3. Ground support systems for the drifts are discussed in Section 6.2.6.3.6. Ventilation system designs are presented in Section 6.2.6.5, and retrieval conditions are discussed in Section 6.4.8.2.2. Geomechanical and ventilation system analyses are presented in Section 6.4.10.2.6.

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There are numerous links to sections in the Site Characterization Plan-Conceptual Design Report (SCP-CDR) (SNL, 1987): Geomechanical discussions are contained in Appendix N of the SCP-CDR. Ventilation discussions are contained in Sections 3.4.2.2 (maintenance and retrieval), 3.4.2.3 (air cooling--vertical emplacement, and 3.4.3.3 air cooling--horizontal emplacement) and in Appendix C (ventilation and cooling analyses). Retrieval conditions are addressed in Section 6.3.1 and Appendices A (temperature), J (normal and abnormal), and L (items important to retrievability).

Parameters

As noted earlier, site data needs are specified by Issue 4.4. However, this information need requires numerous input items (i.e., products from Issue 4.4) for use in analyses to ensure that the performance goals defined for this function are met. These input items and the required content are presented in Table 8.3.5.2-9.

The normal retrieval conditions are being developed in terms of opening stability (rockfall and distortion), rock temperature, and air quality (temperature, humidity, and contaminant levels). Work completed during conceptual design on quantification of these conditions is contained in Sections 6.4.8.2.2 and 6.4.10.6.2. The current set of abnormal conditions was developed during the study of items important to retrieval (SNL, 1987, Appendix L). The list of potential abnormal conditions for Information Need 2.4.2 is presented in Table 8.3.5.2-7 in Section 8.3.5.2.1. As a result of the study of items important to retrieval, a ventilation system malfunction as a result of a seismic event or a maintenance error was considered to be the only abnormal condition that could result in a significant delay in completing retrieval operations. A significant delay was considered to be a delay of six months or more.

Logic

Information Need 2.4.2 uses the results of the performance allocation process for function 1 (Table 8.3.5.2-2) as a starting point and continues the issue resolution process as shown in Figure 8.3.5.2-1. Performance goals are taken from step five in Table 8.3.5.2-2 and used as design criteria. Requirements for input items are developed to ensure that sufficient detail and supporting evidence are available for the compliance analyses to verify that the performance goals are met. Retrieval conditions are developed using existing design information to ensure that a complete set of retrieval scenarios are considered in the design process. The performance goals (design criteria), requests for input items, and retrieval conditions are then sent to Issue 4.4, via Information Need 2.4.1, for use in developing designs, specifying supporting analyses, and for defining tests and demonstrations that are required. Specific work to be performed by Issue 4.4 for this information need includes

1. Developing drift designs using the thermomechanical analyses, G-Tunnel comparisons, and ESF tests.

2. Developing rock support systems based on analytic models, experience gained at G-Tunnel, and ESF tests.

Table 8.3.5.2-9. Input items to be provided by Issue 4.4 for Information Need 2.4.2 (access to emplacement boreholes) (page 1 of 2)

| Item number | Subject | Description |
|---|---|---|
| 1 | Drift and access design and supporting technical evidence | This item includes design concepts for the accesses and drifts and results from design analyses, tests, and demonstrations performed under Issue 4.4 that verify drift and access usability under both normal and credible abnormal conditions. The results from near-field thermomechanical modeling, exploratory shaft facility (ESF) validation testing, and demonstrations of the construction techniques performed at G-Tunnel and the ESF are required. |
| 2 | Rock support system design and supporting analyses | This item includes the design concepts for the rock support system and results from design analyses and tests performed under Issue 4.4. Specific data on estimated sizes and amounts of rockfall under normal and credible abnormal conditions are required. |
| 3 | Monitoring system design (rock movement) and supporting analyses | This item includes the basis for identifying monitoring locations and the design of the monitoring system |
| 4 | Drift and access maintenance program concepts and supporting evidence | To ensure maintainability of the drifts and accesses, maintenance program details including expected schedules, equipment requirements, and analyses used to establish the maintenance program are required. |
| 5 | Ventilation system design and supporting analyses | This item includes ventilation design concepts and supporting analyses to ensure that air quality standards are met for temperature, humidity, particulate contamination, and concentration of contaminant gases, including radon-222, under both normal and credible abnormal conditions. This requires |

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Table 8.3.5.2-9. Input items to be provided by Issue 4.4 for Information Need 2.4.2 (access to waste emplacement boreholes) (page 2 of 2)

| Item number | Subject | Description |
|---|---|---|
| 5 | Ventilation system design and supporting analyses (continued) | results from thermal, moisture, dust suppression, and air flow analyses, determination of inlet air characteristics and underground production rates of contaminant gases (personnel, equipment, and host media). |
| 6 | Basis for ensuring air quality in operational areas | This item includes the identification of the applicable regulations for air quality and the technical basis for verifying that all applicable air quality standards have been met under both normal and credible abnormal conditions. The design for the monitoring system to verify air quality is required. |
| 7 | Environment modification concepts and supporting analyses for closed drifts | Environment modification for operational areas is addressed under item 6. For this item, the concepts and supporting analyses for modification of the environment within closed drifts, for reentry purposes under both normal and credible abnormal conditions, is required. This includes environment modification concepts, including equipment modification requirements, requirements for additional equipment (if needed), results from thermal analyses, and the basis for thermal calculations. |





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3. Developing scenarios to evaluate the performance of the rock support systems under both normal and credible abnormal conditions and performing any tests or demonstrations needed.

4. Developing a maintenance program for drifts and accesses based on G-Tunnel experience, ESF tests, and experience during construction, operation, and caretaker phases. This includes developing contingency plans for installation of additional support system materials, if needed.

5. Developing monitoring systems to detect rock movement.

6. Developing a ventilation system design based on analytical models, G-Tunnel experience, and ESF tests. This development considers continuous ventilation requirements for operational areas and cooldown requirements for closed emplacement drifts.

7. Developing scenarios to evaluate the performance of the ventilation system under both normal and credible abnormal conditions and performing any tests or demonstrations that are needed.

8. Developing monitoring systems to verify air quality in operational areas and to evaluate the conditions within closed drifts before reentry.

The stability of mined openings is of particular concern. Evaluations of the thermal and mechanical effects on the stability of shafts, ramps, drifts, and boreholes have been the focus of about 15 reports or studies synopsized in SCP-CDR Section 8.3.7 (SNL, 1987). Rather than repeating the synopses here, the reader is referred to the SCP-CDR for details. These analyses have used a variety of numerical and empirical approaches: finite-element methods, boundary-element methods, and tunnel-indexing methods. Similarly, different constitutive models were employed: elastic, ubiquitous-joint, compliant-joint, and elastic-plastic models. Other items that have been varied in some of the analyses include opening sizes and shapes, depths, thermal and mechanical properties, fracture properties, and in situ conditions. The common conclusions drawn from the approaches used to date are

1. Drifts, shafts, and ramps, as currently designed, are predicted to remain stable during preclosure.

2. Waste emplacement boreholes are predicted to remain stable during preclosure, although some potential exists for negligible amounts of rock to fall on the liner planned for use in horizontal emplacement holes.

3. Excavation-induced responses of openings in the Topopah Spring tuff should be expected to be similar to those in the Grouse Canyon tuff in G-Tunnel.

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Further studies are planned during the advanced conceptual design phase to evaluate retrieval under potential abnormal conditions like those listed in Table 8.3.5.2-8 in the technical basis section for Information Need 2.4.1 (Section 8.3.5.2.1).

The results of this work in the form of input item responses are obtained from Issue 4.4. A compliance analysis to evaluate whether the design actually provides for the ability to access the emplacement boreholes as required is then performed under Information Need 2.4.2. As shown in Figure 8.3.5.2-1, this compliance analysis evaluates the completeness and sufficiency of the responses to the input items and the retrieval conditions and determines whether the performance goals have been met. For function 1, this involves evaluating the drift designs, rock support system and monitoring system (rock movement) designs, the maintenance program, and all the supporting evidence (results from analyses, G-Tunnel comparisons, ESF tests, and scenario development) to verify that usable openings will be available for 84 yr. In addition, the compliance analysis involves evaluating the ventilation system design, air quality monitoring system design, and all of the supporting evidence (results from ventilation system analyses, G-Tunnel tests, ESF tests, and scenario development) to verify that an acceptable environment can be established within the drifts. Negative responses to the three tests for the compliance analysis can be followed in Figure 8.3.5.2-1. They involve modification of input items, performance goals, or the design. If modification is not possible, a noncompliance exists for the design. If the results of all three tests are positive, then, relative to function 1 (access to the boreholes), compliance exists for the design. The results of the compliance analysis for function 1 are sent to Information Need 2.4.6 to be combined with the results from the other information needs for publication as a topical report.

### 8.3.5.2.3 Information Need 2.4.3: Determination that access to the waste packages can be provided throughout the retrievability period for normal and credible abnormal conditions

This discussion describes the work that will be performed under this information need to ensure safe and reliable access to the emplaced waste package (function 2). As indicated in Table 8.3.5.2-3 the design of the emplacement envelope (borehole, liner, shield plug, and collar) is of primary concern relative to providing access to the waste packages for both normal and credible abnormal conditions.

The work performed here, similar to the previous information need, focuses on (1) developing performance goals (design criteria) for equipment and operations related to the maintenance of access to the emplaced waste, (2) defining retrieval conditions for the emplacement envelope, (3) identifying requirements for products from Issue 4.4 to be used as input items for subsequent compliance analysis, and (4) performing a compliance analysis to ensure that the performance goals for function 2, access to the waste packages, are met.

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Technical basis for addressing the information need

Link to the technical data chapters and applicable support documents

The emplacement envelope layouts are discussed in Sections 6.2.6.2 and 6.2.6.3 for the vertical and horizontal configurations, respectively. Operations are discussed in Section 6.2.3.2. Geomechanical analyses for the emplacement borehole are presented in Section 6.4.10.2.6, and retrieval conditions are discussed in Section 6.4.8.2.2.

There are numerous links to sections in the SCP-CDR (SNL, 1987): Geomechanical discussions are contained in Appendix N. Equipment discussions are contained in Section 3.2.2.2 (operations) and Appendices B (liner stress analysis), D (equipment descriptions), and J (retrieval operations). Retrieval conditions are addressed in Section 6.3.1 and Appendices A (borehole temperatures), J (normal and abnormal), and L (items important to retrievability).

Parameters

This information need requires three input items from Issue 4.4 for use in analyses to ensure that the performance goals defined for function 2, access to the emplaced waste, are met. These required input items are presented in Table 8.3.5.2-10.

The normal retrieval conditions are being developed in terms of borehole stability (rockfall and distortion), borehole rock temperature, radiation levels, and condition of the liner. Work completed to quantify these conditions is described in Sections 6.4.8.2.2. The current set of abnormal conditions was developed during the study of items important to retrieval (Appendix L of SNL, 1987). The list of potential abnormal conditions for Information Need 2.4.3 is presented in Table 8.3.5.2-7 in Section 8.3.5.2.1. As a result of the study of items important to retrieval, three conditions were identified that could result in a significant delay in completing retrieval operations:

1. In the vertical configuration, rockfall into the borehole could occur as a result of a seismic event.

2. In the vertical configuration, a waste container misalignment or "tilt" in the borehole could result from a seismic event.

3. Shield plugs could jam as the result of a seismic event.

Logic

Information Need 2.4.3 uses the results of the performance allocation process for function 2 (see Table 8.3.5.2-3) as a starting point and continues the issue resolution process as shown in Figure 8.3.5.2-1. Performance goals are taken from Table 8.3.5.2-3 and used as design criteria. Input item requirements are developed to ensure that sufficient detail and supporting evidence are available for the compliance analysis. Retrieval

Table 8.3.5.2-10. Input items to be provided by Issue 4.4 for Information Need 2.4.3 (access to waste packages)

| Item number | Subject | Description |
|---|---|---|
| 1 | Waste emplacement envelope design and supporting analyses | The design concepts and supporting analyses used for the waste emplacement envelope are required. This includes providing estimates of rockfall within the borehole (type and amount), borehole distortion, liner deflection, liner stress, liner corrosion rate, and liner lifetime under normal and credible abnormal conditions. |
| 2 | Waste emplacement envelope assessment | This item includes the normal and credible abnormal conditions for the waste emplacement envelope, requirements for assessment of the conditions of the waste emplacement envelope (at the time of retrieval), assessment of equipment design and supporting analyses, and the requirements for and results from any tests or demonstrations. |
| 3 | Corrective actions | This item includes the identification of corrective actions that may be required under normal and credible abnormal conditions, the design and supporting analyses for equipment to perform the corrective actions, and the results of any tests or demonstrations. |



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conditions are developed using existing design information to ensure that a complete set of retrieval scenarios is considered in the design process. Specific work to be performed by Issue 4.4 for this information need includes

1. Developing borehole designs using thermomechanical analyses, G-Tunnel comparisons, and exploratory shaft facility (ESF) tests.

2. Designing the liner based on analytic models, experience in the mining industry, and corrosion test results.

3. Designing the shield plug based on analytic models and experience in the nuclear industry.

4. Designing the shielding closure based on analytic models and experience in the nuclear industry.

5. Developing scenarios to estimate the performance of the emplacement envelope under both normal and credible abnormal conditions and performing any tests or demonstrations needed. It is anticipated that proof-of-principle demonstrations may be required for some of the equipment related to retrieval under abnormal conditions. The reader is referred to Section 3.2.2 and Appendix J in the SCP-CDR (SNL, 1987).

6. Developing the equipment required to verify the condition of the waste emplacement envelope before waste removal.

7. Developing the equipment and operations to perform corrective actions that may be required to restore acceptable access to the waste packages.

The conditions within the emplacement boreholes can be characterized in terms of the following parameters: rock temperature, condition of the opening, radiation levels, and condition of the borehole liner.

1. The predicted temperature histories for the emplacement boreholes for the vertical and horizontal emplacement concepts are discussed in the SCP-CDR, Appendix J (SNL, 1987). As shown in that section, the temperature remains above 100°C throughout the retrievability period; therefore, a dry environment in the emplacement borehole is anticipated.

2. For the vertical emplacement concept, the borehole is expected to be stable with negligible amounts of rockfall into the emplacement borehole under normal conditions. For the horizontal concept, minor rockfall against the liner is anticipated. In addition, as noted previously, a dry environment, as a result of high temperatures, is expected.

3. At the time of emplacement, the waste container surface radiation levels for spent fuel (pressurized water reactor) are estimated at $1 \times 10^5$ rem/h for gamma and $1 \times 10^2$ mrem/h for neutron radiation (O'Brien, 1985). These surface radiation levels are used as the worst-case levels for shielding design.

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4. Under normal conditions, the liner will be intact and provide acceptable access to the emplaced waste containers throughout the design-basis 84-yr retrievability period.

Further studies are planned during the advanced conceptual design phase to evaluate retrieval under credible abnormal conditions like those listed in Table 8.3.5.2-8 in Section 8.3.5.2.1.

Issue 4.4 returns the results of this work in the form of input item responses. Information Need 2.4.3 performs a compliance analysis to evaluate whether the design actually provides for the ability to access the emplacement boreholes as required. As shown in Figure 8.3.5.2-1, this compliance analysis evaluates the completeness and sufficiency of the responses to the input items and the retrieval conditions and determines whether the performance goals have been met. For function 2, this involves evaluating the emplacement envelope design (borehole, liner, shield plug, and shielding collar), the concepts for the assessment of the condition of the waste emplacement envelope, the equipment design and operations for performing corrective action, and supporting evidence (analyses, G-Tunnel comparisons and tests, experience in the mining and nuclear industries, ESF tests, corrosion tests, demonstrations, and scenarios). The results of the compliance analysis for function 2 are sent to Information Need 2.4.6 to be combined with the results from the other information needs.

### 8.3.5.2.4 Information Need 2.4.4: Determination that the waste can be removed from the emplacement boreholes for normal and credible abnormal conditions

The discussion under this information need describes the work that will be performed to ensure that the ability to remove the emplaced waste from the emplacement boreholes is maintained (function 3). Design of the waste package and the transporter waste removal equipment is of primary concern relative to providing the ability to remove the emplaced waste for both normal and credible abnormal conditions.

The work performed under Information Need 2.4.4 focuses on the four steps: (1) developing performance goals, (2) defining retrieval conditions for the waste removal operations, (3) identifying input items and their needed content, and (4) performing a compliance analysis to ensure that the performance goals are met.

#### Technical basis for addressing the information need

Link to the technical data chapters and applicable support documents

Links to the conceptual designs of the repository and waste package (Chapters 6 and 7, respectively) include three categories: removal equipment design and supporting analyses, waste package design and supporting analyses, and retrieval conditions. Equipment design is discussed in Sections 6.2.6.2 (vertical emplacement mode), 6.2.6.3 (horizontal emplacement mode), and

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6.2.9.3 (equipment development). Operations are discussed in Section 6.2.3.2. The current waste package designs and supporting analyses are presented in Sections 7.3 and 7.4. Retrieval conditions are discussed in Section 6.4.8.2.2.

Sections in the SCP-CDR (SNL, 1987) contain discussions related to waste removal from the boreholes. Equipment discussions are contained in Section 3.2.2.2 (operations) and Appendices B (liner stress analysis), D (equipment descriptions), and J (retrieval operations). Waste package discussions are contained in Section 2.1 (basis). Retrieval conditions are addressed in Section 6.3.1 and Appendices A (temperature), J (normal and abnormal), and L (items important to retrievability).

Parameters

This information need requires three input items from Issue 4.4 to ensure that the performance goals defined for function 3 (waste removal) are met. These required input items are presented in Table 8.3.5.2-11.

Normal retrieval conditions are being identified for the waste removal equipment and waste packages. Work completed to quantify the conditions is described in Section 6.4.8.2.2. The current set of credible abnormal conditions was developed during the study of items important to retrieval (Appendix L of SNL, 1987). The list of potential abnormal conditions for Information Need 2.4.4 is presented in Table 8.3.5.2-8 in Section 8.3.5.2.1. As a result of the items important to retrieval study, no conditions were identified that could result in a significant delay in completing retrieval operations.

Logic

Information Need 2.4.4 uses the results of the performance allocation process for function 3 (Table 8.3.5.2-4) as a starting point and continues the issue resolution process as shown in Figure 8.3.5.2-1. Performance goals are taken from Table 8.3.5.2-4 and used as design criteria. Input item requirements are developed to ensure that sufficient detail and supporting evidence are available for the compliance analysis. Retrieval conditions are developed using existing design information to ensure that a complete set of retrieval scenarios are considered in the design process. Specific work to be performed by Issue 4.4 for this information need includes

1. Designing the transporter waste removal equipment based on analytic models, scale models, component testing, and full scale tests (if required).

2. Specifying the design of the waste package interface with retrieval equipment based on analytic models, experience in the nuclear industry, and extensive testing.

3. Developing scenarios to estimate the performance of the waste removal equipment and the waste package under both normal and abnormal conditions and performing any tests or demonstrations needed. It is anticipated that proof-of-principle and prototype demonstrations may be required for some of the removal equipment.

Table 8.3.5.2-11. Input items to be provided by Issue 4.4 for Information Need 2.4.4 (removal of waste from boreholes)

| Item number | Subject | Description |
|---|---|---|
| 1 | Waste package removal design and supporting analyses | This item includes the design concepts and supporting analyses for the waste package removal equipment, the waste package, the dolly (if used), and the shielding collar. |
| 2 | Concepts for borehole preparation for retrieval and supporting evidence | This item includes the design concepts and the supporting analyses related to preparation of the emplacement borehole for waste retrieval under normal and credible abnormal conditions. |
| 3 | Demonstrations of borehole preparation and waste removal | The requirements for and results of any demonstrations for borehole preparation and waste removal under normal and credible abnormal conditions are required. |



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reader is referred to Section 3.2.2 and Appendix J of the SCP-CDR (SNL, 1987) for details of the scenarios and equipment considered to date.)

4. Developing the equipment required to verify the condition of the waste package before waste removal.

5. Developing the equipment and operations to perform corrective actions that may be required to remove the waste package under credible abnormal conditions.

The DOE is evaluating the need for demonstration of selected equipment. The need for testing certain equipment was evaluated as part of the conceptual design and will be further evaluated as part of subsequent design stages.

Results of this work will be returned from Issue 4.4 in the form of input item responses. Information Need 2.4.4 performs a compliance analysis to evaluate whether the design actually provides for the ability to remove the emplaced waste as required. As shown in Figure 8.3.5.2-1, this compliance analysis evaluates the completeness and sufficiency of the responses to the input items and the retrieval conditions and determines whether the performance goals have been met. For function 3, this involves evaluating the transporter waste removal equipment design, the waste package design, the dolly design (horizontal only), the concepts for verifying the condition of the waste package and dolly (horizontal only), the equipment design and operations for performing corrective actions, and all supporting evidence (results from tests and analyses, experience in industry, scenarios, and demonstrations). The results of this compliance analysis for function 3 are sent to Information Need 2.4.6 to be combined with the results of the other information needs.

### 8.3.5.2.5 Information Need 2.4.5: Determination that the waste can be transported to the surface and delivered to the waste-handling surface facilities for normal and abnormal conditions

This section describes the work that will be performed under this information need to ensure that the ability to transport the retrieved waste and unload it at the surface waste-handling building is maintained (function 4). The design of the transporter and the surface unloading equipment is of primary concern relative to providing the ability to transport and unload the retrieved waste for both normal and credible abnormal conditions.

The work performed under this information need focuses on the familiar four steps: (1) developing performance goals (design criteria) for the equipment and operations associated with waste transport and unloading, (2) defining retrieval conditions for the waste transport and unloading operations, (3) identifying requirements for input items, and (4) performing a compliance analysis to ensure that the performance goals for function 4 are met.

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Technical basis for addressing the information need

Link to the technical data chapters and applicable support documents

For Information Need 2.4.5, the links fall into these categories: transporter design and supporting analyses, unloading equipment (surface facility waste-handling building) design and supporting analyses, and retrieval conditions. The transporter is discussed in Sections 6.2.6.2 (vertical emplacement mode), 6.2.6.3 (horizontal emplacement mode), and 6.2.9.3 (equipment development). Operations including those for retrieval are discussed in Section 6.2.3.2. Operations for the waste-handling building are presented in Section 6.2.4. Retrieval conditions are discussed in Section 6.4.8.2.2.

There are links to similar sections in the SCP-CDR (SNL, 1987): Equipment discussions are contained in Section 3.2.2.2 (operations) and Appendices D (equipment descriptions) and J (retrieval operations). The surface facility waste handling is discussed in Sections 3.1 (operations) and 4.2 (design). Retrieval conditions are addressed in Section 6.3.1 and Appendices J (normal and abnormal conditions) and L (items important to retrievability).

Parameters

This information need requires four input items from Issue 4.4 to ensure that the performance goals defined for function 3 (waste removal) are met. These required input items are presented in Table 8.3.5.2-12.

Normal retrieval conditions are being identified for the transporter and the unloading equipment. Work completed during conceptual design on quantification of these conditions is contained in Section 6.4.8.2.2. The current set of credible abnormal conditions was developed during the study of items important to retrieval (Appendix L of SNL, 1987). The list of potential abnormal conditions for Information Need 2.4.5 is presented in Table 8.3.5.2-8 in Section 8.3.5.2.1. As a result of the items important to retrieval study, one condition was identified that could result in a significant delay in completing retrieval operations. This abnormal condition involved a transporter collision with the ramp wall as the result of an operator error.

Logic

As noted previously, Information Need 2.4.5 was derived from function 4, transport and unload the waste at the surface. Information Need 2.4.5 uses the results of the performance allocation process for function 4 (Table 8.3.5.2-5) as a starting point and continues the issue resolution process as shown in Figure 8.3.5.2-1. Performance goals are taken from Table 8.3.5.2-5 and used as design criteria. Input item requirements are developed to ensure that sufficient detail and supporting evidence are available for the compliance analysis. Retrieval conditions are developed using existing design information to ensure that a complete set of retrieval scenarios are consid-ered in the design process. Specific work to be performed by Issue 4.4 for this information need includes

Table 8.3.5.2-12. Input items to be provided by Issue 4.4 for Information Need 2.4.5 (delivery of waste to surface facilities)

| Item number | Subject | Description |
|---|---|---|
| 1 | Transporter design concepts and supporting analyses | This item includes the design and analyses for the transporter under normal and credible abnormal conditions, including the propulsion system, braking system, steering, and radiation shielding. |
| 2 | Unloading equipment design and supporting analyses | This item includes the design and supporting analyses for the unloading equipment within the transporter cask and the surface facility equipment for unloading the waste from the transporter under normal and credible abnormal conditions. |
| 3 | Demonstrations for waste transport | This item includes the requirements for and the results of demonstrations, if required, of the ability to transport waste. |
| 4 | Demonstrations for waste unloading | This item includes the requirements for and the results of of demonstrations, if required, of the ability to unload waste at the surface waste-handling building. |


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1. Designing the transporter based on analytic models, existing equipment, scale models, component testing, and full-scale tests (if required).

2. Designing the surface unloading equipment based on analytic models, experience in the nuclear industry, component testing, and scale models (if required).

3. Developing scenarios to estimate the performance of the transporter and unloading equipment under both normal and credible abnormal conditions and performing any tests or demonstrations that are needed. It is anticipated that proof-of-principle and prototype demonstrations may be required for the transporter and unloading equipment. The reader is referred to Section 3.2.2 and Appendix J of the SCP-CDR (SNL, 1987) for additional information.

Issue 4.4 returns the results of this work in the form of input item responses. Information Need 2.4.5 performs a compliance analysis to evaluate whether the design actually provides for the ability to transport and unload the waste as required. As shown in Figure 8.3.5.2-1, this compliance analysis evaluates the completeness and sufficiency of the responses to the input items and the retrieval conditions and determines whether the performance goals have been met. For function 4, this involves evaluating the transporter design, the design of the unloading equipment at the surface, and all supporting evidence (results from tests, analyses, and demonstrations, experience in the nuclear and mining industries, and results from scenario development). The results of this compliance analysis are sent to Information Need 2.4.6 to be combined with the results from the other information needs.

### 8.3.5.2.6 Information Need 2.4.6: Determination that the retrieval requirements set forth in 10 CFR 60.111(b) are met using reasonably available technology

The discussion under this information need describes the work that will be performed to ensure that the requirements for retrievability contained in 10 CFR 60.111(b) and that the requirement for the use of reasonably available technology imposed by 10 CFR 960.5-1(a)(3) will be met.

The work performed under Information Need 2.4.6 focuses on (1) developing performance goals (design criteria), (2) identifying requirements for input items, and (3) performing a compliance analysis to ensure that the performance goals for retrievability shown in Table 8.3.5.2-7 are met.

Technical basis for addressing the information need

As shown in Figure 8.3.5.2-1, there are four functions that must be performed in order to retrieve emplaced waste. Information Need 2.4.6 combines the results from the other information needs under this issue, and verifies the ability to retrieve any or all of the emplaced waste is maintained throughout the period of retrievability. In addition, Information Need 2.4.6 imposes two additional requirements: (1) that the repository

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design allows for retrieval to be performed on a reasonable schedule and (2) that the repository design includes the use of technology that will be reasonable at the time of repository construction.

Link to the technical data chapters and applicable support documents

For Information Need 2.4.6, the links fall into the following two categories: retrieval schedule and use of reasonably available technology. The design basis retrieval schedule is discussed in Sections 6.1.1.6.4 and 6.2.9.1. The use of reasonably available technology is discussed in Section 6.4.10. In the SCP-CDR (SNL, 1987), time lines for retrieval and the time consequences of abnormal conditions are included in Appendix L-2 (items important to retrievability).

Parameters

This information need requires three input items from Issue 4.4 to ensure that the performance goals for Information Need 2.4.6 are met. These required input items are presented in Table 8.3.5.2-13.

The performance goals (design criteria) developed for this information need are located in Table 8.3.5.2-8 in Section 8.3.5.2.1. These goals were developed as a result of the requirements for retrieval to be completed on a reasonable schedule and for the use of reasonably available technology.

Logic

The work to be accomplished under Information Need 2.4.6 is aimed at verifying that all of the retrievability requirements set forth in 10 CFR 60.111(b) are met using reasonably available technology. To accomplish this, it must be established that the repository design

1. Includes the option to retrieve any or all of the emplaced waste throughout the period of retrievability.

2. Allows for the completion of retrieval of any or all of the emplaced waste on a reasonable schedule.

3. Incorporates the use of reasonably available technology.

Ensuring that the option to retrieve waste is preserved involves verifying that the four retrieval functions can be performed. Information Needs 2.4.2 through 2.4.5 correspond to retrieval functions 1 through 4, respectively. Each of these information needs will complete a compliance analysis relative to a retrieval function and will forward the results to Information Need 2.4.6.

To ensure that the requirements for a reasonable schedule and reasonably available technology are met, Information Need 2.4.6 (1) develops performance goals (design criteria) to ensure that the design considers these requirements, (2) develops input item requirements to ensure that sufficient detail and supporting evidence are available to verify compliance, a (3) performs a compliance analysis to verify that the design meets the performance goals.

Table 8.3.5.2-13. Input items to be provided by Issue 4.4 for Information Need 2.4.6 (compliance with retrieval requirements)

| Item number | Subject | Description |
|---|---|---|
| 1 | Reference operations plan | A complete operations plan is required to ensure compliance with the 10 CFR 60.111 (b) "reasonable schedule" requirement for retrieval. |
| 2 | Use of reasonably available technology | This item includes all technical evidence which confirms the use of reasonably available technology for all retrieval-related equipment. |
| 3 | Reference design and supporting analyses | In support of the compliance analyses, reference design information and supporting analyses for the underground facilities, the surface facilities, and repository equipment are required. |



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The performance goals (design criteria) and requests for input items are sent to Issue 4.4 for use in developing the design, specifying supporting analyses, and defining tests and demonstrations that are required. Specific work to be performed by Issue 4.4 for this information need includes

1. Developing a reference design and performing required supporting analyses.

2. Developing a reference operations plan.

3. Performing the activities necessary to prove the design is based on reasonably available technology.

Issue 4.4 returns the results of this work in the form of input item responses. Information Need 2.4.6 performs a compliance analysis to evaluate whether the design actually provides for the ability to retrieve on a reasonable schedule and uses reasonably available technology. As shown in Figure 8.3.5.2-1, this compliance analysis evaluates the completeness and sufficiency of the retrieval conditions and the responses to the input items and determines whether the performance goals have been met. Specifically, Information Need 2.4.6 evaluates the complete design package relative to the ability to perform retrieval in a reasonable period of time and with the use of reasonably available technology. Negative responses to the three tests in the compliance analysis can be followed in Figure 8.3.5.2-1. They involve modification of input items, performance goals, or the design. If modification is not possible, a noncompliance exists for the design. If the results of all three tests are positive, then, relative to the reasonable schedule and reasonably available technology requirements, compliance exists for the design.

The results of the compliance analysis conducted under Information Need 2.4.6 are combined with the compliance analyses conducted under Information Needs 2.4.2 through 2.4.5 to create a compliance analysis for Issue 2.4, waste retrievability. The objective of this compliance analysis is to demonstrate that all of the performance goals relative to preserving the option of waste retrieval as set forth in 10 CFR Part 60.111(b) using reasonably available technology are met, and that Issue 2.4, waste retrievability, is resolved.

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8.3.5.3 Issue resolution strategy for Issue 2.1: During repository operation, closure, and decommissioning (a) will the expected average radiation dose received by members of the public within any highly populated area be less than a small fraction of the allowable limits and (b) will the expected radiation dose received by any member of the public in an unrestricted area be less than the allowable limits as required by 10 CFR 60.111, 40 CFR 191 Subpart A, and 10 CFR Part 20?

This issue is concerned with the radiation exposure to the general public from the normal operation, closure, and decommissioning of the repository. The issue is divided into two parts: (a) the exposure to members of the public in a highly populated area (a highly populated area is defined in 10 CFR 960.2) and (b) the maximum exposure to any member of the public. The lower radiation dose limit stated in part (a) is intended to limit the total population dose (man-rem exposure). To address part (a) of this issue, the locations of the highly populated areas must be determined in relation to the repository site. To address part (b) of this issue, the dose to individuals in the vicinity of the site must be evaluated. The assessment of the potential doses will allow an evaluation of the impact of the operation, closure, and decommissioning of the repository on the surrounding population. The assessments will be conducted periodically (i.e., at each design phase) throughout the design of the repository to provide feedback to the design process. A monitoring program will provide verification of the results of the analyses. Note that the as low as reasonably achievable (ALARA) criterion (10 CFR 20.1) will be applied in designing the repository to minimize the potential radiation dose to the public. The DOE is presently evaluating how the limits in 40 CFR Part 191 relate to the ALARA criterion. Any decisions will be incorporated into the issues resolution strategy for this issue.

The relationship of this issue with the other issues of the issues hierarchy is shown on an overall scale in Figure 8.3.2.1-1 (Section 8.3.2.1), which illustrates the relationship between design and performance issues and fixes the lines of communication between these issues. To emphasize the relationship of this issue to the other issues with which it has direct or very strong ties, only Issues 2.1 (this issue), 2.2 (Section 8.3.5.4), 2.3 (Section 8.3.5.5), 2.7 (Section 8.3.2.3), and 4.4 (Section 8.3.2.5) are illustrated in Figure 8.3.5.3-1. The figure defines the ties between these issues by indicating the major information items passed between them. The figure also illustrates the connection of all these issues with the site characterization program. The methods to perform preclosure safety analyses are also discussed in Section 8.3.5.1. The scope of an issue is indicated by its size with respect to the other issues in the figure. Note that Issue 4.4 is the largest in scope, and the other issues, including this issue, branch out from Issue 4.4, reducing the scope to more specific areas. In the discussion that follows in this section, the regulatory basis for addressing this issue is presented, the approach to resolving this issue is described, and the interrelationships among the information needs are discussed.

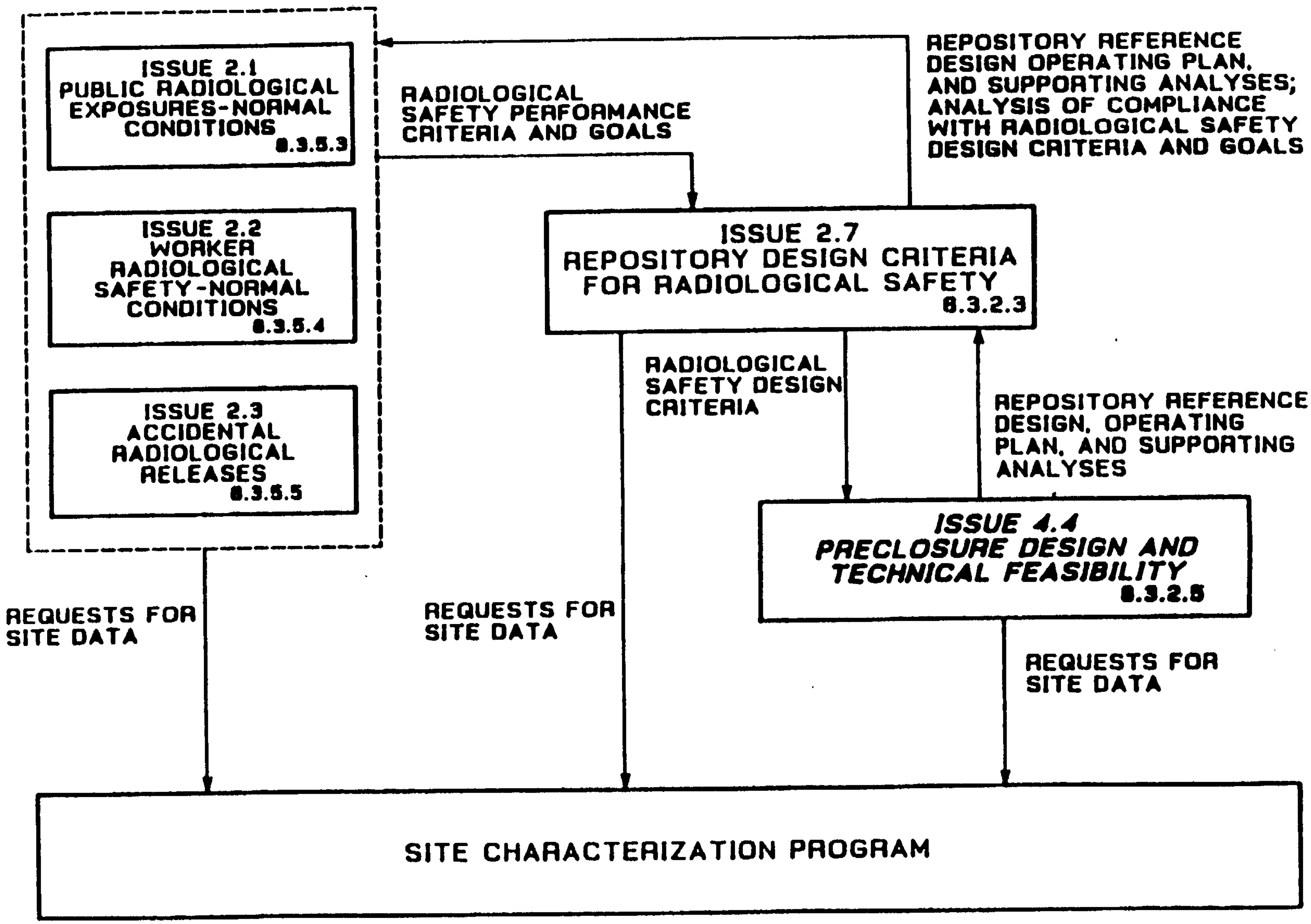


**Figure 8.3.5.3-1.** Relationship of Issue 2.1 (public radiological exposures–normal conditons) to other issues and the site characterization program



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Regulatory basis for the issue

As stated in this issue, the allowable exposure limits are those specified in 10 CFR 60.111, 40 CFR Part 191 Subpart A, and 10 CFR Part 20. In fact, 10 CFR 60.111 only requires conformance with 10 CFR Part 20 and "such generally applicable environmental standards for radioactivity as may have been established by the Environmental Protection Agency" (i.e., 40 CFR Part 191 Subpart A). 10 CFR 60.111 does not impose any additional requirements; therefore, the only regulatory requirements directly applicable to this issue are those in 10 CFR Part 20 and 40 CFR Part 191 Subpart A:

1. 10 CFR 20.105, Permissible levels of radiation in unrestricted areas.

2. 10 CFR 20.106, Radioactivity in effluents to unrestricted areas.

3. 40 CFR 191.03, Standards. This section contains limits on radiation doses to members of the public.

The objective of 10 CFR 20.105 and 20.106 is to limit the radiation dose that members of the public in unrestricted areas may receive to less than 0.5 rem per year to the whole body and other limits specified for particular organs. In addition, 10 CFR 20.1(c) requires that the exposures be maintained as low as reasonably achievable (ALARA). The DOE is currently evaluating how the limits in 40 CFR Part 191 relate to the ALARA criterion.

40 CFR 191.03(a) requires that "management and storage of spent nuclear fuel or high-level or transuranic radioactive wastes at all facilities regulated by the Commission or by Agreement States shall be conducted in such a manner as to provide reasonable assurance that the combined annual dose equivalent to any member of the public in the general environment resulting from: (1) Discharges of radioactive material and direct radiation from such management and storage and (2) all operations covered by Part 190; shall not exceed 25 millirems to the whole body, 75 millirems to the thyroid, and 25 millirems to any other critical organ."

In addition, there are other sections of 10 CFR Part 60 that require compliance with 10 CFR 60.111 and 10 CFR Part 20; these sections, however, do not contain any additional exposure limits relevant to the issue. They include the following:

1. 10 CFR 60.131, General design criteria for the geologic repository operations area.

2. 10 CFR 60.132, Additional design criteria for surface facilities in the geologic repository operations area.

3. 10 CFR 60.133, Additional design criteria for the underground facility.

Section 8.3.2.3 contains a detailed discussion of the design criteria in 10 CFR 60.131 through 60.133. In addition, it is Office of Civilian Radioactive Waste Management (OCRWM) policy that DOE Orders will be followed where they do not conflict with NRC requirements.

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Approach to resolving this issue

Licensing strategy overview

Part (a) of this issue (will the expected average radiation dose received by members of the public within any highly populated area be less than a small fraction of the allowable limits) is not a requirement of the NRC or EPA, but is a qualifying condition on population density and distribution in 10 CFR 960.5-2-1(a)(1). As such, information and results used in resolution of part (a) of Issue 2.1 will also be applicable to the resolution of the corresponding part of Issue 2.5 (Section 8.3.5.6), which deals with the higher level findings of 10 CFR 960.5. This part can be analyzed on the basis of repository design and operational controls, identification of population density and distribution, location of members of the public in the unrestricted area, and calculation of radiation doses to individuals and population groups from the repository and other sources. The part of this issue that deals with population distribution and location of members of the public is the subject of the Population Density Program 8.3.1.10. The remainder of part (a) of Issue 2.1 deals with repository design and assessment of the projected radiation exposures and is within the scope of part (b) of this issue.

Part (b) of this issue (will the expected radiation dose received by any member of the public in an unrestricted area be less than the allowable limits as required by 10 CFR 60.111, 40 CFR 191 Subpart A, and 10 CFR Part 20) addresses radiation doses from the repository and from other potential sources (regulated under 40 CFR Part 190) to nearby individuals. To determine this expected radiation dose, the unrestricted area must be defined and doses to the nearby individuals determined from both repository operation and other uranium fuel cycle facilities. Calculation of this dose will be performed using acceptable analytical models that require radionuclide source terms, locations of release points, location of nearby individuals, exposure pathways, meteorologic, and hydrologic parameters. This will require an iterative analysis because the location of the nearby individuals may change since the contribution from the repository to the combined dose may change both in magnitude and location as the repository design matures (i.e., as predicted source terms and release points may change). Radiation doses from other facilities from all pathways to the unrestricted area will be determined to ensure that the combined doses and radionuclide concentrations are less than the allowable limits.

Doses in the unrestricted area may be derived from direct radiation from sources inside the repository boundary, direct radiation from repository radioactive airborne emissions, inhalation of these airborne radioactive material emissions, and ingestion of radioactive material from liquids and foodstuffs contaminated by radioactive material. Radiation doses to individuals in the unrestricted area are expected to be primarily due to gaseous radioactive material released during waste handling and packaging operations. Doses are expected to be reduced to levels well below the allowable limits by design features such as filtration and by natural dispersion in the atmosphere.

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Additional potential sources of radiation dose to unrestricted areas include radon and radon daughters from the underground portions of the repository that may be entrained in subsurface ventilation air and discharged at surface release points and from radon and other naturally occurring radionuclides that may be released from muck stored on the surface. These releases are not within the scope of Subpart A of 40 CFR Part 191. Even though these releases also do not appear to be within the scope of 10 CFR Part 20, their contribution to offsite releases to unrestricted areas will be assessed. Therefore, analyses are required to quantify the emanation rate of this radionuclide from the mine and from the muck pile. Sources of radiation exposure from the transportation of high-level waste (HLW) to the repository are expected to be addressed under Issue 3.3, as part of the environmental program planned activities. (Transportation of HLW to the repository is excluded from the definition of site characterization by the Nuclear Waste Policy Act.) Transportation of this HLW within the repository boundaries will be considered part of the repository program.

In addition to a primary focus on ensuring radiation doses to the public are at a very low level, both 10 CFR Part 20 and Part 60 require the verification of performance. This requirement for performance verification necessitates the design and installation of in-plant radiation measurement systems for effluent monitoring with alarm mechanisms to warn of significant increases in radioactivity. The radiation monitoring systems must monitor and record concentrations of radioactive material in the effluents and in the surrounding environment. Data from these systems are required to determine radiation exposures to the public and to verify they are within regulatory limits. These requirements are discussed in the Project Radiological Monitoring Plan, which is discussed in Section 8.3.1.13.

In summary, the repository will be designed to limit the expected radiation dose received by any member of the public in an unrestricted area to less than the allowable limits required by 40 CFR Part 191 Subpart A and 10 CFR Part 20. Computer models will be used to evaluate the potential of radiation exposure of any member of the public in the unrestricted area. The performance verification systems, which will be designed and constructed to comply with 10 CFR Part 20 and Part 60 requirements, will be used during operations to ensure that the as-built repository will meet regulatory dose requirements. The preclosure performance monitoring and confirmation program (see Section 8.3.5.16) will provide the mechanism for corrective action, either operational or design, which will ensure successful compliance.

The resolution of this issue will be accomplished by the analysis of the repository design and operational controls and activities and calculation of doses to members of the public in unrestricted areas to ensure that the doses meet allowable limits and are as low as reasonably achievable.

Application of the issue resolution strategy

The logic to be used in the resolution of this issue is illustrated in the logic diagram shown in Figure 8.3.5.3-2. This logic diagram depicts how the generic issue resolution strategy of Section 8.2.2 is to be applied to this issue. The first step of the process (identifying regulatory requirements) was discussed earlier in the section called Regulatory basis for addressing the issue. The following discussions will explain each of the

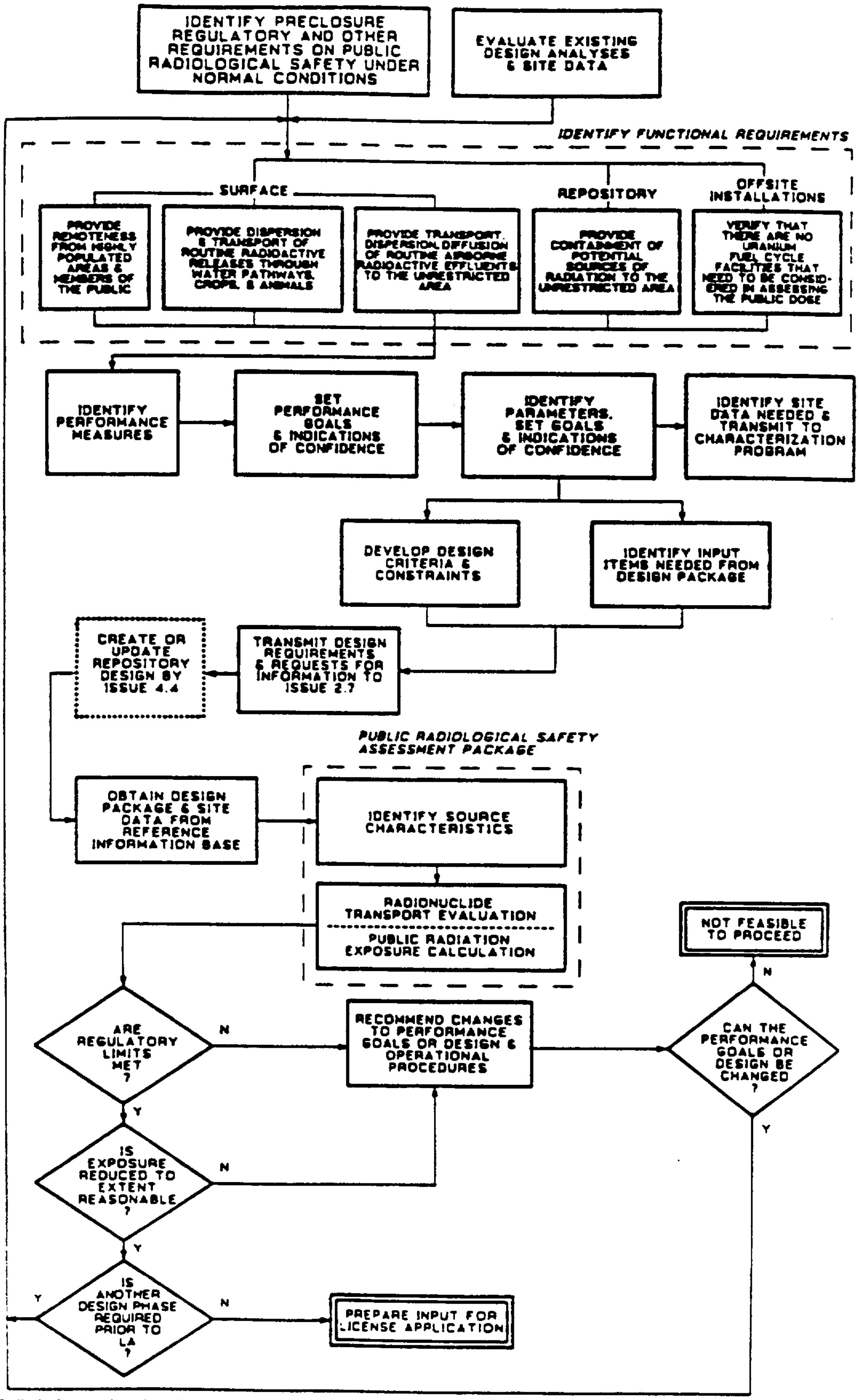


Figure 8.3.5.3-2a. Logic diagram for Issue 2.1 (public radiological exposures–normal conditions). See Figure 8 3.5.3-2b for legend. Section 8.3.2.1 describes the relationships and interfaces between design and performance issues.

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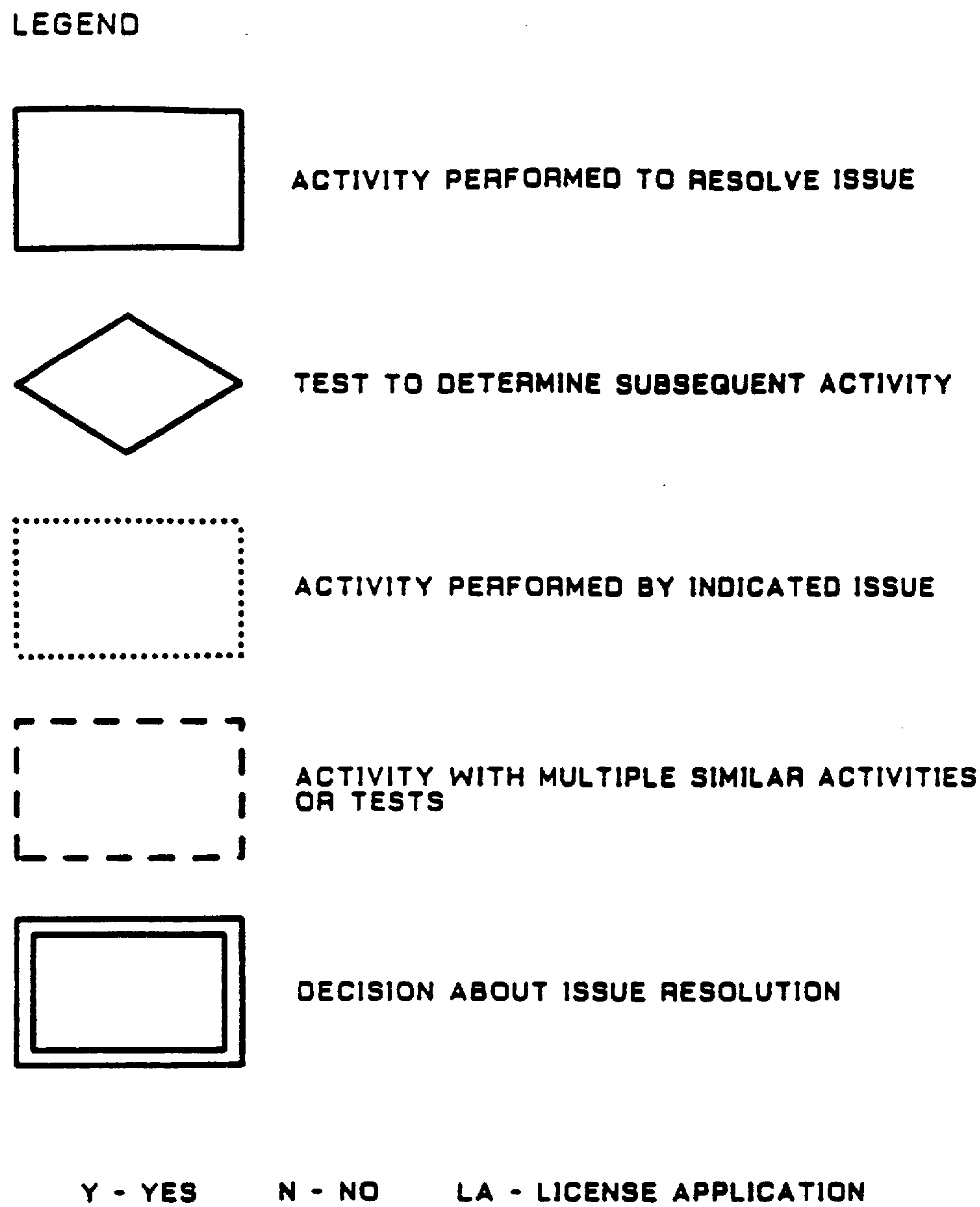


Figure 8.3.5.3-2b. Legend for Figure 8.3.5.3-2a.

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remaining steps in the resolution of this issue as shown in the logic diagram.

Identification of functional requirements. To allocate performance in this issue to specific system elements of the mined geologic disposal system (MGDS) at Yucca Mountain, the functions of these system elements with respect to this issue must be identified. The preclosure portion of the Yucca Mountain MGDS is divided into three major system elements: the site, the repository, and the waste package. The waste package will not be considered in allocating performance for this issue but will be considered in Section 8.3.4. The waste package will be considered as part of the repository system element equipment. The major system elements are further subdivided into more specific system elements; however, for resolving this issue, only the site need be divided further. The site is divided into two systems elements: the surface and the subsurface. In addition to these three system elements from the MGDS requirements, a fourth system element, offsite installation, is required for the resolution of this issue. A description of each of these system elements and their function with respect to this issue follows.

Surface system element. The surface system element affects transport of radionuclides between the repository and the members of the public in the unrestricted area during the preclosure period. Transport mechanisms include atmospheric transport, surface water movement and dilution, bioaccumulation, and consumption of agricultural and indigenous food stuffs.

Atmospheric transport is most likely the dominant mode of transport of radionuclides from the repository to the public. The main processes involved are the physical transport, dispersion, and deposition of potential releases of radionuclides. The atmosphere will impact the potential radiation dose from both the natural and man-made source terms. The radioactivity deposited will then move through the food chain to crops, animals, and man. A lesser contributor to the dose rate in the unrestricted area is direct radiation from the repository. The distance between the repository and the unrestricted area is expected to greatly attenuate the direct radiation. Direct radiation that can contribute to the dose in the unrestricted area has to be controlled to maintain a safe environment for the workers. A possible exception to this is direct radiation exposure of the public due to transportation, which is to be addressed by Issue 3.3.

Concentration of radionuclides in the unrestricted area is also affected by dispersion and transport of routine radioactive releases through water pathways, followed by uptake by crops, animals, and man.

The surface system element also provides a remote location (with respect to highly populated areas and members of the public) for the repository operations. This serves to limit the number of people in the adjacent unrestricted area. Part (a) of this issue requires that the average dose to members in a highly populated area be less than a small fraction of the allowable limit. Since the repository is far from a highly populated area, the doses to the population are expected to be small. Verification of this attribute is directly determined by investigating the local demographics (Section 8.3.1.10, population density and distribution program).

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Repository system element. The repository system element includes all surface and subsurface systems that can impact man-made sources of radiation in the unrestricted area. This includes all systems and operations that control radiation releases and exposures in the unrestricted area. The repository will be designed and analyzed to ensure that the radioactive effluents are below the regulatory limits. Potential effluents are in the form of gases, liquids, and solids, all of which must be evaluated for compliance with the applicable regulations. Analyses to determine compliance of the repository with the regulations will require information on the radioactive sources, systems design, and operations to be performed.

Offsite installations. The exposure standards in 40 CFR 191, Subpart A, apply to releases from the repository and from uranium fuel cycle facilities defined in 40 CFR 190. Therefore, a determination of which of the installations in the vicinity of the Yucca Mountain MGDS are nuclear fuel cycle facilities is required. The function that the offsite installations system element plays, with respect to this issue, is to verify that there are no uranium fuel cycle facilities in the vicinity of Yucca Mountain that would need to be considered when assessing total exposure to the population.

Allocation of performance to the system elements. The next four steps after the identification of functional requirements make up the bulk of the performance allocation process. In these steps, performance measures and performance goals are developed, and needed parameters are defined. The results of these steps may be seen in Tables 8.3.5.3-1 and 8.3.5.3-2.

Development of design criteria and constraints and identification of input items. The only constraints on the design of the repository forthcoming from this issue are those general performance goals shown in Tables 8.3.5.3-1 and 8.3.5.3-2. These performance goals are transmitted to Issue 2.7 (Section 8.3.2.3) where specific design criteria are developed and transmitted to Issue 4.4 (Section 8.3.2.5) for incorporation into the design of the repository. In general, specific design products or information required of either Issue 2.7 or Issue 4.4 and needed by the performance issues are also transmitted to Issue 2.7. However, at this time no specific design products or information items have been identified as being needed by this issue.

Public radiological safety assessment package. The specific analytical approach for resolution of this issue will be developed as part of the preclosure risk assessment methodology (PRAM) program described in Section 8.3.5.1 and other project activities. A general approach is shown in Figure 8.3.5.3-2 in the dashed box labeled "public radiological safety assessment package." The following provides a step-by-step discussion of the analytical approach.

Design Evaluation. The design package and site data are obtained from the reference information base (RIB), and the repository design features related to the radiological safety of the public during normal operations are evaluated. The following is a discussion of what types of information are investigated during this design evaluation. The high level waste (HLW) throughput (schedule and amount of waste received per year) is an important controlling factor in the design of the repository process and storage facilities (e.g., hot cell structure and lag storage). Direct radiation that

Table 8.3.5.3-1. Functions, performance measures, and performance goals for Issue 2.1 (public radiological exposures--normal conditions)

| System element | Function | Process or activity | Performance measure | Tentative goal | Needed confidence |
|---|---|---|---|---|---|
| Surface | Provide remoteness from highly populated areas and members of the public | Locate repository in a low population area | Population density | A. Population densities less than or equal to those required by the qualifying conditions of 10 CFR Part 960 | High |
| | Provide dispersion and transport of routine radioactive releases to the unrestricted area through water pathways, crops, and animals | Analyze dilution, transportation, bioaccumulation of radionuclides in rivers, streams, and food stuffs | Radionuclides concentrations in environmental media and individual doses | B. Dose limits of 40 CFR Part 191, Subpart A and 10 CFR Part 20 as applied to the contribution from radionuclides in food chain pathways | High |
| | Provide transport, dispersion, and diffusion of routine airborne radioactive effluents to the unrestricted area | Analyze atmospheric transport by wind and convection, including dispersion and diffusion | Radionuclides concentrations in environmental media and individual doses | C. Composite dose limits required by 40 CFR Part 191, Subpart A and 10 CFR Part 20 | High |
| Repository | Provide containment of potential sources of radiation to the unrestricted area | Limit releases of routine gaseous, particulate, and liquid radioactive effluents | Radionuclides concentrations in environmental media and individual doses | D. Composite dose limits required by 40 CFR Part 191, Subpart A and 10 CFR Part 20 as applied to routine releases from the repository | High |
| Offsite installations | Verify that there are no nuclear (uranium) fuel cycle facilities that need to be considered in assessing the public dose | Locate and analyze nearby nuclear (uranium) fuel cycle facilities | Number of nuclear (uranium) fuel cycle facilities requiring consideration in assessing the public dose | E. No nuclear (uranium) fuel cycle facilities requiring consideration in assessing the public dose | High |

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Table 8.3.5.3-2. Parameters required for Issue 2.1 (public radiological exposures--normal conditions) (page 1 of 4)

| Related performance goal[a] | Performance or design parameter | Parameter descriptor | Tentative parameter goal | Needed confidence | Expected parameter value | Current confidence | SCP section providing parameter |
|---|---|---|---|---|---|---|---|
| A | Distances from highly populated areas | 140 km radius | ≥5 km | High | About 130 km | Medium | 8.3.1.12, (b) |
| A | Population located in adjacent 1-mile by 1-mile area | Nye and Clark counties | <1,000 persons | High | No permanent population | Medium | (b) |
| A | Population density of the region | Nye and Clark counties | Low population density | High | Section 3.6.2 in environmental assessment (DOE, 1986b) | Medium | (b) |
| B | Bioaccumulation of radionuclides in terrestrial flora | 80 km radius | (c) | Medium | $1 \times 10^{-28}$ to $1 \times 10^{-14}$ Ci/kg (see footnote d) | Medium | (b) |
| B | Bioaccumulation of radionuclides in terrestrial fauna | 80 km radius | (c) | Medium | $1 \times 10^{-25}$ to $1 \times 10^{-15}$ Ci/kg (see footnote e) | Medium | (b) |
| B | Types of crops raised | 80 km radius | (c) | Medium | (f) | Medium | (b) |
| B | Amounts of crops raised | 80 km radius | (c) | Medium | $1 \times 10^{4}$ to $1 \times 10^{7}$ kg/yr (see footnote g) | Medium | (b) |
| B | Types of crops consumed | 80 km radius | (c) | Medium | (h) | Medium | (b) |
| B | Amounts of crops consumed | 80 km radius | (c) | Medium | $1 \times 10^{4}$ to $1 \times 10^{5}$ kg/yr | Medium | (b) |
| B | Types of animals raised | 80 km radius | (c) | Medium | (i) | Medium | (b) |

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Table 8.3.5.3-2. Parameters required for Issue 2.1 (public radiological exposures--normal conditions) (page 2 of 4)

| Related performance goal[a] | Performance or design parameter | Parameter descriptor | Tentative parameter goal | Needed confidence | Expected parameter value | Current confidence | SCP section providing parameter |
|---|---|---|---|---|---|---|---|
| B | Number of animals raised | 80 km radius | (c) | Medium | $1 \times 10^1$ to $1 \times 10^5$ kg/yr | Medium | (b) |
| B | Types of animals consumed | 80 km radius | (c) | Medium | (j) | Medium | (b) |
| B | Amounts of meat consumed | 80 km radius | (c) | Medium | $1 \times 10^4$ to $1 \times 10^6$ kg/yr | Medium | (b) |
| B | Animal consumption of forage | 80 km radius | (c) | Medium | $1 \times 10^1$ to $1 \times 10^4$ kg/yr | Medium | (b) |
| B | Forage storage time | 80 km radius | Goal is values given in Reg. Guide 1.109 (NRC, 1977a) | Medium | Data not available | Data not available | (b) |
| B | Grazing yield and period | 80 km radius | (c) | Medium | 75 to 100% of the year | High | (b) |
| B | Radius of crop and animal area | 80 km radius | (c) | Medium | 50 km to bulk of cropland and farms (W to SW) | High | (b) |
| B | Volumetric flow of surface water to water bodies | 80 km radius | Little or no surface runoff | Medium | Section 3.3.1 in environmental assessment (DOE, 1986b) | Medium | (b) |
| B | Population served by local drinking water | 80 km radius | (c) | Medium | $1 \times 10^2$ to $1 \times 10^4$ | Medium | (b) |
| B | Volumetric flow of local drinking water | 80 km radius | (c) | Low | Section 3.3.1 in environmental assessment (DOE, 1986b) | Medium | (b) |

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Table 8.3.5.3-2. Parameters required for Issue 2.1 (public radiological exposures--normal conditions) (page 3 of 4)

| Related performance goal[a] | Performance or design parameter | Parameter descriptor | Tentative parameter goal | Needed confidence | Expected parameter value | Current confidence | SCP section providing parameter |
|---|---|---|---|---|---|---|---|
| B | Recreational uses of water bodies | 80 km radius | Very little recreational use of water | High | (k) | (k) | (b) |
| C,E | Wind speeds | 80 km radius | (c) | High | Figures 5-3 to 5-7, and Tables 5-6 and 5-7 | Medium | 8.3.1.12 |
| C,E | Wind direction | 80 km radius | (c) | High | Figures 5-3 to 5-7, and Tables 5-6 and 5-7 | Medium | 8.3.1.12 |
| C,E | Atmospheric stability | 80 km radius | (c) | Medium (See footnote l) | Table 5-11 | Medium | 8.3.1.12 |
| C,E | Mixing layer depth | 80 km radius | (c) | Medium | (m) | Medium | 8.3.1.12 |
| C,E | Average ambient temperature | 80 km radius | (c) | Medium | Tables 5-2 and 5-3 | Medium | 8.3.1.12 |
| C,E | Atmospheric moisture | 80 km radius | (c) | Medium | Tables 5-2 and 5-5 | Medium | 8.3.1.12 |
| C,E | Precipitation: type, amount, intensity, etc. | 80 km radius | (c) | Medium | Tables 5-2 and 5-4 | Medium | 8.3.1.12 |
| C,E | Barometric pressure | 80 km radius | (c) | Medium | Table 5-2 | Medium | 8.3.1.12 |
| C,E | Size and distance of topographic features from release points | 80 km radius | Topographic features beneficial to dispersion | Medium | See U.S Geological Survey (USGS) topographic maps | High | Literature |



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Table 8.3.5.3-2. Parameters required for Issue 2.1 (public radiological exposures--normal conditions) (page 4 of 4)

| Related performance goal[a] | Performance or design parameter | Parameter descriptor | Tentative parameter goal | Needed confidence | Expected parameter value | Current confidence | SCP section providing parameter |
|---|---|---|---|---|---|---|---|
| D | Radon emanation rate from tuff | (TSw2 unit)[n] | (c) | High | 0.48 $pCi/m^2$-s | Low | 8.3.1.15 |
| D | Reference repository design and supporting analyses | | No additional site characterization data needed--see footnote o. | | | | |
| E | Location of nearby uranium fuel cycle facilities | 80 km radius | No nearby nuclear fuel cycle facilities | High | No nearby nuclear fuel cycle facilities | Medium | 8.3.1.13 |
| E | Doses from nearby uranium fuel cycle facilities | 80 km radius | Doses less than 40 CFR 191 limits | High | Doses less than 40 CFR 191 limits | Medium | 8.3.1.13 |

[a]The letters in this column key the performance parameters in this table to the tentative performance goals in Table 8.3.5.3-1.
[b]Collection of these data is part of the environmental program planned activities and is addressed in the Radiological Monitoring Plan discussed in Section 8.3.1.13.
[c]Tentative goal is to have further measurements of this parameter verify the range of expected values listed here.
[d]This range covers all flora for which data are now available; specific values are flora and radionuclide specific.
[e]This range covers all fauna for which data are now available; specific values are fauna and radionuclide specific.
[f]Wheat/grains, corn, apples, potatoes, alfalfa, alfalfa seed, hay, silage, peppers, melons, berries, pecans, leafy vegetables, and honey.
[g]Specific values depend on available crops, crop areas, and crop densities.
[h]Includes all crops listed footnote f except alfalfa, hay, and silage.
[i]Beef cattle, dairy cattle, goats, hogs, sheep, and poultry.
[j]All of those in footnote i plus quail, freshwater fish, ducks, geese, rabbit, deer.
[k]Very limited use of Crystal Reservoir; swimming pool data not yet available.
[l]Medium confidence requirements are intended to indicate that these parameters need to be site-specific.
[m]See Quiring (1968).
[n]TSw2 unit is the nonlithophysal Topopah Spring unit (repository horizon).
[o]For communicating the design information needed to evaluate worker radiological safety under normal conditions, the input items from Issue 4.4 (obtained through Issue 2.7) are collectively listed as a parameter.



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can be emitted from the central process area and the amount of routine radioactive effluents will be directly related to the amount of HLW on hand and being processed. These sources of potential doses to the public also depend on how processes are conducted for such activities as waste receipt, lag storage, waste handling and consolidation, transport of waste containers and the heat treatment of spent fuel, if done. Public radiation doses from such activities will also be controlled by administrative procedures (e.g., limits on frequency of tasks and time in storage). Attributes of the repository design that will play a major role in controlling direct radiation and release of radioactive effluents to the unrestricted area include such features as

1. Barrier and shield thicknesses, composition, and distance from the source, and the exposed individuals.

2. Containment and ventilation system characteristics (e.g., repository and hot cell layout, differential pressures between areas, openings, air locks, and filters).

3. Containment characteristics of the waste form (i.e., fuel elements, waste package, etc.)

4. Radioactive material release point characteristics (e.g., stack height, diameter, exit velocity, temperature, and distance from unrestricted area).

In addition, as part of the regulatory performance verification requirements, specific systems and operational controls will be needed to verify that the repository design and operation will maintain the annual radiation dose to the public to less than the regulatory limits. Types of systems that must be provided include (1) gaseous, particulate, and liquid effluent monitoring and control equipment; (2) effluent sampling and measuring equipment; (3) environmental surveillance equipment; and (4) emergency response features. Design of these systems will be incorporated in the normal repository design process. The information needed for this design evaluation will be the product of the design process and will generally not depend directly on the site characterization activities. However, data on background radiological conditions and dust characteristics may affect the design of monitoring equipment. Data on dust characteristics are discussed in Section 8.3.2.4 (nonradiological health and safety) and only mentioned here because worker health concerns require more extensive data on dust.

Identification of radiation source characteristics. Potential sources of radiation that can contribute to the dose to the exposed individuals in the unrestricted area can be categorized as resulting from (1) repository operations, (2) operation of offsite facilities, and (3) miscellaneous operations. Examples of radiation sources resulting from repository operations are receipt of HLW shipping casks, releases during spent fuel consolidation, transport of HLW containers, and naturally occurring radionuclides (e.g., releases from ventilation exhausts and the muck pile).

The specific information needed about the potential source terms includes radionuclides involved, quantity and concentration, decay radiation and energies, and physical and chemical forms. General information needed about the source terms for dose evaluation include

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1. Planned repository operational details (e.g., scheduled HLW throughput and inventories, generated low-level waste (LLW) and transport rates, and normal effluent release rates).

2. Repository design features (e.g., radionuclide barriers, normal effluent release locations, layout distances, containment, leakage, and filtration details).

3. Environmental details (e.g., pathways for transport or dispersion of radioactive materials through the soil, air, and water to vegetation, animals, and the public, and location of other relevant offsite facilities and their radionuclide release rates).

4. Natural radionuclide sources (e.g., radon emanation rate).

Depending on the characteristics of the source terms, the information needs will be satisfied by either the site characterization program, the repository design process, or the environmental and socioeconomic sampling and monitoring programs. Development of the analytical tools needed to evaluate potential adverse public impacts of the source terms will be coordinated with the preclosure risk assessment methodology (PRAM) program requirements and recommendations.

Radionuclide transport evaluation. The next element in the public radiological safety assessment package is radionuclide transport evaluation following release to the environment of radioactive material from normal repository activities. Radioactive releases to the environment from relevant offsite facilities must also be considered since these releases this can contribute to the dose to the public in the repository unrestricted area.

The pathways for the initial concentration of radionuclides released from the repository central process area and offsite facilities to the public in the unrestricted area need to be described. The possible pathways to the public can be directly through the air, water, and soil, or indirectly through vegetation and animals.

The dispersion of airborne radioactive materials can (1) result in radionuclide concentrations in the air that can cause an external dose by direct radiation or an internal dose through inhalation, or (2) result in ground deposition of radioactive material. Similarly, dispersion of waterborne radioactive effluents can result in an external dose by direct radiation, result in an internal dose through drinking of the water, or result in the deposition of radioactive material. Radionuclides deposited on the ground, plants, or riverbanks can cause a direct radiation dose but, more importantly, they can enter the food chain through uptake and bioaccumulation in plants and animals. Examples would be eating cattle that grazed on local grass or eating grain irrigated with local water.

Analytical tools in the form of dispersion and pathway models will be required to perform the radionuclide transport evaluation. Meteorological data (e.g., wind speed and direction atmospheric stability) will be needed as input to the dispersion model. This need for site data will be satisfied by the site characterization program. Specific data (e.g., type of crops raised and bioaccumulation of radionuclides in plants and animals) will be required

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for the food chain pathway models. This data need will be addressed by the socioeconomic and environmental monitoring program.

Public radiation exposure calculation. The last step in the analysis is the evaluation of radiological exposure that quantifies the maximum dose to the public postulated from routine operation of the repository and offsite facilities.

The maximum dose to an individual at the nearest unrestricted location is normally considered the greatest potential adverse impact and is used as the basis for calculations. The furthest distance the unrestricted area can be from the repository is 5 km. The Bureau of Land Management limits occupancy at this location. Occupancy at a site about 15 km away from the repository will be assumed to be 24 hours per day, 365 days per year. Individuals are conservatively assumed to do such things as drink local water, eat local animals and fish, eat foodstuffs grown using local water, and spend recreational time in local water bodies. Analytical models will be used to quantify the public dose. The following types of analytical tools will be needed:

1. Building ventilation, filtration, and leakage models.
2. Radiation shielding models.
3. Atmospheric dispersion models.
4. Radiological impact models for transportation of LLW.
5. Food chain pathways models.
6. Radiological consequence assessment models.

The information needed to calculate doses using these analytical tools will be provided as discussed in the previous steps. This information will be the product of the site characterization program, the socioeconomic and environmental monitoring program, and the normal repository design process. Following is a list of some technical guidance documents that will be evaluated for applicability to the development of the above analytical tools. A list of analytical tools that are available for use is contained in Section 8.3.5.19 (completed analytical techniques). Further discussions of analytical tools are contained in Sections 8.3.5.20 (techniques requiring development).

1. Regulatory Guide 1.21--Measuring, Evaluating, and Reporting Radioactivity in Solid Waste and Release of Radioactivity in Liquid and Gaseous Effluents From Light-Water-Cooled Nuclear Power Plants (Revision 1, June 1974) (AEC, 1974).

2. Regulatory Guide 1.23--Onsite Meteorological Programs (NRC, 1980).

3. Regulatory Guide 1.109--Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purposes of Evaluating Compliance With 10 CFR 50, Appendix I (Revision 1, October 1977) (NRC, 1977a).

4. Regulatory Guide 1.111--Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases From Light-Water-Cooled Reactors (Revision 1, July 1977) (NRC, 1977c).

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5. Regulatory Guide 1.112--Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents From Light-Water-Cooled Power Reactors (Revision O-R, May 1977) (NRC, 1976b).

6. Regulatory Guide 1.113--Estimating Aquatic Dispersion of Effluents From Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I (Revision 1, April 1977) (NRC, 1977b).

7. Safety Series No. 60--Criteria for Underground Disposal of Solid Radioactive Waste (IAEA, 1983b).

8. Safety Series No. 68--Performance Assessment for Underground Radioactive Waste Disposal Systems (IAEA, 1985).

9. DOE/EP-0023--A Guide for Environmental Radiological Surveillance at U.S. Department of Energy Installations (July 1981) (Corley and Denham, 1981).

10. DOE/EP-0096--A Guide for Effluent Radiological Measurements at DOE Installations (April 1982) (Corley and Corbit, 1983).

Performance evaluation for compliance with goals. The remainder of Figure 8.3.5.3-2 deals with the final evaluation of the results documented in the public radiological safety assessment package. The results are compared with the regulatory limits contained in the regulations listed earlier in the section called "Regulatory basis for addressing this issue." If all the limits are met, then the results are examined to see if the ALARA criterion has been met. If both the regulatory limits and the ALARA criterion have been met and if the design is in the final design phase, then the design is ready for license application and a favorable issue resolution has been achieved. If both the regulatory limits and the ALARA criterion have been met but the design is not in the final design phase, then this process is repeated for the next design phase.

If the results of the public radiological safety assessment package do not meet either the regulatory limits or the ALARA criterion, then design, procedural, or operational changes are recommended to correct the situation. If these changes cannot be made and the performance goals cannot be reasonably changed, then an unfavorable resolution of the issue has occurred. However, if the design, procedural, or operational changes can be made or the performance goals can be reasonably changed, then the recommended changes are implemented and the whole process is repeated.

Interrelationships of information needs

The questions asked by this issue address the radiological health and safety of the public. The basic question is will the expected doses to the public be within the regulatory limits contained in 10 CFR Part 20 and 40 CFR Part 191 Subpart A? The resolution of this issue can be obtained by answering three other questions. These questions are as follows:

1. What site and design information is required to predict the expected radiation doses to the public from the normal operation of the repository and nearby uranium fuel cycle facilities?

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2. What are the projected releases of radioactive material from the normal operations of repository and nearby uranium fuel cycle facilities that could be transported to the unrestricted area and cause radiation doses to the public?

3. Are the combined radiation doses to the public resulting from the projected releases of radioactive material from the normal operations of the repository and nearby uranium fuel cycle facilities within applicable limits?

These questions have been designated as information needs. Questions 1, 2, and 3 are Information Needs 2.1.1, 2.1.2, and 2.1.3, respectively. All site data required to perform the dose calculations and assessments are collected under Information Need 2.1.1. Information Needs 2.1.2 and 2.1.3 use the data called for by Information Need 2.1.1 to perform the release determinations, radionuclide transport calculations, and public dose assessment, but do not collect any site data on their own. For this reason, only Information Need 2.1.1 is discussed in this report. The functions and performance measures (associated with the MGDS system elements) necessary for answering these two questions and resolving issue are listed in Table 8.3.5.3-1. The site data needed to answer these two questions are listed in Table 8.3.5.3-2. Information Needs 2.1.2 and 2.1.3 (together with Issue 2.1 in its entirety) will be discussed in the repository design plan (RDP). The RDP will be published approximately one year after publication of the SCP.

Information Need 2.1.1 (Section 8.3.5.3.1) describes the site and design information required to resolve this issue. The detailed site data needed is shown in Table 8.3.5.3-2, along with an indication of the confidence with which the information must be known. The design information required is not listed in any detail at this point. It is sufficient to say that the repository reference design and supporting analyses will be required.

Information Need 2.1.2 is a determination of the expected releases of radioactive materials from the repository during normal operations. Included in this information need are the releases of radioactive materials from nearby uranium fuel cycle facilities. Releases from the repository will be determined from the reference repository design and supporting analyses. A brief discussion of some of the processes is presented previously under design evaluation. Information on releases from nearby uranium fuel cycle facilities will be collected as part of site characterization and a determination of the expected releases from these facilities will be performed as part of this information need.

Information Need 2.1.3 is a determination of whether predicted doses to the public resulting from the expected releases of radioactive materials are within applicable limits or a small fraction of those limits. As described earlier in the section called public radiologic safety assessment package, the doses to the public are predicted using radionuclide transport and dispersion models to estimate the amounts of radionuclides that eventually reach the public. The final resolution of this issue will take place under this information need when the results of the dose calculations are evaluated and compared with the regulatory limits contained in 10 CFR Part 20 and 40 CFR Part 191 Subpart A.

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### 8.3.5.3.1 Information Need 2.1.1: Site and design information needed to assess preclosure radiological safety

Technical basis for addressing the information need

Link to the technical data chapters and applicable support documents

Chapter 3 discusses the present state of the knowledge on the site hydrology, including uses of surface water and ground water. Chapter 4 discusses the water chemistry of the site. Section 4.1.2.6 (background radioactivity (of repository ground water)), contains a discussion on what is known about the radionuclide content of repository ground water to date. Chapter 5 discusses the present state of the knowledge on the meteorology of the site and surrounding region. Further discussions on the subject of radiological protection of the public may be found in Sections 6.1.1.4.1 (radiological protection design requirements) and 6.4.4 (Issue 2.1: radiological exposure expected to public). Section 8.3.5.1 discusses the preclosure risk assessment methodology (PRAM) program, which includes radiological risk to the public during normal operations as part of its scope. Sections 2.5 (radiological protection) and 6.1 (radioactive releases during normal operations) of the site characterization plan-conceptual design report (SCP-CDR) (SNL, 1987) also contain discussions relevant to this issue. Section 6.1 of the SCP-CDR is especially informative because it contains some preliminary estimates of expected releases during normal operations of the repository.

Parameters

The parameters required by this information need are those site and design parameters relevant to the determination that the expected doses to the public are within applicable limits. Design information required for this purpose is listed in Table 8.3.5.3-2 simply as the reference repository design and supporting analyses. Reference repository design information and supporting analyses will be obtained from the reference information base (RIB) and will contain all design details necessary to perform the dose calculations to resolve this issue.

The site data required to resolve this issue are obtained through various site characterization programs. Following is a summary table of the required site data and the SCP section providing the information.

| Data requirement | SCP section |
|---|---|
| POPULATION DENSITY DATA | |
| Distance of the repository from highly populated areas | (a) |
| Population located in adjacent 1-mile by 1-mile area | (a) |
| Population density of the region around the repository | (a) |

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| Data requirement | SCP section |
|---|---|
| AGRICULTURAL DATA | |
| Bioaccumulation of radionuclides in the terrestrial flora | (a) |
| Bioaccumulation of radionuclides in the terrestrial fauna | (a) |
| Types and amounts of crops raised | (a) |
| Types and amounts of crops consumed | (a) |
| Types and amounts of animals raised | (a) |
| Types and amounts of meat consumed | (a) |
| Animal consumption of forage | (a) |
| Forage storage time | (a) |
| Grazing yield and period | (a) |
| Radius of the crop and animal area | (a) |
| SURFACE-WATER DATA | |
| Volumetric flow of surface water to water bodies | (a) |
| Population served and the volumetric flow of drinking water from affected water bodies | (a) |
| Recreational uses of area water bodies | (a) |
| METEOROLOGICAL DATA | |
| Wind speeds in the region | 8.3.1.12.1 |
| Prevailing wind directions | 8.3.1.12.2 |
| Atmospheric stability of the area | 8.3.1.12.2 |
| Atmospheric mixing layer depth of the region | 8.3.1.12.2 |
| Average ambient temperature of the area | 8.3.1.12.2 |
| Atmospheric moisture of the area | 8.3.1.12.2 |

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| Data requirement | SCP section |
|---|---|
| Area precipitation, including type, amount, intensity, etc. | 8.3.1.12.2 |
| Size and distance of major topographic features from release points | Existing data should be adequate |
| REPOSITORY ROCK DATA | |
| Radon emanation rate from the tuff | 8.3.1.15.1.6.2 |
| OFFSITE INSTALLATION DATA | |
| Location of nearby uranium fuel cycle facilities | 8.3.1.13.1.2 |
| Liquid, particulate, and gaseous radionuclide releases from nearby uranium fuel cycle facilities | 8.3.1.13.1.3 |
| Meteorological data for nearby uranium fuel cycle facilities | 8.3.1.12.1, 8.3.1.12.2 |

[a]Collection of these data is part of the environmental program planned activities and is addressed in the Radiological Monitoring Plan discussed in Section 8.3.1.13.

This Table summarizes information listed in Table 8.3.5.3-2, which was also discussed earlier.

As shown in Table 8.3.5.3-2, these parameters are needed with differing levels of confidence and for different locations on and around the site.

Logic

The assessment of the preclosure radiological safety of the public under normal repository conditions requires a thorough understanding of the repository design and operating procedures. This information is obtained from the repository reference design and supporting analyses. The radiation source terms can be developed from the design, the repository rock and water data, and the offsite installation data. After developing the source terms, calculations of radionuclide transport through the atmosphere and other environmental pathways are performed. These calculations require the agricultural and meteorological data. Finally, to assess the doses to the public, the population density data are needed. A more detailed discussion of the dose

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assessment process is presented earlier in the section called "public radiological safety assessment package."

The activities described here are related to all of this issue and not just to Information Need 2.1.1. Three distinct activities are planned under this information need during site characterization in support of performance analyses for public radiological safety. The first activity concerns the refinement of site parameters needs for this issue. The second activity deals with the development of methods to perform evaluations of public radiological safety and is connected with the PRAM program. The third activity is a performance assessment of public radiological safety for the advanced conceptual design (ACD).

### 8.3.5.3.1.1 Performance Assessment Activity 2.1.1.1: Refinement of site data parameters required for Issue 2.1

Objectives

The objective of this activity is to refine the list of site-data parameters presented earlier in this section in Table 8.3.5.3-2. This list may be incomplete or the level of confidence required may be inappropriate.

Parameters

The list of parameters presented in Table 8.3.5.3-2 is the starting point for this activity. As the activity progresses and matures, parameters may be added to or deleted from this list.

Description

There are three ways in which the parameter list will be refined. First, during the course of site-characterization reviews and activities by those organizations specified to collect data will discover problems with parameter lists. These problems will be resolved and parameter lists will be revised. Second, the PRAM program will be developing methods for radiological performance analyses (Performance Assessment Activity 2.1.1.2, Section 8.3.5.3.1.2). During the development of these methods, lists of required parameters for each type of analysis are expected to be created. A review of the parameter list resulting from PRAM methods development activities may result in refinement of the Issue 2.1 parameter list. Finally, a performance assessment of the ACD and license application design (LAD) for public radiological safety may uncover deficiencies in the current parameter list. This is an ongoing activity whose end date is the completion of the license application.

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8.3.5.3.1.2 Performance Assessment Activity 2.1.1.2: Development of performance assessment activities through the preclosure risk assessment methodology program

Objectives

The objective of this activity is to benefit from the PRAM program performance assessment methods development efforts. The Yucca Mountain Project will participate in the PRAM program and will adapt PRAM program to the Yucca Mountain program. A secondary objective of this activity is to use the information developed in this activity to assist in refining the site data parameters list for this issue (Performance Assessment Activity 2.1.1.1).

Parameters

There are presently no parameters for this activity; however, a list of parameters may result from the PRAM program development.

Description

A part of PRAM will be concerned with the assessment of public radiological safety during the normal operations of a repository. The Yucca Mountain Project will participate in this program and assist in the development of the overall methodology. Methods developed in the PRAM program will be adapted for use in the Yucca Mountain Project assessment of public radiological safety during the normal operations of the Yucca Mountain repository (Performance Assessment Activity 2.1.1.3). Since the PRAM program is expected to continue through license application design, this activity will be ongoing through license application. A more detailed discussion of the PRAM program is presented in Section 8.3.5.1.

8.3.5.3.1.3 Performance Assessment Activity 2.1.1.3: Advanced conceptual design assessment of the public radiological safety during the normal operations of the Yucca Mountain repository

Objectives

The objective of this activity is to perform a public radiological safety assessment of the Yucca Mountain repository advanced conceptual design. Secondary objectives of this activity are to provide information for the refinement of the site data parameter list for Issue 2.1 (Performance Assessment Activity 2.1.1.1) and to provide feedback to the PRAM program for future methods development activities (Performance Assessment Activity 2.1.1.2).

Parameters

The parameters necessary for this activity are those listed in the site data parameter list for Issue 2.1 presented in Table 8.3.5.3-2.

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Description

This activity will assess the Yucca Mountain repository advanced conceptual design for public radiological safety during normal operations. A general description of the process presented earlier in this section under "public radiological safety assessment package."

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## 8.3.5.4 Issue resolution strategy for Issue 2.2: Can the repository be designed, constructed, operated, closed, and decommissioned in a manner that ensures the radiological safety of workers under normal operations as required by 10 CFR 60.111, and 10 CFR Part 20?

This performance issue addresses the radiological safety of workers during normal operations. To resolve this issue, the mined geologic disposal system (MGDS) at Yucca Mountain will be designed to limit the normal radiation doses to workers during construction, operation, closure, and decommissioning of the repository to less than the limits specified in 10 CFR Part 20. The design process will be an iterative process as the design proceeds through the various phases. Design criteria and assumptions will be needed for both repository system operation and worker radiation safety. Many of the same parameters will apply to both areas and require appropriate input from design development. Further, the regulatory requirement to maintain radiation doses as low as reasonably achievable (ALARA) imposes additional iterations on the design to implement the differential cost-benefit analyses for the ALARA process. In these iterative design activities, DOE and other guidelines will be used in designing for repository worker radiation safety. Administrative procedures will be required to limit personnel exposure (e.g., personnel monitoring, limited access, and operational changes) for any operational activities for which design features are not able to preclude the possibility of dose rates to personnel above the guidelines.

The relationship of this issue with the other issues of the issues hierarchy is discussed in Section 8.3.2.1. That section discusses the relationship between design and performance issues and fixes the lines of communication between these issues. To be more specific about the relationship of this issue to the other issues with which it has direct or very strong ties, only Issues 2.1 (Section 8.3.5.3), 2.2 (this issue), 2.3 (Section 8.3.5.5), 2.7 (Section 8.3.2.3), and 4.4 (Section 8.3.2.5) are illustrated in Figure 8.3.5.4-1. The figure defines the ties between these issues by indicating the major information items passed between them. The figure also illustrates the connection of all these issues with the site characterization program. The scope of an issue is indicated by its size with respect to the other issues in the figure. Note that Issue 4.4 is the largest in scope, and the other issues, including this issue, branch out from Issue 4.4, reducing the scope to more specific areas.

In the discussion that follows in this section, the regulatory basis for addressing this issue is presented, the approach to resolving this issue is described, and the interrelationships among the information needs are discussed.

### Regulatory basis for the issue

While the issue refers to both 10 CFR 60.111(a) and 10 CFR Part 20, 10 CFR 60.111(a) simply refers to 10 CFR Part 20 and 40 CFR Part 191 Subpart A. Because 40 CFR Part 191 Subpart A is only applicable to members of the public, 10 CFR Part 20 (standards for protection against radiation) is the only regulation directly relevant to this issue. In addition, there are other sections of 10 CFR Part 60 that either require conformance with 10 CFR Part 20 or for which compliance with 10 CFR Part 20 is relevant. These include the following:

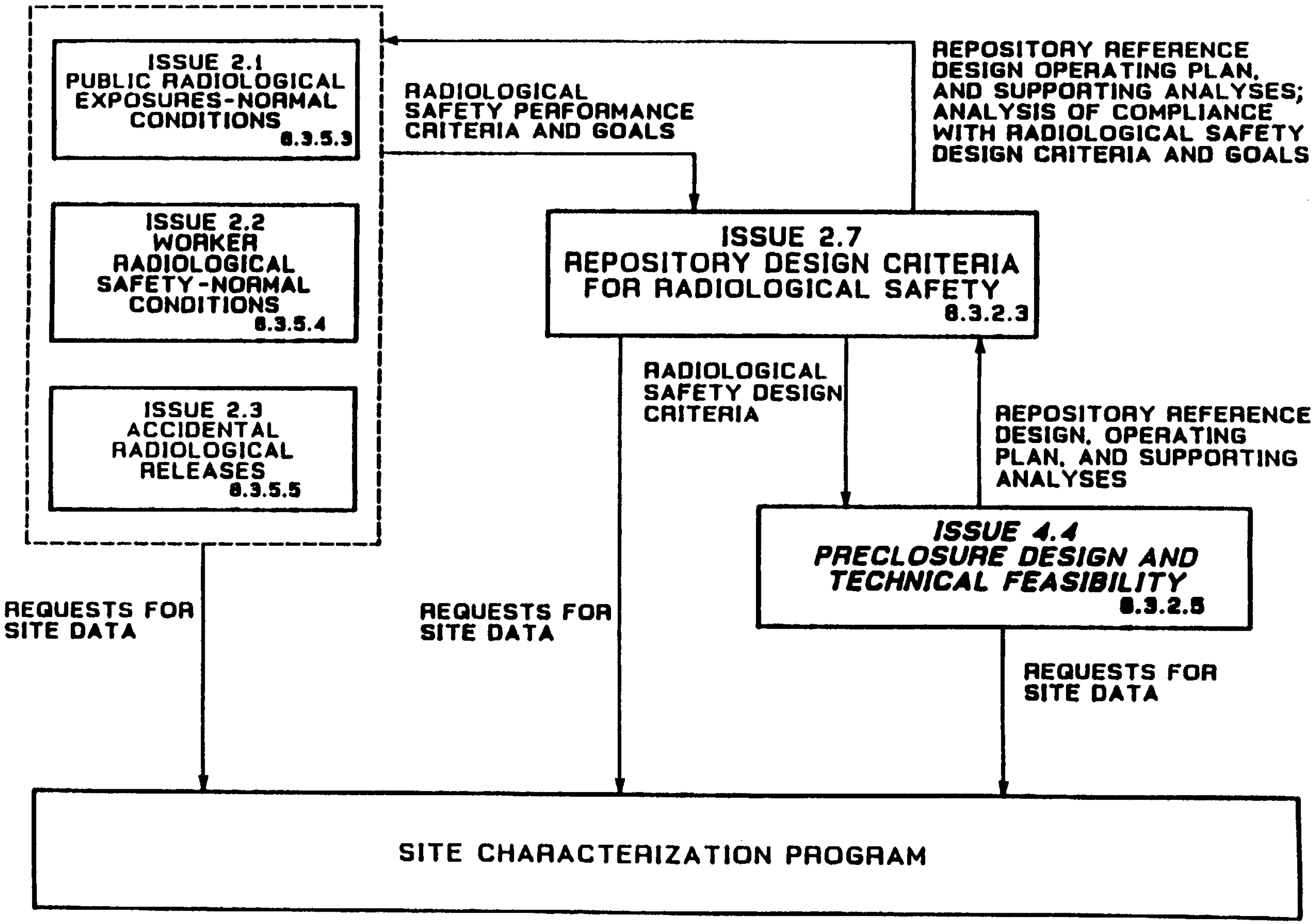


**Figure 8.3.5.4-1.** Relationship of Issue 2.2 (worker radiological safety—normal conditions) and site characterization programs.



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1. 10 CFR 60.21(c)(7), which requires description of the program to maintain effluents and occupational exposures in accordance with 10 CFR Part 20.

2. 10 CFR 60.131, which requires the design to meet the radiation protection requirements of 10 CFR Part 20.

3. 10 CFR 60.132, which requires the design to provide effluent control and monitoring in accordance with 10 CFR 60.111(a), which in turn invokes 10 CFR Part 20.

4. 10 CFR 60.133, which requires the underground ventilation system to maintain radionuclide concentrations and releases in accordance with 10 CFR 60.111(a) (which invokes 10 CFR Part 20).

Detailed discussions of these sections of 10 CFR Part 60 can be found with the issue resolution strategies for Issue 2.7 (repository design criteria for radiological safety, Section 8.3.2.3) and Issue 2.6 (preclosure waste package characteristics, Section 8.3.4.3). Additional guidance that will be evaluated for relevance to this issue includes the following:

1. Regulatory Guide 8.10--Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as is Reasonably Achievable (NRC, 1975).

2. Regulatory Guide 8.12--Criticality Accident Alarm System (NRC, 1981a).

3. Regulatory Guide 8.15--Acceptable Programs for Respiratory Protection (October 1976) (NRC, 1976a).

4. DOE Order 5480.11, Chapter 11--Radiation Protection Requirements (September 28, 1986) (DOE, 1985c).

5. DOE Order 6430.1--General Design Criteria Manual (December 1983) (DOE, 1983a).

6. 3 CFR--Radiation Protection Guidance to Federal Agencies for Occupational Exposure (Recommendations Approved by the President). (3 CFR, 1987).

7. ICRP 26 and 30--Recommendations of the International Commission on Radiation Protection (ICRP, 1977; 1978).

8. NUREG/CR 3254--Licensee Programs for Maintaining Occupational Exposure to Radiation ALARA (Munson, 1983).

9. DOE/EV/1830-T5--A Guide to Reducing Radiation Exposure to As Low As Reasonably Achievable (ALARA) (Kathren et al., 1980).

10 CFR Part 20 specifies the regulatory requirements for control of occupational radiation exposure. The concept and application of ALARA also applies to worker radiation exposure. In addition to the requirements that

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worker doses be maintained less than regulatory limits and conform to an ALARA philosophy, design guidelines are generally established at a fraction of the limits to ensure that necessary operations can be performed and occupational doses maintained below allowable limits. The establishment of design criteria for radiological safety is performed under Issue 2.7, which uses the performance criteria established in this issue to develop the design criteria.

10 CFR Part 20 and Part 60 also require a performance verification program during repository operations that ensures area radiation levels, airborne activity concentrations, contamination levels, and criticality controls are known and routinely verified. These operational requirements necessitate including systems to perform the verification of the design and operation of the facility. To ensure that the occupational radiation doses from the operation of the repository are less than the allowable levels, regulatory requirements must be known, both by designers to produce a design, and by evaluators to ensure that requirements are met.

The 10 CFR Part 20 and Part 60 requirements for verification of radiological performance necessitate special considerations for radiation measuring and monitoring systems. These requirements include "each licensee shall make or cause to be made such surveys as (1) may be necessary for the licensee to comply with the regulations in this part, and (2) are reasonable under the circumstances to evaluate the extent of radiation hazards that may be present," "means to monitor and control the dispersal of radioactive contamination," "a radiation alarm system to warn of significant increases in radiation levels, concentrations of radioactive material in air, and of increased radioactivity in effluents," and "the effluent monitoring systems shall be designed to measure the amount and concentration of radionuclides in any effluent with sufficient precision to determine whether releases conform to the design requirements for effluent control." Radiological measurement and monitoring systems that will be required for performance verification include air monitoring systems, criticality monitoring systems, gaseous effluent monitoring and sampling systems, liquid effluent monitoring and sampling systems, and personnel monitoring systems. The criteria for the testing, operation, and performance of these systems are found in documents issued by the various organizations and government agencies setting the standards.

In addition to complying with 10 CFR Part 20, the DOE has voluntarily agreed to comply with the radon monitoring and control provisions established by the Mine Safety and Health Administration in 30 CFR Part 57. To ensure adequate protection of repository workers, the contribution of radon and its daughter products to occupational exposure will be considered in assessing compliance with the applicable standards of 10 CFR Part 20.

Approach to resolving the issue

Licensing strategy overview

The repository will be designed to limit the expected radiation doses to workers during construction, operation, and closure as low as reasonably achievable (ALARA) below allowable limits required by 10 CFR Part 20. To ensure that the occupational exposure limits are met, design guidelines in

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the form of performance goals will be specified in this issue and transmitted to Issue 2.7, where radiological safety design criteria will be developed based on these design guidelines. The design criteria will specify dose rates in normally occupied areas and annual individual dose limits from penetrating radiation. The design criteria will also specify airborne radioactivity concentration limits in normally occupied areas. For some operational activities, design features may not be able to preclude the possibility of dose rates to personnel above the guidelines. In these instances, administrative procedures will be required to limit personnel exposure.

The personnel exposure performance verification systems, which will be designed and constructed to comply with 10 CFR Part 20 and Part 60 requirements, will be used during operations to ensure that the as-built repository systems will meet regulatory dose limits. Performance verification monitoring will provide the mechanism for corrective actions, either operational or design, and will ensure successful compliance. The provisions of the performance verification process significantly enhance the probability of successfully resolving this issue.

Resolution of this issue will occur when assurance is established that the repository can be designed, constructed, operated, closed, and decommissioned in a manner that provides for the radiological safety of workers under normal operations. This will be done by detailed analysis of the design and quantification of expected worker doses.

This strategy is not based on prior numerical evaluations of worker exposure since the actual operations of the repository are only conceptual. However, since there is currently considerable design flexibility available in terms of remote operations, shielding, restricted access procedural controls, etc., and since more significant operations already exist within the nuclear industry, it is expected that the radiation limits of the regulations can be met.

Application of the issue resolution strategy

The logic to be used in the resolution of this issue is illustrated in the logic diagram shown in Figure 8.3.5.4-2a and 8.3.5.4-2b. This logic diagram depicts how the generic issue resolution strategy of Section 8.2.2 is to be applied to this issue. The first step of the process, identifying regulatory requirements, has already been discussed in the section entitled "regulatory basis for the issue." The following discussions will explain each of the remaining steps in the resolution of this issue as shown in the logic diagram.

<u>Identification of functional requirements</u>. To allocate performance in this issue to specific system elements of the mined geologic disposal system (MGDS) at Yucca Mountain, the functions of these system elements with respect to this issue and to the radiological safety of the repository workers must be identified. The preclosure portion of the MGDS is divided into three major system elements: the site, the repository, and the waste package. The waste package will not be considered by itself in allocating performance for this issue but will be considered in Section 8.3.4. The waste package will be considered as part of the repository system element equipment. The major system elements are further subdivided into more specific system elements;

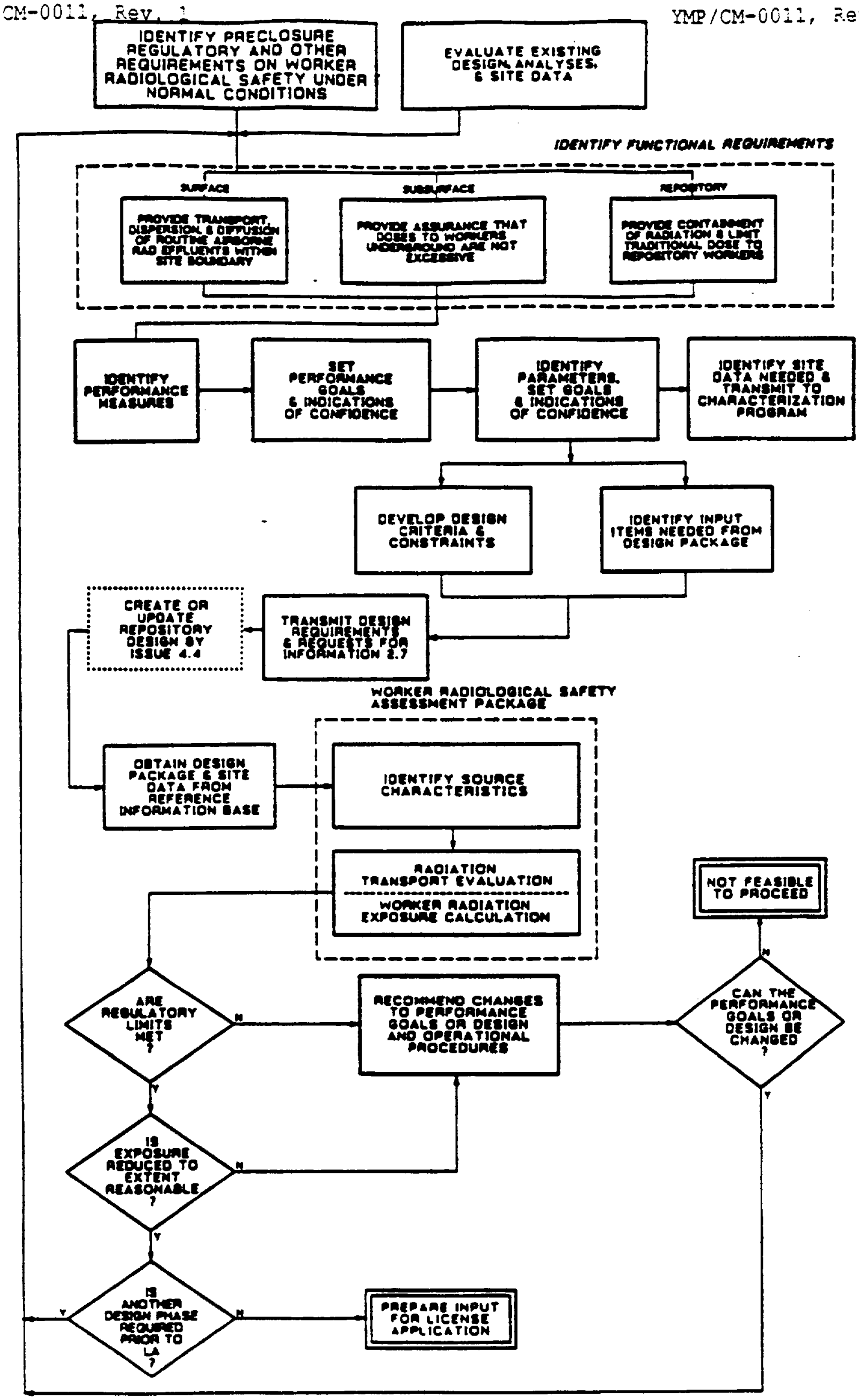


**Figure 8.3.5.4-2a.** Logic diagram for Issue 2.2 (worker radiological safety—normal conditions). See Figure 8.3.5.4-2b for legend. Section 8.3.2.1 describes the relationships and interfaces between design and performance issues.

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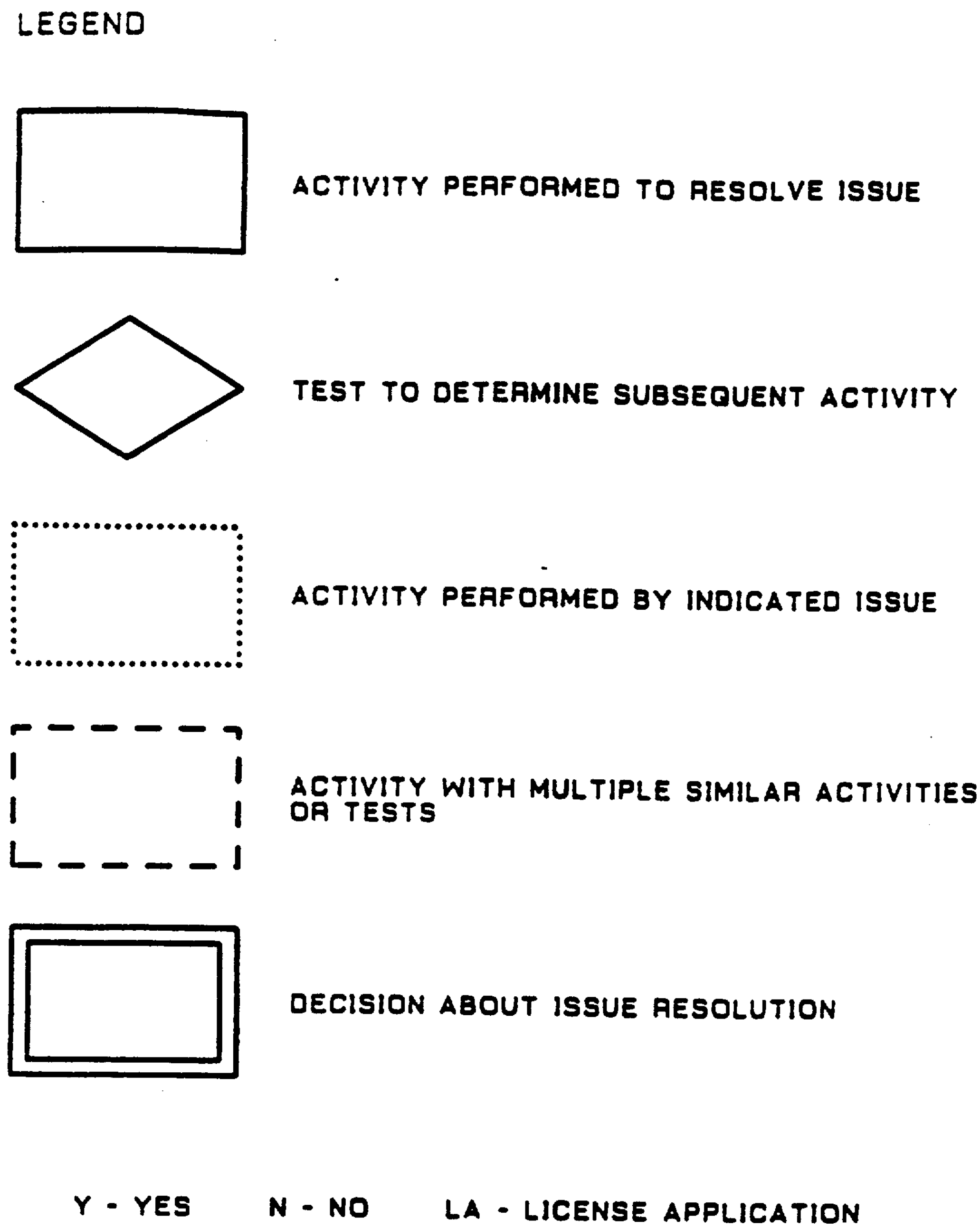


Figure 8.3.5.4-2b. Legend for Figure 8.3.5.4-2a.

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however, for resolving this issue, only the site need be divided further. The site is divided into two systems elements: the surface and the subsurface. The following sections describe each of these system elements and their function with respect to this issue.

Surface system element. The surface system element includes all radionuclide transport mechanisms which affect the occupational exposure of repository workers. Because of the proximity of the repository workers to the source, only the atmospheric transport mechanisms are important. The main processes involved are the physical transport, dispersion, and deposition of potential releases of radionuclides within the site boundary.

Subsurface system element. The subsurface system element includes the natural systems of the site that have a potential to impact the radiological safety of the repository workers while in the underground facilities. The natural radioactivity in the host rock (i.e., uranium, thorium, and radon, and their daughter products) is a source that has the potential to increase the radiation level in the restricted areas. The release of radionuclides from the site system would result from mining, transporting, and storing of the mined tuff and of the mine dewatering processes (if any). These sources are not expected to be significant. Note that exposure to naturally occurring radionuclides is not specifically regulated under 10 CFR Part 20. However, prudence dictates that total worker exposure be monitored and controlled through the implementation of applicable Mine Safety and Health Administration regulations (30 CFR 57) and DOE orders.

Repository system element. The repository system element includes all surface and subsurface systems that can be sources of man-made radiation exposure in the restricted area. The principal source of exposure to radiation in the repository system element is expected to be from high-level waste (HLW) handling operations. For these operations, the dominant source of occupational exposure is expected to be penetrating radiation (gamma rays and neutrons) emitted by the radioactive constituents of HLW. Exposure to radiation fields can be reduced by shielding or by limiting occupancy in the affected areas. These factors, among others, such as remote operation, will be considered in analysis of the repository design when recommendations are made for alternative means of meeting the performance goals for this issue and reducing the occupational exposures to ALARA levels.

Secondary radioactive wastes that will be generated on the site and processed by the waste treatment systems are another source of occupational exposure in the repository system element. The dominant mode of exposure to these sources is expected to be external exposure to the resultant radiation fields. As with the waste handling operations, the waste treatment system design will be periodically analyzed and, if necessary, modified to ensure that occupational exposure will be adequately controlled.

In addition to external exposure from the contained sources discussed above, there is a potential for internal exposure from radionuclides that may be released from containment and entrained in the ventilation air flow or brought to the surface by the mine dewatering system (if any). These expos-

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ures will be precluded during normal operations by measures such as radiation monitoring and sampling and ventilation control. In addition, protective clothing, and respiratory protection equipment will be available for use, if necessary.

The construction and operation of the repository may also require the use of radioactive sources not generated from HLW handling operations. These would include sealed sources used for the calibration of health physics and radiation monitoring equipment and the radiography sources used for nondestructive examination of welds and radioactive sources used in scientific investigations. Control of exposure to these sources will be accomplished primarily by following the proper operational procedures, instituting appropriate administrative controls on their use, and adequate training.

The last potential source of occupational radiation exposure is from the decommissioning of the facilities. The exposure during this phase of operation would be from contaminated and activated equipment, buildings, and natural materials. The worker dose will be controlled by designing the facilities for easy disassembly, control, and consolidation of contaminated materials, and limiting the generation of neutron activation products. Note that the retrieval of waste containers is considered an operation activity.

Allocation of performance to the system elements. The next four steps after the identification of functional requirements make up the bulk of the performance allocation process. In these steps performance measures, performance goals, and needed parameters are developed. The results of these steps may be seen in Tables 8.3.5.4-1 and 8.3.5.4-2. The rationale for the assignment of confidence levels and the calls for site data are presented in the information need discussions following this discussion.

Development of design criteria and constraints and identification of input items. The only constraints on the design of the repository forthcoming from this issue are those general performance goals shown in Tables 8.3.5.4-1 and 8.3.5.4-2. These performance goals are transmitted to Issue 2.7 (repository design criteria for radiological safety, Section 8.3.2.3) where specific design criteria are developed and transmitted to Issue 4.4 (preclosure design and technical feasibility, Section 8.3.2.5) for incorporation in the design of the repository. Specific design products or information required of either Issue 2.7 or Issue 4.4 are also transmitted to Issue 2.7. At this time, no specific design products or information items have been identified.

Worker radiological safety assessment package. The specific analytical approach for use in the resolution of this issue will be developed as part of the preclosure risk assessment methodology (PRAM) program described Section 8.3.5.1. Although some work has been performed to obtain preliminary estimates of worker doses, the approach used may not be the same as the final technique developed in the PRAM program. Therefore, only a general approach, shown in Figure 8.3.5.4-2 in the dashed box labeled "worker radiological safety assessment package" is discussed below. The following discussion provides a step-by-step explanation of the general approach to predict worker radiation doses during the normal operation of the repository.

Table 8.3.5.4-1. Functions, performance measures, and performance goals for Issue 2.2 (worker radiological safety--normal conditions) (page 1 of 2)

| System element | Function | Process or activity | Performance measure | Tentative goal | Needed confidence |
|---|---|---|---|---|---|
| Surface | Provide transport, dispersion, and diffusion of routine airborne radioactive effluents within site boundaries | Analyze atmospheric transport and dispersion characteristics within the site boundaries | Transport characteristics of atmosphere within site boundaries | A. Adequate atmospheric transport characteristics to assist in meeting dose limits | High |
| | | Analyze worker doses from outdoor airborne radionuclides within site boundaries | Doses resulting from airborne radionuclide concentrations around repository facilities | B. Total doses below limits of 10 CFR Part 20 and ALARA[a] | High |
| Subsurface | Provide assurance that doses to workers underground are not excessive | Analyze shielding of workers from direct radiation using properties of the host rock | Effective attenuation of direct radiation by host rock | C. Significant attenuation of direct radiation using host rock properties | High |
| | | Analyze the natural radiation released in the underground facilities | Release rates and concentrations of naturally occurring radionuclides | D. Natural radiation levels low enough to pose no significant health hazard to the workers | High |
| | | Analyze radiation levels from miscellaneous sources of radiation such as calibration and testing sources | Direct radiation and contamination levels from miscellaneous sources | E. Insignificant levels of direct radiation and contamination from miscellaneous sources | High |
| Repository | Provide containment of radiation and limit radiation doses to repository workers | Analyze direct radiation levels in all areas of the repository | Direct radiation levels in all areas of the repository | F. Levels low enough to keep doses to workers below limits of 10 CFR Part 20 and ALARA | High |





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Table 8.3.5.4-1. Functions, performance measures, and performance goals for Issue 2.2 (worker radiological safety--normal conditions) (page 2 of 2)

| System element | Function | Process or activity | Performance measure | Tentative goal | Needed confidence |
|---|---|---|---|---|---|
| Repository (continued) | Provide containment of radiation and limit radiation doses to repository workers (continued) | Analyze high-level waste containment and handling operations | Doses due to worker occupancy in direct radiation areas | G. Total doses below limits of 10 CFR Part 20 and ALARA | High |
| | | Analyze site-generated waste containment, handling, and treatment operations | Doses due to worker occupancy in direct radiation areas | H. Total doses below limits of 10 CFR Part 20 and ALARA | High |
| | | Analyze radiation levels from miscellaneous sources of radiation such as calibration and testing sources | Direct radiation and contamination levels from miscellaneous sources | I. Insignificant levels of direct radiation and contamination from miscellaneous sources | High |
| | | Analyze shielding provided by structures, containments, equipment, and waste packages | Effective attenuation of direct radiation levels | J. Significant attenuation of direct radiation from all sources | High |
| | | Analyze ventilation and filtration of repository airstreams | Contamination and airborne radionuclide concentrations in repository airstreams | K. Total doses below limits of 10 CFR Part 20 and ALARA | High |

*ALARA - as low as reasonably achievable.



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Table 8.3.5.4-2. Parameters required for Issue 2.2 (worker radiological safety--normal conditions) (page 1 of 2)

| Related performance goal[a] | Performance or design parameter | Parameter descriptor | Tentative parameter goal | Needed confidence | Expected parameter value(s) | Current confidence | SCP section providing parameters |
|---|---|---|---|---|---|---|---|
| A,B | Wind speeds | Site area | (b) | High | Figures 5-3 to 5-7, and Tables 5-6 and 5-7 | Medium | 8.3.1.12 |
| A,B | Wind direction | Site area | (b) | High | Figures 5-3 to 5-7, and Tables 5-6 and 5-7 | Medium | 8.3.1.12 |
| A,B | Atmospheric stability | Site area | (b) | Medium[c] | Table 5-11 | Medium | 8.3.1.12 |
| A,B | Mixing layer depth | Site area | (b) | Medium | (d) | Medium | 8.3.1.12 |
| A,B | Average ambient temperature | Site area | (b) | Medium | Tables 5-2 and 5-3 | Medium | 8.3.1.12 |
| A,B | Atmospheric moisture | Site area | (b) | Medium | Tables 5-2 and 5-5 | Medium | 8.3.1.12 |
| A,B | Precipitation type, amount, intensity, etc. | Site area | (b) | Medium | Tables 5-2 and 5-4 | Medium | 8.3.1.12 |
| A,B | Barometric pressure | Site area | (b) | Medium | Table 5-2 | Medium | 8.3.1.12 |
| A,B | Dust particle size distributions | Site area | 1 to 10 micron, normal | High | Data not available | Data not available | (e) |
| A,B | Size and distance of topographic features from release points | Site area | Topographic features beneficial to dispersion | Medium | See U.S Geological Survey topographic maps | High | Literature |
| B | Routine releases | (f) | (f) | (f) | (f) | (f) | (f) |
| B | Surface facilities layout | (f) | (f) | (f) | (f) | (f) | (f) |





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Table 8.3.5.4-2. Parameters required for Issue 2.2 (worker radiological safety--normal conditions) (page 2 of 2)

| Related performance goal[a] | Performance or design parameter | Parameter descriptor | Tentative parameter goal | Needed confidence | Expected parameter value(s) | Current confidence | SCP section providing parameters |
|---|---|---|---|---|---|---|---|
| C | Elemental composition of host rock | TSw2 unit[g] | Normal composition for tuffs | High | Normal composition for tuffs | Medium | 8.3.1.3 |
| C | Bulk density of host rock | TSw2 unit | (b) | High | 2.26 to 2.33 g/cc | Medium | 8.3.1.15 |
| C | Water content of host rock | TSw2 unit | (b) | High | 65% saturation | Medium | 8.3.1.16 |
| D | Radon emanation rate from tuff | TSw2 unit | (b) | High | 0.48 pCi/$m^2$-s | Low | 8.3.1.15[e] |
| E,F,G,H,I,J,K | Reference repository design, operating plan, and supporting analyses | No additional site characterization data needed--see footnote f | | | | | |

[a]The letters in this column key the performance parameters in this table to the tentative goals in Table 8.3.5.4-1.
[b]Tentative goal is to have further measurements of this parameter verify the range of expected values listed here.
[c]Medium confidence requirements are intended to indicate that these parameters need to be site specific.
[d]See Quiring (1968).
[e]Collection of these data is part of the environmental program planned activities and is addressed in the Radiological Monitoring Plan discussed in Section 8.3.1.13.
[f]For purposes of communicating the design information needed to evaluate worker radiological safety under normal conditions, the input items from Issue 4.4 (obtained through Issue 2.7) are collectively listed as a parameter.
[g]TSw2 unit is the nonlithophysal Topopah Spring unit (repository horizon).




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Design evaluation. The design package and site data are obtained from the reference information base (RIB), and the repository design features related to the radiological safety of the worker during normal operations are evaluated. The following text discusses what types of information are investigated during this design evaluation. The high-level waste (HLW) throughput (schedule and amount of waste received per year) is an important controlling factor in the design of the repository process and storage facilities (e.g., hot cell structure and lag storage) and, hence, in the radiation doses predicted for workers. Direct radiation that can be emitted from the central process area and the amount of routine radioactive effluents will be directly related to the amount of HLW on hand and being processed. These sources of potential dose to the workers also depend on how processes are conducted for such activities as waste receipt, lag storage, waste handling and consolidation, and transport of waste containers. Worker radiation doses from such activities will be controlled principally by design features and administrative procedures (e.g., limits on frequency of tasks and time in storage), which will be a secondary control on worker exposure. Attributes of the repository design that will play a major role in controlling direct radiation or release of radioactive effluents to the restricted area include such features as

1. Operations plan parameters such as number of workers present and time to complete tasks.

2. Remote-handling equipment used for tasks in high radiation or high frequency tasks.

3. Maintenance requirements of remote-handling and hot-cell equipment.

4. Barrier and shield thicknesses, composition, and distance to workers from the source.

5. Containment and ventilation system characteristics (e.g., repository and hot cell layout, differential pressures between areas, openings, air locks, and filters).

6. Radioactive material release point characteristics (e.g., stack height, diameter, exit velocity, temperature, and location within the restricted area).

In addition, as part of the regulatory performance verification requirements, specific systems and operational controls will be needed to verify that the repository design and operation does maintain annual radiation doses to the workers to less than the regulatory limits. Examples of the systems that must be provided include gaseous and liquid effluent monitoring and control equipment, effluent sampling and measuring equipment, area radiation and airborne monitoring equipment, and personnel and area dosimetry equipment. Design of these systems will be incorporated in the normal repository design process. The information needed for this design evaluation will be the product of the design process and will not depend directly on the site characterization activities.

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Identification of radiation source characteristics. Potential sources of radiation that can contribute to worker exposure in the restricted area can be categorized as (1) resulting from repository operations or (2) miscellaneous operations. Examples of radiation sources resulting from repository operations are receipt of HLW shipping casks, releases during spent fuel consolidation, transport of HLW containers, direct radiation from storage of disposal containers, direct radiation from emplacement activities, and naturally occurring radionuclides. Other miscellaneous operations that are potential radiation sources include treatment and transportation of site-generated low-level waste (LLW) and gamma and neutron radiation-producing equipment used in construction and nondestructive testing.

The specific information needed about the potential source terms includes the radionuclides involved and the quantity and concentration, decay radiation and energies, and physical and chemical forms of these radionuclides. General information needed about the source terms for dose evaluation include

1. Planned repository operational details (e.g., scheduled HLW throughput and inventories, LLW generation and transport rates, and normal effluent release rates).

2. Repository design features (e.g., radionuclide barriers, normal effluent release locations, layout distances, containment, leakage, and filtration details).

3. Environmental details (e.g., airborne transport and dispersion of radioactive materials within the restricted area).

4. Natural radionuclide sources (e.g., concentrations in tuff and ground water at the repository location).

Depending on the characteristics of the source terms, the information needs will be satisfied by the site characterization program (e.g., naturally occurring radionuclides), the repository design process (e.g., HLW and site-generated waste), or the environmental and socioeconomic monitoring programs (e.g., offsite installations and background radiation). Development of the analytical tools needed to evaluate potential adverse impacts of the source terms on worker safety will be coordinated with the PRAM program requirements and recommendations.

Radionuclide transport evaluation. The next element in the worker radiological safety assessment package is radionuclide transport evaluation following release from containment systems or repository facilities of radioactive material as a result of normal repository activities. The dispersion of airborne radioactive materials can result in radionuclide concentrations in the air that can cause an external dose by direct radiation or an internal dose through inhalation, or result in ground deposition of radioactive material. The dominant pathway for occupational exposure to airborne radionuclides is expected to be from radionuclides entrained in repository airstreams. Analysis of this pathway will require data on the radionuclide source terms, air volumetric flow rates, air patterns, and location of

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workers and length of occupancy. Analytical tools will be required for determining direct radiation dose rates in all areas of the repository, as well as for determining ventilation leakage and filtration of airborne radionuclides in the repository airstreams.

Analytical tools in the form of dispersion and pathway models also will be required to perform the radionuclide transport evaluation for restricted areas outside the facility. Meteorological data (e.g., wind speed, wind direction, and atmospheric stability) in the vicinity of the repository buildings, as well as repository design information, will be needed as input to the dispersion model. This information need will be satisfied by Characterization Program 8.3.1.12 (meteorology).

Worker radiation exposure calculation. The last step in the analysis is the radiological exposure evaluation that quantifies the dose to the individual worker from routine operation of the repository and offsite installations. The quantification of radiation doses will be performed by the use of accepted analytical models and knowledge of the various design features as input into the models. Some design features needed include

1. The processes and activities necessary for the functioning of the repository.

2. The layout and physical design features (i.e., location of processes and activities, wall thickness and material, personnel occupied areas, source location and storage, transport, and personnel corridors).

3. Repository throughput of radioactive materials.

4. Source terms (i.e., radionuclides involved, low-level waste generated, material quantities, material form (solid, liquid, particulate, or gaseous), container parameters, and industrial sources).

5. Duration and frequency of tasks.

6. Number of workers involved.

Accepted analytical methods for the calculation of personnel exposures will be selected or developed as part of the preclosure safety assessment activities consistent with the methodology described in Section 8.3.5.1. Computer models will be used to evaluate the potential radiation doses to workers where appropriate. Design-limiting assumptions will be specified for the code input parameters (e.g., radionuclide sources). The following types of analytical tools will be needed:

1. Repository operations models.
2. Building ventilation, filtration, and leakage models.
3. Radiation shielding models.
4. Atmospheric dispersion models.
5. LLW treatment and transportation radiological impact models.
6. Radiological consequences assessment models.

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The information needed to calculate doses using these analytical tools will be provided as discussed in the previous steps. This information will be the product of the site characterization program, the socioeconomic and environmental monitoring program, and the normal repository design process. The following list indicates some technical guidance documents that might be applicable to the development of the analytical tools. A list of analytical tools that are available for use is contained in Section 8.3.5.19 (completed analytical techniques). Further discussions of analytical tools still needed are contained in Section 8.3.5.20 (techniques requiring development).

1. Regulatory Guide 1.69--Concrete Radiation Shields for Nuclear Power Plants (December 1973) (NRC, 1973).

2. Regulatory Guide 8.19--Occupational Radiation Dose Assessment in Light-Water-Reactor Power Plants--Design Stage Man-rem Estimates (Rev. 1, July 1979) (NRC, 1979a).

3. Safety Series No. 60--Criteria for Underground Disposal of Solid Radioactive Waste (IAEA, 1983b).

4. Safety Series No. 68--Performance Assessment for Underground Radioactive Waste Disposal Systems (IAEA, 1985).

5. DOE/EV/1830-T5--A Guide to Reducing Radiation Exposures to As Low As Reasonably Achievable (ALARA) (Kathren et al., 1980)

6. DOE Order 6430.1--General Design Criteria Manual, as applicable (December 1983) (DOE, 1983a).

<u>Performance evaluation for compliance with goals</u>. The remainder of Figure 8.3.5.4-2 deals with the final evaluation of the results documented in the worker radiological safety assessment package. The results are compared with the regulatory limits contained in the regulations listed in the section entitled "regulatory basis for this issue". If all the limits have been met, then the results are examined to see if the ALARA criterion has been met. If both the regulatory limits and the ALARA criterion have been met and if the design is in the final design phase, then the design is ready for license application and a favorable issue resolution has been achieved. If both the regulatory limits and the ALARA criterion have been met but the design is <u>not</u> in the final design phase, then this process is repeated for the next design phase.

If the results of the worker radiological safety assessment package do not meet either the regulatory limits or the ALARA criterion, then design, procedural, or operational changes are recommended to correct the situation. If these changes cannot be made and the performance goals cannot be reasonably changed, then an unfavorable resolution of the issue has occurred (i.e., not feasible to proceed). However, if the design, procedural, or operational changes can be made or the performance goals can be reasonably changed, then the recommended changes are implemented and the whole process is repeated.

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Interrelationships of information needs

The question asked by this issue (2.2) addresses the radiological health and safety of the workers during the normal operations of the repository. The resolution of this issue can be obtained by answering two questions:

1. Given the repository design, what is the expected radiation environment on the surface and in the surface and subsurface facilities due to natural and man-made sources of radiation?

2. For the normal operations of the repository, what are the projected worker radiation doses for the normal operations of the repository and do these doses meet applicable requirements?

There is a one-to-one correspondence between these questions and the two steps in Figure 8.3.5.4-2 in the box labeled worker radiological safety assessment package. The two questions have been designated Information Needs 2.2.1 and 2.2.2, respectively. Information Need 2.2.1 describes the radiation environments that workers may be subjected to during the course of their work. This information need requires (1) site data to determine the radiation environments resulting from natural radioactivity and the background radiation of the site for baseline definition purposes and (2) design data to evaluate the effects of the design on the radiation environment.

Information Need 2.2.2 is a determination of the expected exposure conditions and worker radiation doses resulting from the normal operations of the repository. As described earlier in the section called worker radiological safety assessment package, the doses to the workers are predicted using radionuclide transport and dispersion models, radiation shielding models, the repository operating plan, and radiological dose assessment models. The final resolution of this issue will take place under this information need when the results of the dose calculations are evaluated and compared with the regulatory limits contained in 10 CFR Part 20.

The functions and performance measures (associated with the MGDS system elements) necessary for answering these two questions and resolving this issue are listed in Table 8.3.5.4-1. The site data needed to answer these two questions are listed in Table 8.3.5.4-2.

## 8.3.5.4.1 Information Need 2.2.1: Determination of radiation environment in surface and subsurface facilities due to natural and manmade radioactivity

Technical basis for addressing the information need

Link to the technical data chapters and applicable support documents

Further discussions on the subject of radiological protection of the workers may be found in Sections 6.1.1.4.1 (radiological protection design requirements) and 6.4.5 (Issue 2.2: radiological safety of workers--normal conditions). Section 8.3.5.1 contains discussions on the preclosure risk assessment methodology (PRAM) program, which, as part of its scope, includes

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radiological risk to the workers during normal operations. Sections 2.5 (radiological protection) and 6.1 (radioactive releases during normal operations) of the Site Characterization Plan-Conceptual Design Report (SCP-CDR) (SNL, 1987) also contain discussions relevant to this issue. Section 6.1 of the SCP-CDR is especially informative because it contains some preliminary estimates of expected releases during normal operations of the repository.

Parameters

The parameters required by this information need are those site and design parameters relevant to the determination of the radiation environment on the surface and in the surface and subsurface facilities. The relevant design information is noted in Table 8.3.5.4-2 and further information on these needs is not required at this time. The reference repository design and supporting analyses will be obtained from the reference information base (RIB) and will contain all design details necessary to perform the required evaluations.

There is only one piece of site data needed to satisfy this information need: the radon emanation rate of the mined tuff. Collection of these data is part of the environmental program planned activities and is addressed in the Yucca Mountain Project Radiological Monitoring Plan discussed in Section 8.3.1.13. All other data is design data and will be obtained from the reference information base.

Logic

The determination of the radiation environment on the surface and in the surface and subsurface facilities requires information about the site, the potential sources of radiation, and the repository design. Information about the repository design is obtained from the repository reference design, as is information about the potential man-made sources of radiation. Information about the site is obtained through the site characterization program. Using this information, airborne radionuclide concentrations are estimated for the surface and subsurface facilities and for the area on the surface surrounding the repository. Radiation levels from direct radiation sources are then calculated to establish dose rates from the different source terms. Once potential sources of radiation are accounted for, radiation areas are established and associated radiation levels for both direct and airborne radiation are determined. Table 8.3.5.4-2 lists the data (in addition to radon emanation rate from the tuff) required to perform this task. After these calculations are completed, this information need is satisfied, and the results can then be used in Information Need 2.2.2 to determine radiation doses to workers.

Objectives

The objective of this activity is to refine the list of site data parameters just presented in the technical basis section for Information Need 2.2.1. This list may be incomplete or the level of confidence (as shown in Table 8.3.5.4-2) required may be inappropriate.

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8.3.5.4.1.1 Activity 2.2.1.1: Refinement of site data parameters required for Issue 2.2

Parameters

The list of parameters presented in the technical basis section for Information Need 2.2.1 is the starting point for this activity. As the activity progresses parameters may be added to or deleted from this list.

Description

The parameter list will be refined in three ways. First, during site characterization, reviews and activities by those organizations responsible for collecting data will discover problems with parameter lists. These problems will be resolved and parameter lists will be revised. Second, the PRAM program will be developing methods for radiological performance analyses (Performance Assessment Activity 2.2.2.2 in Section 8.3.5.4.2.2). During the development of these methods, it is expected that lists of required parameters for each type of analysis will be created. A review of these parameter lists may result in refinement of the Issue 2.2 parameter list. Finally, a performance assessment of the advanced conceptual design (ACD) and license application design (LAD) for worker radiological safety (Performance Activity 2.2.1.2) may uncover deficiencies in the current parameter list. This is an ongoing activity whose end date is the completion of the license application.

8.3.5.4.1.2 Activity 2.2.1.2: Advanced conceptual design assessment of the worker radiological safety during the normal operations of the Yucca Mountain repository

Objectives

The objective of this activity is to perform a worker radiological safety assessment of the ACD for a Yucca Mountain repository. Secondary objectives of this activity are to provide information for the refinement of the site data parameter list for Issue 2.2 (Performance Assessment Activity 2.2.1.1 in the previous sections) and to provide feedback to the PRAM program for future methods development activities (Performance Assessment Activity 2.2.2.2, Section 8.3.5.4.2.2).

Parameters

The parameters necessary for this activity are those listed in the site data parameter list for Issue 2.2 presented in the technical basis section for Information Need 2.2.1.

Description

This activity will assess the ACD for worker radiological safety during normal operations. A general description of the process is presented earlier in the section on worker radiological safety assessment package.

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8.3.5.4.2 Information Need 2.2.2: Determination that projected worker exposures and exposure conditions under normal conditions meet applicable requirements

Technical basis for addressing the information need

Link to the technical data chapters and applicable support documents

Chapter 3 discusses the present state of the knowledge on the site hydrology, including uses of surface water and ground water. Chapter 5 discusses the present state of the knowledge on the meteorology of the site and surrounding region. Further discussions on the subject of radiological protection of the workers may be found in Sections 6.1.1.4.1 (radiological protection design requirements) and 6.4.5 (Issue 2.2: radiological safety expected to workers--normal conditions). Section 8.3.5.1 contains discussions on the preclosure risk assessment methodology (PRAM) program. The PRAM program includes radiological risk to workers during normal operations as part of its scope. Sections 2.5 (radiological protection) and 6.1 (radioactive releases during normal operations), of the Site Characterization Plan-Conceptual Design Report (SCP-CDR) (SNL, 1987) also contain discussions relevant to this issue. Section 6.1 of the SCP-CDR is especially informative because it contains some preliminary estimates of expected releases during normal operations of the repository.

Parameters

The parameters required by this information need are those site and design parameters relevant to the prediction of worker radiation doses during the normal operations of the repository. The calculation of worker doses due to airborne radionuclides within the facilities requires only design information; however, the determination of worker doses due to airborne radionuclides outside the facilities does require site data. Design information needed for this purpose is noted in Table 8.3.5.4-2, and further information on these needs is not required at this time. Reference repository design information and supporting analyses will be obtained from the reference information base (RIB), which will contain all design the details necessary to perform the required evaluations.

The site data required to satisfy this information need are obtained through various characterization programs and also through the RIB. Following is a summary of the required site data and the SCP section providing the information:

| Data requirement | SCP section |
|---|---|
| METEOROLOGICAL DATA | |
| Wind speeds in the region | 8.3.1.12.2 |
| Prevalent wind directions | 8.3.1.12.2 |

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| Data requirement | SCP section |
|---|---|
| Atmospheric stability of the area | 8.3.1.12.2 |
| Atmospheric mixing layer depth of the area | 8.3.1.12.2 |
| Average ambient temperature of the area | 8.3.1.12.2 |
| Atmospheric moisture of the area | 8.3.1.12.2 |
| Area precipitation, including type, amount, intensity, etc. | 8.3.1.12.2 |
| Barometric pressure | 8.3.1.12.2 |
| Dust particle size distributions | 8.3.1.12.2 |
| Size and distance of major topographic features from release points | 8.3.1.14.1 |
| REPOSITORY ROCK AND GROUND-WATER DATA | |
| Elemental composition of the host rock | 8.3.1.3.2 |
| Bulk density of the host rock | 8.3.1.15.1 |
| Water content and saturation of the host rock | 8.3.1.12.3 |

These parameters are needed with differing levels of confidence and for different locations on and around the site as shown in Table 8.3.5.4-2.

Logic

Once the results of Information Need 2.2.1 are obtained, the prediction of worker doses during normal operations may begin. The calculation of worker exposures to airborne radionuclides on the surface outside the facilities depends on the concentrations of radionuclides released from the repository (obtained from the reference repository design and Information Need 2.2.1), the meteorological conditions surrounding the facilities, and to a lesser extent, sources in the environment. Worker doses from airborne radionuclides inside the repository facilities are determined from the radiation levels estimated by Information Need 2.2.1 and the repository operations plan. Worker doses resulting from direct radiation in the surface facilities can be predicted using the characteristics of the repository design, the information on radiation areas supplied by Information Need 2.2.1, and the repository operations plan. The prediction of doses resulting from direct ra-

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diation from emplaced waste in the underground facilities requires data on the density and composition of the repository rock. With this information, the shielding provided by the host rock can be determined and the worker doses from emplaced waste predicted. The repository design will provide shielding data needs for the transporter and other emplacement and retrieval equipment. Once all these contributions to worker doses are determined and combined, the results are used to predict compliance with applicable requirements and provide a resolution of Issue 2.2. In addition, ground-water data will be obtained for assessing its contribution to shielding of gamma and neutron radiation emitted by the waste package.

8.3.5.4.2.1 Activity 2.2.2.1: Refinement of site data parameters required for Issue 2.2

Objectives

The objective of this activity is to refine the list of site data parameters presented in the technical basis section for Information Need 2.2.2. This list may be incomplete or the level of confidence required (as shown in Table 8.3.5.4-2) may be inappropriate.

Parameters

The list of parameters presented in the technical basis section for Information Need 2.2.2 is the starting point for this activity. As the activity progresses parameters may be added to or deleted from this list.

Description

The parameter list will be refined in three ways. First, during site characterization, reviews and activities by those organizations responsible for collecting data will discover problems with parameter lists. These problems will be resolved and parameter lists will be revised. Second, the PRAM program will be developing methods for radiological performance analyses (Performance Assessment Activity 2.2.2.2). During the development of these methods, it is expected that lists of required parameters for each type of analysis will be created. A review of these parameter lists may result in refinement of the Issue 2.2 parameter list. Finally, a performance assessment of the advanced conceptual design for worker radiological (Performance Assessment Activity 2.2.2.3) safety may uncover deficiencies in the current parameter list. This is an ongoing activity whose end date is the completion of the license application.

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8.3.5.4.2.2 Activity 2.2.2.2: Development of performance assessment activities through the preclosure risk assessment methodology program

Objectives

Performance assessment methods development efforts in the preclosure risk assessment methodology. The objective of this activity is to benefit from the PRAM program. The Yucca Mountain Project will participate in the PRAM program through the PRAM Working Group and will adapt the PRAM program to the Yucca Mountain Program. A secondary objective of this activity is to use the information developed in this activity to assist in refining the site data parameters list for this issue (Performance Assessment Activity 2.2.2.1 described in the previous section).

Parameters

Initially there are no parameters for this activity; however, a list of parameters will develop as a result of the PRAM program and other project activities.

Description

The objective of the PRAM program is to develop a consistent preclosure safety assessment methodology. A part of this methodology will be concerned with the assessment of worker radiological safety during the normal operations of a repository. The Yucca Mountain Project will participate in this program and assist in the development of the overall methodology. Methods developed in the PRAM program will be adapted for use in the Yucca Mountain Project assessment of worker radiological safety during the normal operations of the Yucca Mountain repository (Performance Assessment Activity 2.2.2.3 in Section 8.3.5.4.2.3). Since the PRAM program is expected to continue through license application design this activity will be ongoing through license application. A more detailed discussion of the PRAM program is presented in Section 8.3.5.1.

8.3.5.4.2.3 Activity 2.2.2.3: Advanced conceptual design assessment of the worker radiological safety during the normal operations of the Yucca Mountain repository

Objectives

The objective of this activity is to perform a worker radiological safety assessment of the advanced conceptual design for the Yucca Mountain repository. Secondary objectives of this activity are to provide information for the refinement of the site data parameter list for this issue (Performance Assessment Activity 2.2.2.1) and to provide feedback to the PRAM program for future methods development activities (Performance Assessment Activity 2.2.2.2).

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Parameters

The parameters necessary for this activity are those listed in the site data parameter list for this issue presented in the technical basis section for Information Need 2.2.2.

Description

This activity will assess the Yucca Mountain repository advanced conceptual design for worker radiological safety during normal operations. A general description of the process is presented in the section on worker radiological safety assessment package.

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8.3.5.5 Issue resolution strategy for Issue 2.3: Can the repository be designed, constructed, operated, closed, and decommissioned in such a way that credible accidents do not result in projected radiological exposures of the general public at the nearest boundary of the unrestricted area, or workers in the restricted area, in excess of applicable limiting values?

Resolution of this issue requires the assurance that during the preclosure period the repository will not pose any undue radiological risk to the health and safety of the public and repository workers as a result of possible accidents. This will be initially established by an analysis documenting the adequacy of structures, systems, and components provided for the prevention of accidents and mitigation of consequences. The structures, systems, and components to be analyzed are those that will be presented to the NRC in the safety analysis report (SAR) of the license application. Frequent interactions with the NRC on site-specific preclosure activities are planned. Regulatory closure of this issue will first occur when the NRC issues a favorable safety evaluation report (SER) on the license application.

The relationship of this issue with the other issues of the issues hierarchy is shown in Figure 8.3.2.1-1 (Section 8.3.2.1), which illustrates the relationship between design and performance issues and fixes the lines of communication between these issues. To be more specific about the relationship of this issue to the other issues with which it has direct or very strong ties, only Issues 2.1 (Section 8.3.5.3), 2.2 (Section 8.3.5.4), 2.3 (this issue), 2.7 (Section 8.3.2.3), and 4.4 (Section 8.3.2.5) are shown in Figure 8.3.5.5-1. The figure defines the ties between these issues by indicating the major information items passed between them. The figure also illustrates the connection of all these issues with the site characterization program. The scope of an issue is indicated by its size with respect to the other issues in the figure. Note that Issue 4.4 is the largest in scope, and the other issues, including this issue, branch out from Issue 4.4, reducing the scope to more specific areas. In the discussion that follows in this section, the regulatory basis for addressing accidental radiological releases is presented, the approach to resolving this issue is described, and the interrelationships among the information needs are discussed.

Regulatory basis for the issue

Although the issue states that radiation exposures resulting from credible accidents must be maintained below applicable limits, there are currently no regulatory limits for radiation exposures to either members of the public or repository workers from accidents at a repository. 10 CFR Part 60 does not specify an accident dose guideline to the public. The DOE has initiated steps to petition the NRC to amend the rule so as to include an accident dose guideline in Part 60. When such guideline is promulgated, it will be addressed in the repository design. Regulatory criteria pertaining to worker exposure during accidents for other situations and facilities will be considered.

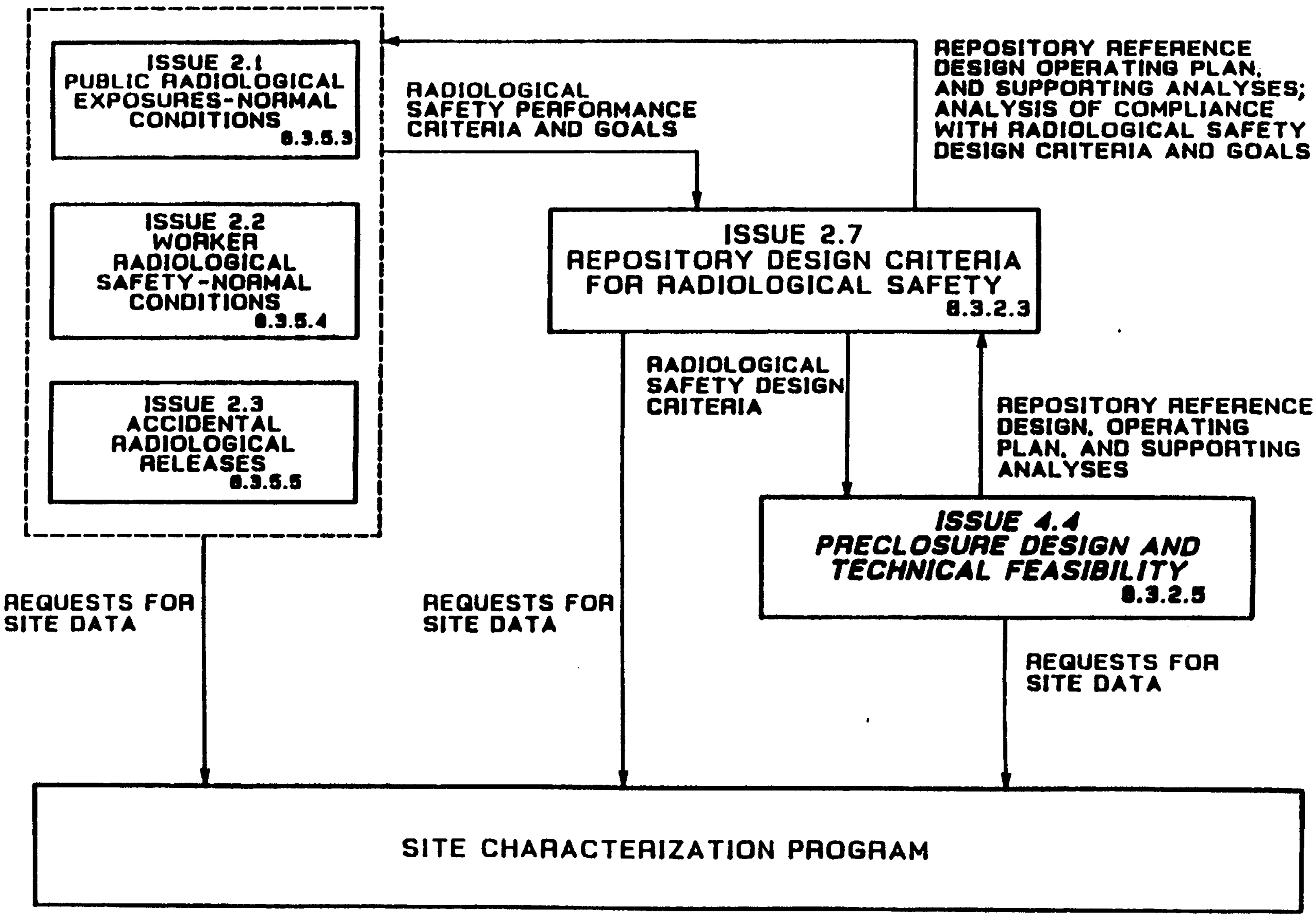


Figure 8.3.5.5-1. Relationship of Issue 2 3 (accidental radiological releases) to other issues and the site characterization programs



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Approach to resolving the issue

Licensing strategy overview

As stated earlier, the resolution of this issue requires the assurance that during the preclosure period the repository will not pose any undue radiological risk to the health and safety of the public and essential repository workers as a result of possible accidents. The preclosure period encompasses all activities associated with repository operation; simultaneous mining, construction, and emplacement; retrieval; decommissioning; and closure. The possibility of accidents will be considered for all underground and surface facilities, systems, or operations within the repository site boundary. The features initially assumed for issue resolution are (1) the reference repository design and operations described in the site characterization plan-conceptual design report (SCP-CDR) (SNL, 1987) and (2) site characteristics known to date from reconnaissance investigations. These encompass the appropriate elements from the hierarchy for the mined geologic disposal system (MGDS) at Yucca Mountain. The accident initiators that will be considered are natural phenomena, equipment failure or malfunction, and man-made events, including human error. Besides the initiating event that starts the accident sequences, other events or failures (called intermediate events) that are direct or consequential results will be considered in developing the accident sequence.

Using the methods consistent with those developed by the NRC, the DOE, and the preclosure risk assessment methodology (PRAM) program (see Section 8.3.5.1), the full range of the accident sequences will be identified, developed, and screened to establish the set of design-basis accidents for which radiological consequence assessments will be made. The definition of credible accidents is still being discussed within the DOE. Probabilistic analyses are expected to be performed to support, or perhaps to establish, the design-basis accident selections and to estimate the radiological risk to the public resulting from the repository. As part of the safety analyses, evaluations will also be made of the systems designed to prevent the accidents, to detect the accidents, and to mitigate the radiological consequences of the accidents. The protection of public health and safety will be demonstrated by comparing the doses calculated in the radiological consequence assessments with the criteria established within the repository program or with regulatory limits, if and when the regulatory limits are established.

These analyses will be reviewed at each design phase to determine the need for improvements or updating due to new information. The iteration of design and safety analysis, taking into consideration a proper balance between risk and cost, is expected to result in a well-designed MGDS. Finally, this issue will be resolved when (1) the set of credible design basis accidents has been established and analyzed using a deterministic approach, (2) supporting probabilistic risk analyses have been completed, and (3) both have been described in a format appropriate for the safety analysis report.

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Application of the issue resolution strategy

The logic to be used in resolving this issue is illustrated in the logic diagrams shown in Figures 8.3.5.5-2 and -3. These logic diagrams depict how the generic issue resolution strategy of Section 8.2.2 is to be applied to this issue. The first step of the process, identifying regulatory requirements, was discussed earlier in the section entitled "regulatory basis for the issue." The following discussions will explain each of the remaining steps in the resolution of this issue as shown in the logic diagram.

Identification of functional requirements. To allocate performance in this issue to specific system elements of the MGDS, the functions of these system elements with respect to this issue must be identified. The pre-closure portion of the MGDS is divided into three major system elements: the site, the repository, and the waste package. The waste package will not be considered by itself in allocating performance for this issue but will be considered in Section 8.3.4. The waste package will be considered as part of the repository system element equipment. The major system elements are further subdivided into more specific system elements; however, for resolving this issue, this level of detail is sufficient. In addition to these two system elements from the MGDS requirements, a third system element (offsite installations) is required for the resolution of this issue. The following sections describe each of these system elements and their role with respect to this issue.

Site system element. Disturbances in the site system element can induce accidents in the repository. The site events that could initiate accidents would primarily be natural disruptive phenomena such as earthquakes, rockfall, or potential methane or water intrusion. Structures, systems, and components important to safety (as defined in 10 CFR 60.2) must be protected against these phenomena.

Atmospheric transport of airborne radionuclides is expected to be the dominant pathway by which members of the public can be impacted by an accidental release of radioactive material. This is an important pathway for accident analysis because exposure can occur shortly after the release, before implementation of protective actions and, thus, must be dealt with through design. The relevant processes include atmospheric transport and dispersion, plume depletion, and deposition on the ground and in bodies of water. Atmospheric transport of airborne radionuclides is also important with respect to radiation exposure of the repository workers; however, direct exposure to penetrating radiation may be a more important source of radiation exposure for some workers in the vicinity of an accident. Exposure of essential workers is controlled by design features and is therefore in the domain of the repository system element.

The surface environment also includes longer-term pathways through which public exposure could occur after an accident. These pathways include surface-water bodies into which radioactive liquids could be accidentally released or into which radionuclides initially deposited on land could be washed down by precipitation. Long-term pathways also include inhalation of resuspended material deposited on the ground and ingestion of food products contaminated by uptake into plants, milk, and meat animals. Since these pathways are amenable to protective actions such as interdiction and decon-

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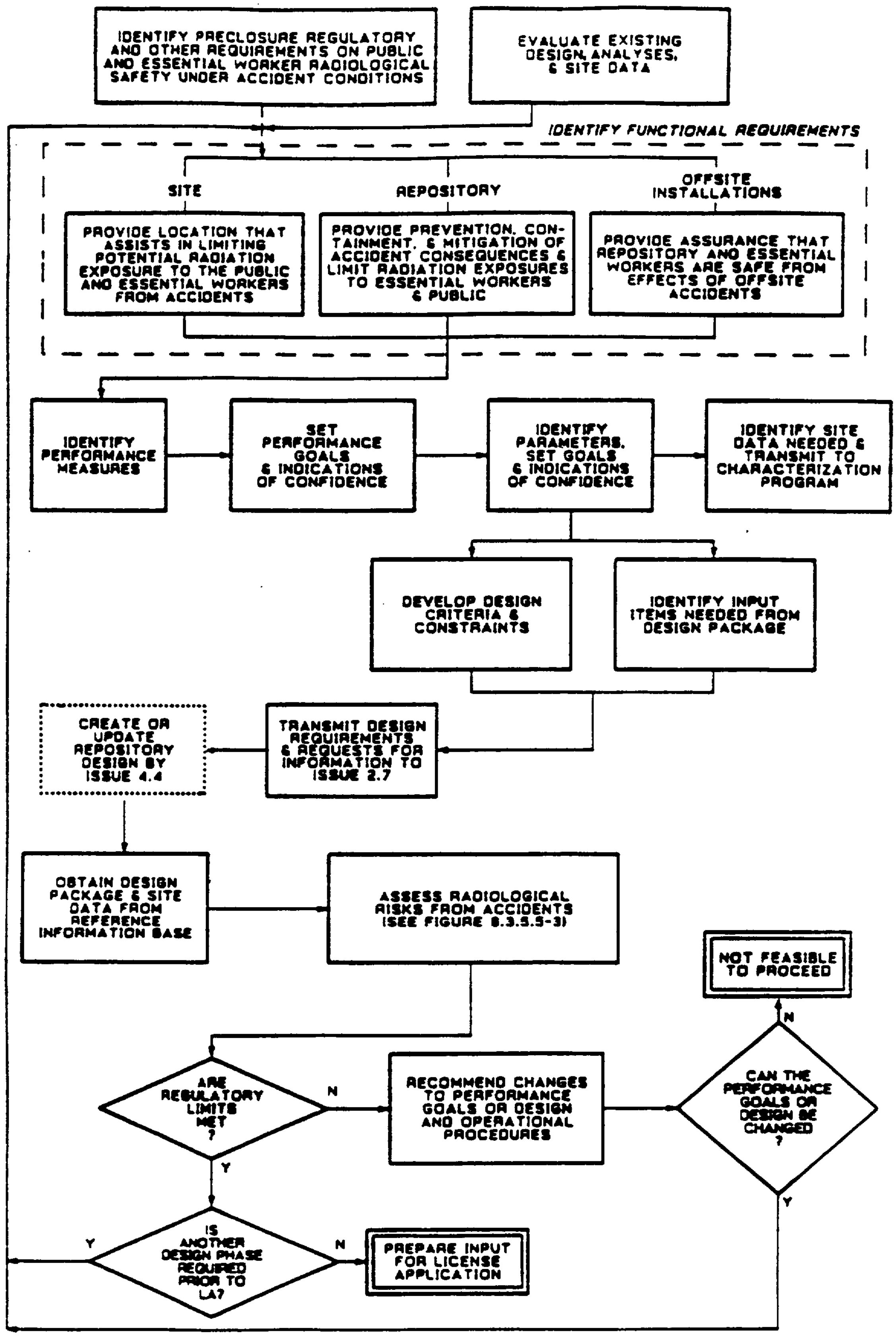


**Figure 8.3.5.5-2a.** Logic diagram for Issue 2.3 (accidental radiological releases). See Figure 8.3.5.5-2b for legend. Section 8.3.2.1 describes the relationships and interfaces between design and performance issues.

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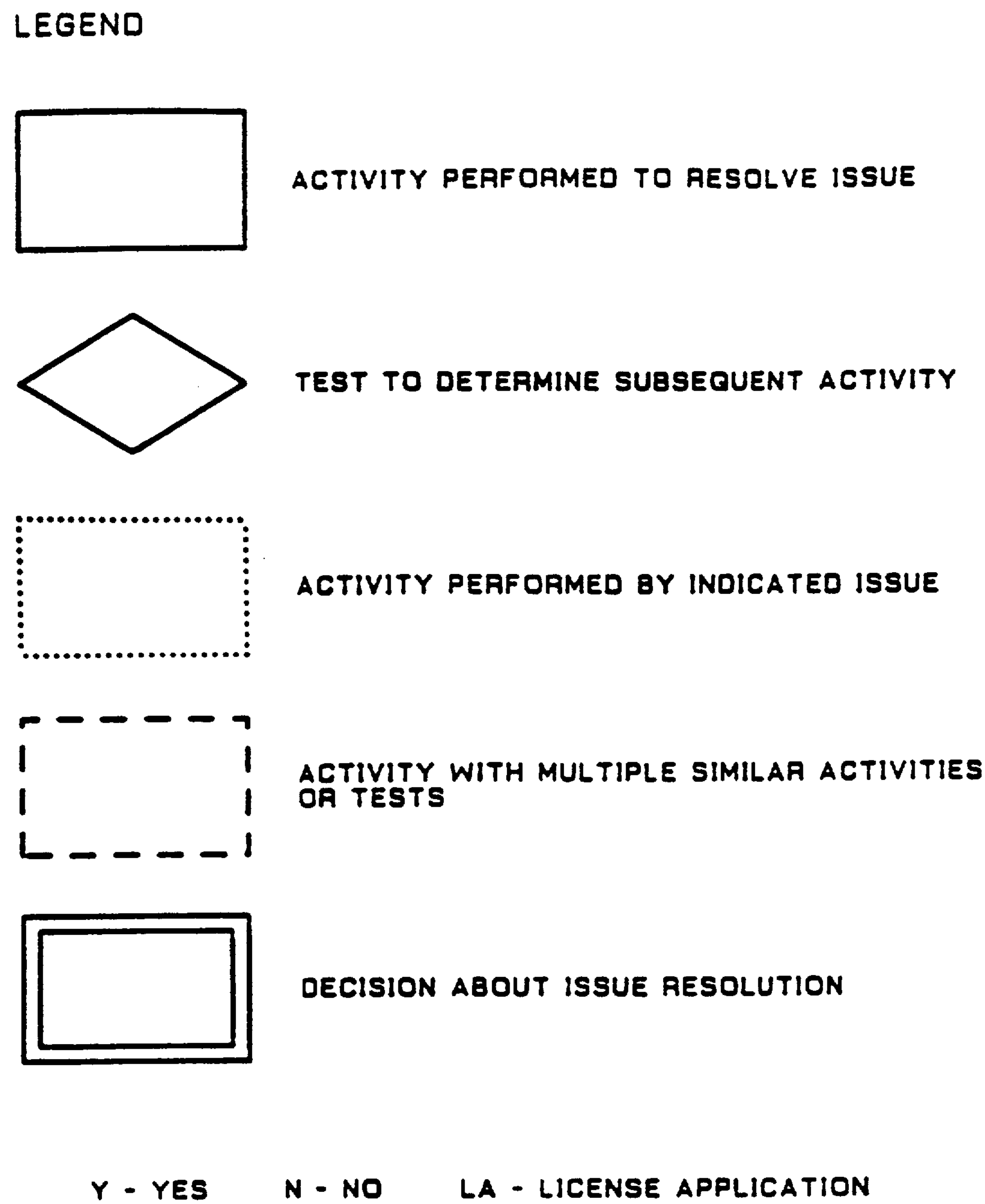


**Figure 8.3.5.5-2b. Legend for Figure 8.3.5.5-2a.**

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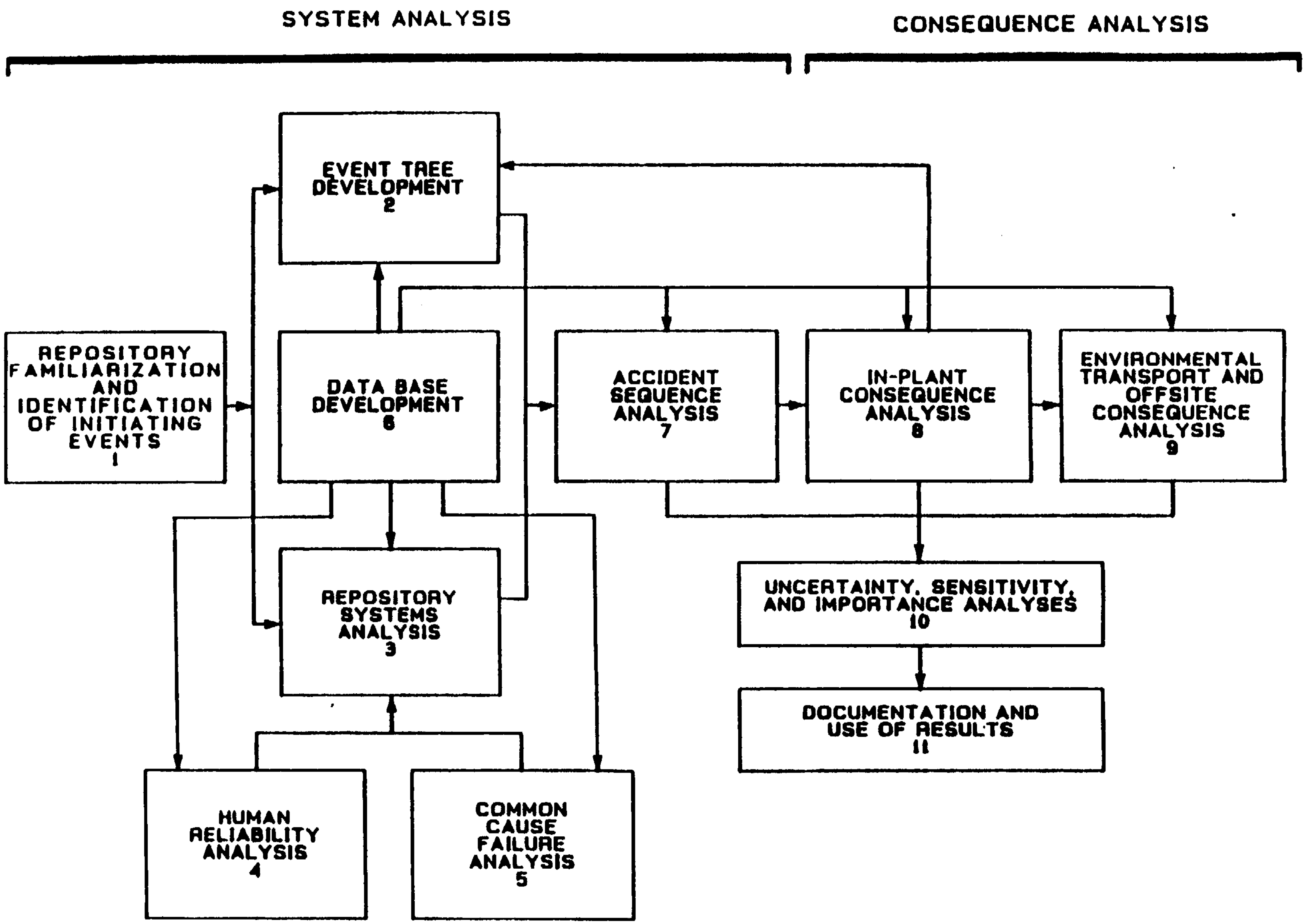


Figure 8.3.5.5-3. Analytical steps for assessing radiological risks from accidents. (The numbers in the diagram are keyed to text discussion )

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tamination of contaminated land and food products, they are not expected to be significant.

In view of the previous discussion, the site system element provides a location that assists in limiting potential radiation exposure to the public and repository workers from accidents. The site system element attributes that affect accidental radiation exposures include: (1) the remoteness of the site, (2) site-related accident initiators, (3) quick-acting radionuclide transport pathways, and (4) long-term radionuclide transport pathways. These four attributes define the analysis activities required to evaluate the effectiveness of the site system element in limiting accidental exposures.

Repository system element. The repository is important to the resolution of this issue because it contains the radioactive material available for release during an accident and, thus, will provide the source term for accident analysis. The radioactive material includes the high-level waste handled in the facilities and any secondary wastes generated on the site. In addition to providing the source term for accident analysis, the site facilities system is important because it contains the systems whose failure can initiate or continue an accident, as well as the systems that can prevent or mitigate an accident.

The repository will be designed to prevent, contain, and mitigate accident consequences and to limit radiation exposures to essential repository workers and the public. To evaluate the repository system element performance with respect to these objectives (for this issue), four main analysis activities are required. These are (1) analysis of the probabilities and consequences of design-related initiating events; (2) analysis of the vulnerability of the repository to the effects of natural, site-related, and design-related initiating events; (3) analysis of the effectiveness of preventive systems and design features; and (4) analysis of the effectiveness of mitigative systems and design features. All these analyses are interrelated and will be performed in parallel.

Offsite installations. Offsite installations are relevant to resolution of this issue because accidents at those facilities could be the initiating events for accidents at the repository. The operations performed in the local area include defense operations, transportation, surface disposal and storage of radioactive waste, and possibly nuclear fuel cycle operations. Potential hazards to the repository from these operations will be assessed and may contribute to the design basis for structures, systems, and components important to safety and to initiating events for accident analysis. Because of the large distances involved and past history, it is expected that the safety of the repository and essential workers from offsite accidents can be fulfilled by systems designed to handle onsite accidents.

Allocation of performance to the system elements. The next four steps in Figure 8.3.5.5-2 after the identification of functional requirements make up the bulk of the performance allocation process. In these steps performance measures, performance goals, and needed parameters are developed. For each system element, the functions it will serve in the resolution of this issue are listed in Table 8.3.5.5-1. The processes and activities that take place in serving each of these functions are also listed. Since this is a performance issue, the purpose of which is to analyze the performance of the

Table 8.3.5.5-1. Functions, performance measures, and performance goals for Issue 2.3 (accidental radiological releases) (page 1 of 2)

| System element | Function | Process or activity | Performance measure | Tentative goal | Needed confidence |
|---|---|---|---|---|---|
| Site | Provide location that assists in limiting potential radiation exposure to the public and essential workers from accidents | Analyze remoteness of repository location | Population density of region | A. Resolution of Issue 2.5 (higher level findings--preclosure radiological safety) | High |
| | | Analyze probabilities and consequences of accidents caused by natural or site-related phenomena[a] | Consequences of credible site-related accidents | B. Radiation doses well below applicable limiting values | High |
| | | Analyze short-term public and essential worker radiation exposure mitigation features of the site | Quick-acting dispersion and transport characteristics of the site | C. Adequate short-term transport characteristics to assist in limiting doses | High |
| | | Analyze long-term public and essential worker radiation exposure mitigation features of the site | Long-term dispersion, diffusion, and bioaccummulation characteristics of the site | D. Adequate long-term transport characteristics to assist in limiting doses | High |
| Repository | Provide prevention, containment, and mitigation of accident consequences and limit radiation exposures to essential workers and public | Analyze probabilities and consequences of design-related accidents | Consequences of credible design-related initiating events | E. Radiation doses well below applicable limiting values | High |
| | | Analyze design for vulnerability to effects of natural, site-related, and design-related accidents | Sensitivity of repository design to possible accidents | F. Repository designed to provide desirable responses to possible accidents | High |



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Table 8.3.5.5-1. Functions, performance measures, and performance goals for Issue 2.3 (accidental radiological releases) (page 2 of 2)

| System element | Function | Process or activity | Performance measure | Tentative goal | Needed confidence |
|---|---|---|---|---|---|
| Repository (continued) | | Analyze design for effectiveness of preventive features | Effectiveness of preventive design features | G. Near total prevention of accidents and consequences | High |
| | | Analyze design for effectiveness of mitigative features | Effectiveness of mitigative design features | H. Mitigation of accident consequences to well below applicable limiting values | High |
| Offsite installations | Provide assurance that repository and essential workers are safe from effects of offsite accidents | Analyze vulnerability of repository and essential workers to effects of offsite accidents | Consequences of credible offsite accidents that could affect the repository and the essential workers | I. Radiation doses well below applicable limiting values | High |

[a]Short-term radiation exposures will be evaluated to assess compliance with applicable accident dose limits. Long-term radiation exposures will be used to assess risk from credible accidents.





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repository systems with respect to radiological health and safety under accident conditions, these processes and activities are analyses rather than physical processes. The quantity used to measure the performance under each analysis, called a performance measure, is listed for each process, together with a goal for that performance measure. These goals are selected so that if they are met, then the regulatory requirements are satisfied with some margin of safety. The present goals are tentative, permitting future adjustment in the allocation between subsystems, if necessary. Finally, the confidence needed in meeting these goals is an indication of the relative importance of each performance measure in contributing to meeting the ultimate regulatory requirement. The results of the performance allocation steps may be seen in Tables 8.3.5.5-1 and 8.3.5.5-2.

Development of design criteria and constraints and identification of input items. The only constraints on the design of the repository forthcoming from this issue are those general performance goals shown in Tables 8.3.5.5-1 and 8.3.5.5-2. These performance goals are transmitted to Issue 2.7 (Section 8.3.2.3) where specific design criteria are developed and transmitted to Issue 4.4 (Section 8.3.2.5) for incorporation in the design of the repository (see Figure 8.3.5.5-1). Specific design products or information required for either Issue 2.7 or Issue 4.4 are also transmitted to Issue 2.7. At this time, no specific design products or information items have been identified.

Analytical approach for radiological safety assessment of accidents. The general analytical approach for the assessment of radiological risks from accidents to the public and essential workers is illustrated in Figure 8.3.5.5-3. The following is a brief discussion of the steps shown in Figure 8.3.5.5-3. This methodology is discussed in more detail in Section 8.3.5.1.

Step 1--Repository familiarization and identification of initiating events. The objectives of step 1 are to (1) identify and describe the physical configurations and processes of the repository systems and support systems to be modeled, and (2) identify the accident initiating events to be considered in the risk assessment. The accident initiators of most concern are those that occur during waste handling on the surface and during waste emplacement and retrieval, if required, in the underground facilities.

Step 2--Event tree development. The objective of step 2 is to identify the potential accident sequences that could occur following the initiating events. Accident sequences are commonly identified using the event tree technique. Accident sequences are developed for initiating events affecting all operations including, to the extent, practical, retrieval operations.

Step 3--Repository systems analysis. The objective of step 3 is to develop the reliability models for the repository systems and support systems to be analyzed. This step obtains information from steps 1, 4, 5, and 6 as shown in Figure 8.3.5.5-3.

Table 8.3.5.5-2. Parameters required for Issue 2.3 (accidental radiological releases) (page 1 of 5)

| Related performance goal[a] | Performance or design parameter | Parameter descriptor | Tentative parameter goal | Needed confidence | Expected parameter value | Current confidence | SCP section providing parameter |
|---|---|---|---|---|---|---|---|
| A | Population density of region | Nye and Clark counties | Low population density | High | Section 3.6.2 in Environmental Assessment (DOE, 1986[b]) | Medium | 8.3.1.10 |
| B | Frequency and magnitudes of | | | | | | |
| | Tornadoes | At facility | (b) | High | Section 5.1.1.6 (see footnote c) | Medium | 8.3.1.12 |
| | Cloud-to-ground lightning strikes | At facility | (b) | Medium | About 18/yr, magnitude unknown | Medium | 8.3.1.12 |
| | Sandstorms and wind-storms | At facility | (b) | High | Table 5-8, and Section 5.1.1.6 | Medium | 8.3.1.12 |
| | Snow fall and ice storms | At facility | Rare, low magnitude | High | Rare, low magni-tude | Medium | 8.3.1.12 |
| | Repository surface flooding | At facility | PMF[d] | High | PMF[d] | Medium | 8.3.1.16 |
| | Surface and sub-surface seismic events | In region | (b) | High | (e) | (e) | 8.3.1.17 |
| | Fault movement within the repository | Surface and subsurface | (b) | High | Section 1.5.2 | Medium | 8.3.1.17 |
| | Drift roof fall and collapse or failure | Underground | (f) | Medium | Data not available | Data not avail-able | 8.3.2.4 |
| | Landslides | At facility | (f) | Medium | Data not available | Data not avail-able | 8.3.1.14 |



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Table 8.3.5.5-2. Parameters required for Issue 2.3 (accidental radiological releases) (page 2 of 5)

| Related performance goal[a] | Performance or design parameter | Parameter descriptor | Tentative parameter goal | Needed confidence | Expected parameter value | Current confidence | SCP section providing parameter |
|---|---|---|---|---|---|---|---|
| | Volcanic ash fall | At facility | (f) | Medium | Data not available | Data not avail-able | 8.3.1.17 |
| | Nearby brush fires | Near facilities | (f) | Low | Data not available | Data not avail-able | 8.3.1.13 |
| | Aircraft crashes | At facility | (f) | High | 1 x $10^{-5}$ to 1 x $10^{-7}$ per year | Medium | 8.3.1.13 |
| | Criticality events | In surface and subsurface | Criticality events precluded | High | Not credible | High[g] | 8.3.5.5 |
| | Other potential accidents | Natural or site-related | (h) | High | (h) | (h) | PRAM program[i] |
| C,I | Wind speeds | 80 km radius | (b) | High | Figures 5-3 to 5-7, and Tables 5-6 and 5-7 | Medium | 8.3.1.12 |
| C,I | Wind direction | 80 km radius | (b) | High | Figures 5-3 to 5-7, and Tables 5-6 and 5-7 | Medium | 8.3.1.12 |
| C,I | Atmospheric stability | 80 km radius | (b) | Medium[j] | Table 5-11 | Medium | 8.3.1.12 |
| C,I | Mixing layer depth | 80 km radius | (b) | Medium | (k) | Medium | 8.3.1.12 |
| C,I | Average ambient temperature | 80 km radius | (b) | Medium | Tables 5-2 and 5-3 | Medium | 8.3.1.12 |
| C,I | Atmospheric moisture | 80 km radius | (b) | Medium | Tables 5-2 and 5-5 | Medium | 8.3.1.12 |
| C,I | Precipitation type, amount, intensity, etc. | 80 km radius | (b) | Medium | Tables 5-2 and 5-4 | Medium | 8.3.1.12 |

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Table 8.3.5.5-2. Parameters required for Issue 2.3 (accidental radiological releases) (page 3 of 5)

| Related performance goal[a] | Performance or design parameter | Parameter descriptor | Tentative parameter goal | Needed confidence | Expected parameter value | Current confidence | SCP section providing parameter |
|---|---|---|---|---|---|---|---|
| C,I | Barometric pressure | 80 km radius | (b) | Medium | Table 5-2 | Medium | 8.3.1.12 |
| C,I | Size and distance of topographic features from release points | 80 km radius | Topographic features beneficial to dispersion | Medium | See U.S. Geological Survey topographic maps | High | Literature |
| D | Bioaccumulation of radionuclides in terrestrial flora | 80 km radius | (b) | Medium | $1 \times 10^{-28}$ to $1 \times 10^{-14}$ Ci/kg (see footnote m) | Medium | (1) |
| D | Bioaccumulation of radionuclides in terrestrial fauna | 80 km radius | (b) | Medium | $1 \times 10^{-25}$ to $1 \times 10^{-15}$ Ci/kg (see footnote n) | Medium | (1) |
| D | Types of crops raised | 80 km radius | (b) | Medium | (o) | Medium | (1) |
| D | Amounts of crops raised | 80 km radius | (b) | Medium | $1 \times 10^{4}$ to $1 \times 10^{7}$ kg/yr (see footnote p) | Medium | (1) |
| D | Types of crops consumed | 80 km radius | (b) | Medium | (q) | Medium | (1) |
| D | Amounts of crops consumed | 80 km radius | (b) | Medium | $1 \times 10^{4}$ to $1 \times 10^{5}$ kg/yr | Medium | (1) |
| D | Types of animals raised | 80 km radius | (b) | Medium | (r) | Medium | (1) |
| D | Number of animals raised | 80 km radius | (b) | Medium | $1 \times 10^{1}$ to $1 \times 10^{5}$ kg/yr | Medium | (1) |
| D | Types of animals consumed | 80 km radius | (b) | Medium | (s) | Medium | (1) |
| D | Amounts of meat consumed | 80 km radius | (b) | Medium | $1 \times 10^{4}$ to $1 \times 10^{6}$ kg/yr | Medium | (1) |
| D | Animal consumption of forage | 80 km radius | (b) | Medium | $1 \times 10^{1}$ to $1 \times 10^{4}$ kg/yr | Medium | (1) |



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Table 8.3.5.5-2. Parameters required for Issue 2.3 (accidental radiological releases) (page 4 of 5)

| Related performance goal[a] | Performance or design parameter | Parameter descriptor | Tentative parameter goal | Needed confidence | Expected parameter value | Current confidence | SCP section providing parameter |
|---|---|---|---|---|---|---|---|
| D | Forage storage time | 80 km radius | Values given in Regulatory Guide 1.109 (NRC, 1977a) | Medium | Data not available | Data not available | (1) |
| D | Grazing yield and period | 80 km radius | (b) | Medium | 75 to 100% of the year | High | (1) |
| D | Radius of crop and animal area | 80 km radius | (b) | Medium | 50 km to bulk of cropland and farms (W to SW) | High | (1) |
| D | Volumetric flow of surface water to water bodies | 80 km radius | Little or no surface runoff | Medium | Environmental Assessment Section 3.3.1 | Medium | (1) |
| D | Population served by local drinking water | 80 km radius | (b) | Medium | $1 \times 10^2$ to $1 \times 10^4$ | Medium | (1) |
| D | Volumetric flow of local drinking water | 80 km radius | (b) | Low | Section 3.3.1 in Environmental Assessment (DOE, 1986b) | Medium | (1) |
| D | Recreational uses of water bodies | 80 km radius | Very little recreational use of water | Low | (t) | (t) | (1) |
| E,F,G,H,I, J,K | Reference repository design, operating plan, and supporting analysis | No additional site characterization data needed--see footnote u. | | | | | |

[a]The letters in this column key the performance parameters on this table to the tentative performance goals in Table 8.3.5.5-1.
[b]Tentative goal is to have further measurements of this parameter verify the range of expected values listed here.
[c]Probability at Yucca Mountain is approximately $7.5 \times 10^{-4}$ in any given year; magnitude is F-0 on Fujita tornado scale (very weak).



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Table 8.3.5.5-2. Parameters required for Issue 2.3 (accidental radiological releases) (page 5 of 5)

Footnotes (continued)

[d]PMF = probable maximum flood; the PMF is still under investigation.

[e]Information on seismic events may be found in "Ground Motion Evaluation at Yucca Mountain, Nevada, with Application to Repository Conceptual Design and Siting," (URS/Blume, 1986).

[f]Parameter goal to be evaluated in terms of frequency and consequence.

[g]Design will preclude criticality accidents per 10 CFR 60.131(b)(7).

[h]Other accident-specific goals to be evaluated as appropriate under preclosure risk assessment methodology.

[i]PRAM = preclosure risk assessment methodology.

[j]Medium confidence requirements are intended to indicate that these parameters need to be site-specific.

[k]See Quiring (1968).

[l]Collection of these data are part of the environmental program planned activities and is addressed in the Radiological Monitoring Plan discussed in Section 8.3.1.13.

[m]This range covers all flora for which data are now available; specific values are flora and radionuclide specific.

[n]This range covers all fauna for which data are now available; specific values are fauna and radionuclide specific.

[o]Wheat/grains, corns, apples, potatoes, alfalfa, alfalfa seed, hay, silage, peppers, melons, berries, pecans, leafy vegetables, and honey.

[p]Specific values depend on available crops, crop areas, and crop densities.

[q]Includes all crops listed in footnote o except alfalfa, hay, and silage.

[r]Beef cattle, dairy cattle, goats, hogs, sheep, and poultry.

[s]Includes all animals listed in footnote r plus quail, freshwater fish, ducks, geese, rabbit, and deer.

[t]Very limited use of Crystal Reservoir; swimming pool data not yet available.

[u]For purposes of communicating the design information needed to evaluate worker radiological safety under normal conditions, the input items from Issue 4.4 (obtained through Issue 2.7) are collectively listed as parameter.



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Step 4--Human reliability analysis. The objectives of step 4 are to (1) identify the human errors that should be included in the preclosure risk assessment, (2) provide the probability estimates for these errors, and (3) develop human recovery actions to mitigate accident consequences.

Step 5--Common cause failure analysis. The objective of step 5 is to identify the failures of multiple equipment items occurring from a single cause that is common to all the equipment items; for example, the loss of electric power may cause the failure of several repository systems.

Step 6--Data base development. The objective of step 6 is to develop the data bases for the analytical steps of the preclosure risk assessment. The data base will provide data for use in steps 3, 4, 5, 7, and 8.

Step 7--Accident sequence analysis. The objective of step 7 is to quantify the frequency of occurrence of the accident sequences developed in step 3 by linking system logic models from step 3 and using data from step 6.

Step 8--In-plant consequence analysis. The objective of step 8 is to determine accident sequence consequences within the repository site boundary, including the surface and underground facilities.

Step 9--Environmental transport and offsite consequence analysis. The objective of step 9 is to determine accident sequence consequences outside the repository boundary. The consequences include radiation doses to the public.

Step 10--Uncertainty, sensitivity, and importance analyses. There are three objectives in step 10. The first objective is to estimate the uncertainty in the results due to the parameters, modeling, and completeness at the various analytical steps of the risk assessment. The second objective is to determine how much the results of the analyses change with respect to variation of the input data. This is needed to perform the uncertainty analyses. The final objective is to identify and rank the important accident sequences, system failures, component failures, and human errors with regard to the accident sequence frequency of occurrence estimates. This importance analysis will be used in the identification of systems, structures, and components important to safety.

Step 11--Documentation and use of results. The objective of step 11 is to document the risk assessment methodology and results to support the various repository program activities and the resolution of this issue.

Performance evaluation for compliance with goals. The remainder of Figure 8.3.5.5-2 deals with the final evaluation of the results documented in the accident risk assessment package. The results are compared with the performance goals and any regulatory limits that may be developed. If the goals or limits have been met and if the design is in the final design phase, then the design is ready for license application and a favorable issue resolution has been achieved. If the goals or limits have been met but the design is not in the final design phase, then this process is repeated for the next design phase.

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If the results of the accident risk assessment package do not meet the goals or limits, then design, procedural, or operational changes are recommended to correct the situation. If these changes cannot be made and the performance goals cannot be reasonably changed, then an unfavorable resolution of the issue has occurred. However, if the design, procedural, or operational changes can be made or the performance goals can be reasonably changed, then the recommended changes are implemented and the whole process is repeated.

Interrelationships of information needs

The question asked by this issue addresses the potential threat to health and safety of essential repository workers and the public from radiological accidents at the repository. The resolution of this issue can be obtained by answering two questions:

1. What are the credible accident sequences and their respective frequencies that can occur at the repository that can adversely affect the health and safety of the workers and the public?

2. What are the predicted releases of radioactive material and the projected public and essential worker exposures resulting from credible accidents at the repository, and are these exposures within applicable limiting values?

These questions are addressed by Information Needs 2.3.1 and 2.3.2. Information Need 2.3.1 determines and describes the possible accidents that could occur at the repository. Once the list of accidents is developed, the list is screened to determine those that are both applicable to the repository and credible. This information need requires both site and design data to determine all credible natural, site-related, and design-related accidents. This process is described previously in the section on accident risk assessment package. The final product of Information Need 2.3.1 is a list of credible accidents along with their frequencies of occurrence and resulting scenarios. This information need corresponds to the system analysis steps in assessing accident risks discussed previously.

This set of accident sequences is used under Information Need 2.3.2 to predict essential worker and public exposures resulting from accidents. First, the projected releases of radioactive material resulting from the credible accidents are determined, which requires detailed information about the repository design as well as information about the characteristics of the accidents. This step will require very little site data. The major factors affecting releases of radioactive material during accidents are the source terms present and the response of the repository structures and systems. It is here that information about which systems, structures, and components are important to safety will be developed and refined. When the releases of radioactive material have been determined, the process of resolving Issue 2.3 continues with the determination of essential worker and public radiation exposures. Information Need 2.3.2 determines the radiation exposures for the repository workers and the public due to accidents and compares the results to applicable limiting values. This step will require a great deal of site data to perform the necessary radionuclide transport calculations. Along with the exposure values, there will be a frequency of occurrence associated with each accident. The combination of the frequency of occurrence and the consequence defines the risk for a given accident. The sum of all the accident risks defines the repository risk. Accident risk quantification (sensitivity analyses and documentation) is the responsibility of this information need. The final results of the accident risk assessment package will be documented as part of this information need. At this point, this issue is finally resolved.

The functions and performance measures (associated with the MGDS system elements) necessary for answering these two questions and resolving this issue are listed in Table 8.3.5.5-1. The site data needed to answer these questions are listed in Table 8.3.5.5-2.

### 8.3.5.5.1 Information Need 2.3.1: Determination of credible accident sequences and their respective frequencies applicable to the repository

Technical basis for addressing the information need

Link to the technical data chapters and applicable support documents

Chapter 3 discusses the present state of the knowledge on the site hydrology, including uses of surface water and ground water. Chapter 5 contains discussions about the present state of the knowledge on the meteorology of the site and surrounding region. Further discussions on the subject of radiological protection of the public may be found in Sections 6.1.1.4.1 (radiological protection design requirements), Section 6.1.4 (items important to safety) and 6.4.6 (Issue 2.3: accidental radiological releases). Section 8.3.5.1 discusses the preclosure risk assessment methodology (PRAM) program. The PRAM program includes radiological risk to the public and workers under accident conditions as part of its scope. Sections 2.5 (radiological protection) and 6.2 (releases under abnormal conditions) of the site characterization plan-conceptual design report (SCP-CDR) (SNL, 1987) also contain discussions relevant to this issue. Also, Sections 4.6.1 and 7.4.1 of the SCP-CDR contain brief discussions of items important to safety. Finally, two appendices of the SCP-CDR have information especially relevant to this information need. These are Appendix F (preclosure radiological safety analysis), which is a preliminary analysis of accidents at the Yucca Mountain repository prepared to support the development of the preliminary Yucca Mountain Project Q-list, and Appendix L (items important to safety and retrievability), which discusses the method used and results of the preliminary Q-list. The methodology described in Appendix F of the SCP-CDR will be considered by the PRAM program.

Parameters

The parameters required by this information need are those parameters relevant to the determination of credible accident sequences and their respective frequencies for the Yucca Mountain repository. There is a great deal of design information required for this purpose; this information is listed in Table 8.3.5.5-2. Reference repository design information and supporting analyses will be obtained from the reference information base (RIB).

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The site data required to determine credible site-related accidents are obtained through various characterization programs. A summary of the required site data and the associated investigation or information need follows.

| Data requirement (frequency and characteristics) | SCP section |
|---|---|
| EVENTS ON THE SITE | |
| Tornadoes | 8.3.1.12.4 |
| Cloud-to-ground lightning strikes | 8.3.1.12.4 |
| Sandstorms | 8.3.1.12.4 |
| Snowfall | 8.3.1.12.4 |
| Ice storms | 8.3.1.12.4 |
| Repository surface flooding | 8.3.1.16.1, 8.3.1.16.3 |
| Repository flooding from ground-water inflow | 8.3.1.16.1, 8.3.1.16.3 |
| Surface and subsurface seismic events | 8.3.1.17.3 |
| Fault movement within the repository | 8.3.1.17.2 |
| Drift roof fall, collapse, or failure | 8.3.2.4.1 |
| Surface landslides | 8.3.1.14.1 |
| Volcanic ash fall | 8.3.1.17.1 |
| Nearby forest or brush fires | 8.3.1.13.1, 8.3.1.13.2 |
| Aircraft and helicopter crashes in the area of the surface facilities | 8.3.1.13.1, 8.3.1.13.2 |
| Other potential accidents | Preclosure risk assessment methodology (PRAM) program, 8.3.1.13.1, 8.3.1.13.2 |

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| Data requirement (frequency and characteristics) | SCP section |
|---|---|
| OFFSITE INSTALLATION ACCIDENTS IN THE REGION | |
| Explosive shockwave | 8.3.1.13.1, 8.3.1.13.2 |
| Toxic and chemical gases | 8.3.1.13.1, 8.3.1.13.2 |
| Missiles | 8.3.1.13.1, 8.3.1.13.2 |
| Flammable vapor clouds | 8.3.1.13.1, 8.3.1.13.2 |
| Incendiary fragments | 8.3.1.13.1, 8.3.1.13.2 |

Logic

The determination of credible accidents for the Yucca Mountain repository requires a great deal of site and design information. Site data are required to determine site-related accidents. These data include severe weather phenomena, seismic phenomena, tectonic phenomena, offsite installation activities, and military activities. Using the site data, the frequencies and characteristics of these various phenomena and activities can be assessed, and a decision can be made as to what accidents are credible. The definition of credible has not yet been firmly established; however, it is expected that a credible accident will be defined in terms of frequency of occurrence. In some instances, where an accident may have extremely severe consequences, an accident with a very low frequency of occurrence may be included in those to be analyzed. Therefore, it is possible that, for conservatism, the severity of an accident could be a factor in deciding whether to classify an accident as credible or not credible. Design-related accidents will also be investigated, and site data are needed to determine the consequences of these accidents; however, accident consequences are developed as part of Information Need 2.3.2. The function of this information need corresponds to the system analysis steps shown in Figure 8.3.5.5-3. Once a list of credible accident sequences and their respective frequencies applicable to the Yucca Mountain repository is developed, the process of analyzing the effects of these accidents on the repository is continued in Information Need 2.3.2, where radioactive material releases resulting from these accidents are estimated.

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8.3.5.5.1.1 Performance Assessment Activity 2.3.1.1: Refinement of site data parameters required for Issue 2.3

Objectives

The objective of this activity is to refine the list of site data parameters presented in the technical basis for this information need. This list may be incomplete or the level of confidence (Table 8.3.5.5-2) may be inappropriate.

Parameters

The list of parameters presented earlier is the starting point for this activity. This list may be added to or shortened as work in this area progresses.

Description

As the accident risk assessment activities progress, more information may be required to better define accidents or their characteristics. In addition, feedback from the site characterization program will be an important source of information about parameter confidence requirements and may result in identification of more parameters that are needed.

8.3.5.5.1.2 Performance Assessment Activity 2.3.1.2: Determination of credible accident sequences and their respective frequencies applicable to the Yucca Mountain repository

Objectives

The objective of this activity is to develop a comprehensive list of accidents that are both credible and applicable to the Yucca Mountain repository.

Parameters

The site data parameters required for this task are those listed earlier for the information need. A great deal of design data will also be required to perform this activity.

Description

This activity consists of performing the system analysis steps shown in Figure 8.3.5.5-3 and discussed in detail in Section 8.3.5.1. Note that uncertainty, sensitivity, and importance analyses and documentation are included in these steps.

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8.3.5.5.1.3 Performance Assessment Activity 2.3.1.3: Development of candidate design-basis accidents for the Yucca Mountain repository

Objectives

The objective of this activity is to develop a set of candidate design-basis accidents to be analyzed as part of the total safety analysis.

Parameters

The parameters to be used in this analysis are those listed earlier for the information need and the repository reference design.

Description

A set of design-basis accidents will be developed to be analyzed as part of the total safety analysis of the repository. The procedure to be used in developing this set of accidents has not yet been established; however, the PRAM program will address this need. The development of the list of credible accidents and the development of design-basis accidents are complementary and will be performed in parallel. Design-basis accidents do not necessarily have to be credible; indeed, they are generally less likely than what are usually considered credible events. Design-basis accidents are proposed to show that the repository response to these accidents is acceptable, and, therefore, the repository can be expected to withstand any expected or credible accidents.

8.3.5.5.2 Information Need 2.3.2: Determination of the predicted releases of radioactive material and projected public and worker exposures under accident conditions and that these exposures meet applicable requirements

Technical basis for addressing the information need

Link to the technical data chapters and applicable support documents

The data chapters and technical support documents for this information need are the same as those listed for Information Need 2.3.1 (see Section 8.3.5.5.1).

Parameters

The parameters required by this information need are those parameters relevant to the determination of the radioactive material releases and the projected essential worker and public exposures resulting from the credible accidents developed in Information Need 2.3.1. The information required is mainly site data for public exposures and a mixture of site data and design data for worker exposures. The design information needed is listed in Table 8.3.5.5-2. Reference repository design information and supporting analyses will be obtained from the reference information base (RIB), which

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and will contain all design details necessary to perform the dose calculations to resolve this issue.

The site data required to determine radioactive material releases resulting from the credible accidents are obtained through various characterization issues. This information need uses mostly meteorological and agricultural data, which are given in the following summary.

The population density data required are given in Section 8.3.5.6. (Data will also be gathered under Characterization Program 8.3.1.10.)

The following table presents the data required, as well as the SCP section providing the information.

| Data requirement | SCP section |
|---|---|
| METEOROLOGICAL DATA | |
| Wind speeds | 8.3.1.12.1, 8.3.1.12.2 |
| Wind direction | 8.3.1.12.1, 8.3.1.12.2 |
| Atmospheric stability | 8.3.1.12.1, 8.3.1.12.2 |
| Mixing layer depth | 8.3.1.12.1, 8.3.1.12.2 |
| Average ambient temperature | 8.3.1.12.1, 8.3.1.12.2 |
| Atmospheric moisture | 8.3.1.12.1, 8.3.1.12.2 |
| Precipitation type, amount, intensity, etc. | 8.3.1.12.1, 8.3.1.12.2 |
| Size and distance of topographic features from releases points | 8.3.1.14.1 |
| Meteorological data for offsite installations | 8.3.1.12.1, 8.3.1.12.2 |
| AGRICULTURAL DATA | |
| Bioaccumulation of radionuclides in terrestrial flora | 8.3.1.13(a) |
| Bioaccumulation of radionuclides in terrestrial fauna | 8.3.1.13(a) |
| Types and amounts of crops raised | 8.3.1.13(a) |

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| Data requirement | SCP section |
|---|---|
| AGRICULTURAL DATA (continued) | |
| Types and amounts of crops consumed | 8.3.1.13(a) |
| Types and amounts of animals raised | 8.3.1.13(a) |
| Types and amounts of meat consumed | 8.3.1.13(a) |
| Animal consumption of forage | 8.3.1.13(a) |
| Forage storage time | 8.3.1.13(a) |
| Grazing yield and period | 8.3.1.13(a) |
| Radius of crop/animal area | 8.3.1.13(a) |
| Volumetric flow of surface water to water bodies | 8.3.1.13(a) |
| Population served and volumetric flow of drinking water | 8.3.1.13(a) |
| Recreational uses of water bodies | 8.3.1.13(a) |

[a]Collection of these data is part of the environmental program planned activities and is addressed in the Radiological Monitoring Plan discussed in Section 8.3.1.13.

Logic

This information need corresponds to the consequence analysis steps of Figure 8.3.5.5-3 and is discussed in detail in Section 8.3.5.1. The determination of releases of radioactive material requires detailed information about the repository design and the characteristics of accidents, but little site data. The determination of radionuclide transport and radiation exposure requires a great deal of site data. To calculate doses to essential workers, both design and site information is needed. For workers, the short-term pathways, which are dominated by atmospheric transport, are important. For the public, both short-term and long-term pathways must be evaluated. Meteorological data are needed to evaluate the short-term atmospheric transport pathways, and agricultural data are needed to evaluate the long-term transport pathways. To perform the uncertainty, sensitivity, and importance analyses, a great deal of design information and very little site data are needed. The final resolution of Issue 2.3 will take place after the safety assessment has been documented and a comparison with applicable limiting values is completed.

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Four performance assessment activities are planned and are discussed in the following sections.

8.3.5.5.2.1 Performance Assessment Activity 2.3.2.1: Refinement of site data parameters required for Issue 2.3

Objectives

The objective of this activity is to refine the list of site data parameters presented earlier for this information need. This list may be incomplete or the level of confidence (Table 8.3.5.5-2) may be inappropriate.

Parameters

The list of parameters presented earlier for this information need is the starting point for this activity. This list may be amended as work in this area progresses.

Description

As the accident risk assessment activities progress, more information may be required to better define accidents or their characteristics. In addition, feedback from the site characterization program will be an important source of information about parameter confidence requirements and may identify more parameters that are needed.

8.3.5.5.2.2 Performance Assessment Activity 2.3.2.2: Consequence analyses of credible accidents at the Yucca Mountain repository

Objectives

The objective of this activity is to determine the consequences of credible accidents in terms of radiation doses to the essential repository workers and the public.

Parameters

The site data parameters required to perform this activity are listed earlier for the information need. This activity will require little design data.

Description

This activity consists of performing the consequence analysis steps shown in Figure 8.3.5.5-3 and discussed in detail in Section 8.3.5.1. A determination of consequences involves calculations of radiation transport within the repository facilities and in the environment, radionuclide removal by repository systems and environmental systems, and doses to workers and the public. Radiation doses resulting from both short-term transport mechanisms (e.g., atmospheric transport) and long-term transport mechanisms (e.g.,

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through crops and animals) will be estimated. Consequences will also be calculated in terms of economic losses to the repository and to others. An example of the types of analyses that are to be performed can be found in Appendix F of the SCP-CDR (SNL, 1987).

8.3.5.5.2.3 Performance Assessment Activity 2.3.2.3: Sensitivity and importance analyses of credible accidents at the Yucca Mountain repository

Objectives

The objectives of this activity are (1) to quantify uncertainties and sensitivities in the accident risk assessment and (2) to establish importance rankings for systems, structures, and components of the repository with respect to radiological safety.

Parameters

The parameters required to perform this activity consist mainly of failure rate data for repository systems, structures, and components. Some site data will be required to quantify uncertainties of initiating event frequencies of occurrence and uncertainties in meteorological data.

Description

Quantifying uncertainties in the accident risk assessment analyses will require an extensive data base from which to draw statistical data. The development of these data bases is within the scope of the PRAM program as discussed in Section 8.3.5.1. Sensitivity analyses will be performed to establish important parameters. This work will help to refine the site data parameter lists for safety assessment early in the design process. By the time the design is ready for license application, most of the sensitivity and uncertainty work will have been completed. Importance analyses will be performed using computer codes to analyze the event trees developed in Performance Assessment Activity 2.3.2.2. These analyses will establish the importance of repository systems, structures, and components with respect to radiological safety and will be used to refine the Yucca Mountain Project Q-list.

8.3.5.5.2.4 Performance Assessment Activity 2.3.2.4: Documentation of results of safety analyses and comparison to applicable "limiting" values

Objectives

The objectives of this activity are (1) to produce documentation of the results of the accident risk assessment in the necessary format and (2) to make comparisons of these results to applicable limiting values. This activity will complete the resolution of this issue at the end of the license application design.

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Parameters

The only parameters required for this activity are the analyses and results of the preceding performance assessment activities.

Description

This activity consists of presenting the results of the accident risk assessment in a manner consistent with the needs of the NRC, the DOE, and the repository program in general. The PRAM program will recommend an annotated outline for the documentation of these analyses. Included in the documentation is a comparison of the results with any regulatory limits that may be established in the future. Currently, no regulatory limits exist for repository accident consequences. In the absence of regulatory limits, the PRAM program will recommend appropriate limiting values for both design-basis accidents and the credible accidents evaluated in the accident risk assessment. The completion of this activity will mark the final resolution of this issue and will supply written documentation of that resolution.

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### 8.3.5.6 Issue resolution strategy for Issue 2.5: Can the higher-level findings required by 10 CFR Part 960 be made for the qualifying condition of the preclosure system guideline and the qualifying and disqualifying conditions of the technical guidelines for population density and distribution, site ownership and control, meteorology, and offsite installations and operations?

Regulatory basis for the issue

The DOE has established a set of siting guidelines to be used as a basis for evaluating the suitability of potential repository sites during the site selection process.* These siting guidelines, which are set forth in 10 CFR Part 960, are separated into two categories: those that address postclosure conditions (10 CFR 960.4) and those that address preclosure conditions (10 CFR 960.5). The manner in which the siting guidelines must be addressed during the siting process is described in the DOE Implementation Guidelines (10 CFR 960.3).

The DOE's preclosure guidelines that relate to preclosure radiological safety under normal and anticipated operating conditions are the subject of this issue (2.5). These guidelines consist of a system guideline and four technical guidelines. The system guideline is concerned with the expected performance of the repository system as a whole during the period before permanent closure, while each technical guideline is concerned with the effect of some specific aspect of the site on the preclosure performance. Each preclosure technical guideline has one qualifying condition that must be met for a site to be acceptable. In addition, two of the technical guidelines have one or more disqualifying conditions; a site is unacceptable if any one of the disqualifying conditions is found to be present. The technical guidelines also identify favorable conditions and potentially adverse conditions that describe characteristics of the setting that, if present, could contribute to or detract from the postclosure performance of a site.

The Implementation Guidelines require that the qualifying and disqualifying conditions of the system and technical guidelines be evaluated and that specific findings be made for each condition at principal decision points in the siting process. These findings are stated in 10 CFR Part 960, Appendix III, and are shown in Table 8.3.5.6-1.

There are four levels of findings. Disqualifying and qualifying conditions both require a lower-level and a higher-level finding. Lower-level findings must be made to determine if a site may be nominated as suitable for characterization or recommended as a candidate site for characterization. Higher-level findings, however, must be made to determine if a site may be recommended for the development of a repository. Disqualifying conditions require Level 1 and Level 2 findings, and qualifying conditions require Level 3 and Level 4 findings. Each level has both a positive finding and a negative finding associated with it.

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*The passage of the Nuclear Waste Policy Amendments Act of 1987 (NWPAA, 1987) may impact the manner in which this process is implemented.

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Table 8.3.5.6-1. Findings for qualifying and disqualifying conditions

Disqualifying condition--lower-level findings

Level 1 (a) The evidence does not support a finding that the site is disqualified.

(b) The evidence supports a finding that the site is disqualified.

Disqualifying condition--higher-level findings

Level 2 (a) The evidence supports a finding that the site is not disqualified on the basis of that evidence and is not likely to be disqualified.

(b) The evidence supports a finding that the site is disqualified or is likely to be disqualified.

Qualifying condition--lower-level findings

Level 3 (a) The evidence does not support a finding that the site is not likely to meet the qualifying condition.

(b) The evidence supports a finding that the site is not likely to meet the qualifying condition, and therefore the site is disqualified.

Qualifying condition--higher-level findings

Level 4 (a) The evidence supports a finding that the site meets the qualifying condition and is likely to continue to meet the qualifying condition.

(b) The evidence supports a finding that the site cannot meet the qualifying condition or is unlikely to be able to meet the qualifying condition, and therefore the site is disqualified.

Table 8.3.5.6-2 shows the findings previously made for the qualifying and disqualifying conditions concerned with preclosure radiological safety. These findings and the evidence supporting them are given in the Yucca Mountain environmental assessment (DOE, 1986b). The available evidence was sufficient to support a positive higher-level finding for the first two disqualifying conditions of the population density and distribution technical guideline, and a positive lower-level finding for the remaining qualifying

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and disqualifying conditions. To determine if the Yucca Mountain site is suitable for the development of a repository, higher-level findings must be made for all the qualifying and disqualifying conditions.

Table 8.3.5.6-2. Preliminary findings for the qualifying and disqualifying condition concerned with preclosure radiological safety[a]

| Preclosure radiological safety guidelines qualifying and disqualifying conditions (10 CFR Part 960) | | Preliminary finding[b] |
|---|---|---|
| 960.5-1(a)(1) | System qualifying condition | Level 3(a) |
| 960.5-2-1 | Population density and distribution | |
| (a) | Qualifying condition | Level 3(a) |
| (d)(1) | Disqualifying condition 1 | Level 2(a) |
| (d)(2) | Disqualifying condition 2 | Level 2(a) |
| (d)(3) | Disqualifying condition 3 | Level 1(a) |
| 960.5-2-2 | Site ownership and control | |
| (a) | Qualifying condition | Level 3(a) |
| 960.5-2-3 | Meteorology | |
| (a) | Qualifying condition | Level 3(a) |
| 960.5-2-4 | Offsite installations and operations | |
| (a) | Qualifying condition | Level 3(a) |
| (d) | Disqualifying condition | Level 1(a) |

[a]Preliminary findings from DOE (1986b).
[b]See Table 8.3.5.6-1 for an explanation of the finding levels.

The DOE siting guidelines do not require any findings similar to lower-level or higher-level findings to be made for the favorable or potentially adverse conditions of the technical guidelines. As stated in the Supplementary Information (DOE, 1984b) for 10 CFR Part 960 (Overview of the Guidelines), these conditions were intended to be used to predict the suitability of a site and provide a preliminary indication of system performance before the start of detailed site characterization studies. These conditions were considered and used in the identification of potentially acceptable sites and in the nomination and recommendation of sites as suitable for characterization. By the completion of site characterization,

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however, sufficient data will be available to directly evaluate site performance against the qualifying conditions of the system and technical guidelines. Therefore, the favorable and potentially adverse conditions will not be considered in specific terms as they were for the environmental assessment (DOE, 1986b).

Approach to resolving the issue

To resolve Issue 2.5, sufficient evidence must be available to support either a positive or negative higher-level finding for each qualifying and disqualifying condition associated with preclosure radiological safety. Each of the qualifying conditions makes reference either directly or through the system guideline to regulatory requirements of the NRC (specifically, 10 CFR Part 60). To support a higher-level finding for the qualifying conditions, evidence must show whether the preclosure radiological releases under normal and projected operating conditions will be within the limits set by the NRC and the EPA, given the conditions that exist at the site. The system guideline looks at the site conditions as a whole, and the technical guideline looks at specific conditions. The disqualifying conditions are also related to NRC regulations, but not always as explicitly as the qualifying conditions.

Figure 8.3.5.6-1 shows the strategy for resolving Issue 2.5. The first step is to eliminate from further consideration the qualifying and disqualifying conditions for which higher-level findings have already been made. This is the case for the first two disqualifying conditions of the population density and distribution technical guideline. Next, for each remaining condition, it is determined whether the evidence presently available is sufficient to support a higher-level finding. This evidence consists of the information presented in the Yucca Mountain environmental assessment (DOE, 1986b) and in Chapters 1 through 7 of the SCP. If the evidence is sufficient, the finding and the evidence are documented.

For the qualifying and disqualifying conditions for which there is not adequate evidence available, the planned site characterization studies are reviewed to determine if the conditions will be investigated. This is accomplished by evaluating the resolution strategies of other preclosure radiological safety performance issues (Issues 2.1 and 2.2, Sections 8.3.5.3 and 8.3.5.4) that assess the ability of the site to comply with the NRC's preclosure radiological safety regulatory requirements under normal and anticipated operating conditions. As discussed previously, the qualifying and disqualifying conditions are linked to NRC regulatory requirements, and evidence to support a higher-level finding must show that the condition does not prevent compliance with the referenced requirements. Therefore, if the concerns of the qualifying and disqualifying conditions are being considered in the resolution strategies of the issues that assess compliance with the regulations, it can be expected that the evidence to support higher-level findings will be made available through the information and analyses that support resolution of these issues. A correlation of the qualifying and disqualifying conditions and the issues that will supply the information is shown in Table 8.3.5.6-3.

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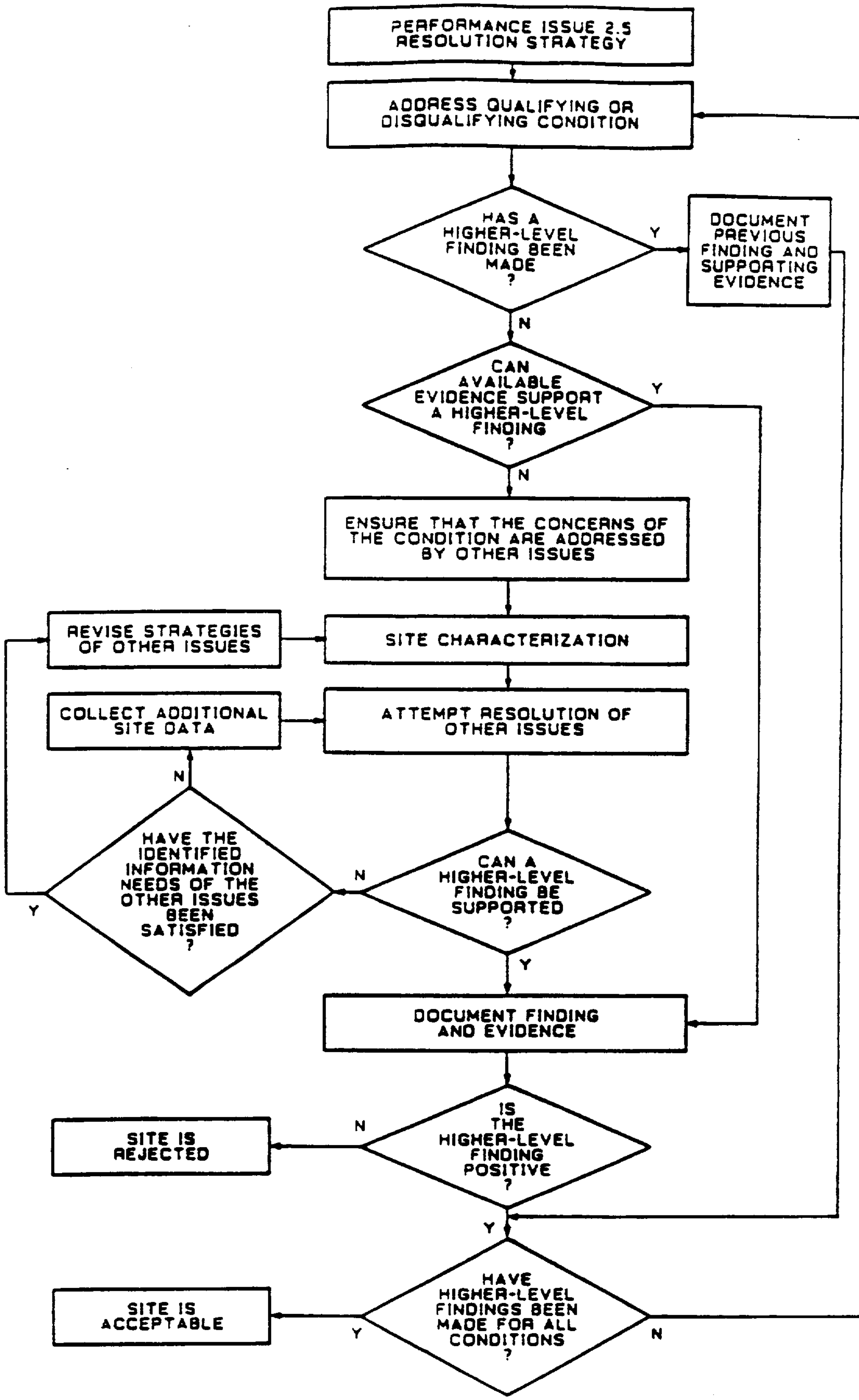


**Figure 8.3.5.6-1.** Issue resolution strategy for Issue 2.5 (higher-level findings-preclosure radiological safety).

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Table 8.3.5.6-3. Preclosure performance issues that address the concerns of the preclosure radiological safety qualifying and disqualifying conditions covered by Issue 2.5

| Guideline | Issue 2.1 (Public radiological exposures--normal conditions) | Issue 2.2 (Worker radiological safety--normal conditions) |
|---|---|---|
| Preclosure system guideline | | |
| Preclosure radiological safety qualifying condition | x | x |
| Population density and distribution | | |
| Qualifying condition | x | |
| Disqualifying condition 1[a] | | x |
| Disqualifying condition 2[a] | x | |
| Disqualifying condition 3[b] | | |
| Site ownership and control | | |
| Qualifying condition[b] | | |
| Meteorology | | |
| Qualifying condition | x | |
| Offsite installations and operations | | |
| Qualifying condition | x | |
| Disqualifying condition | x | |

[a]Higher-level findings have been made in the environmental assessment (DOE, 1986b); see Table 8.3.5.6-2.
[b]Not addressed by the issues and is outside the scope of site characterization and this document.

After ensuring that the qualifying and disqualifying conditions will be investigated, the information necessary to assess compliance will be obtained during site characterization. Upon completion of the assessments, the results will be evaluated to determine if sufficient evidence is available to support higher-level findings. If the evidence is sufficient, the findings and the evidence will be documented. If the evidence shows that a negative higher-level finding must be made for any one of the conditions, i.e., that a disqualifying condition is present or that a qualifying condition is not

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present, then the site will be disqualified. This evaluation will continue until positive higher-level findings can be supported for all the conditions or until a negative higher-level finding must be made.

If, in evaluating the results of the assessments, insufficient information is found to support either a positive or a negative higher-level finding for a qualifying or disqualifying condition, additional information may be necessary to satisfy existing information needs. Otherwise, the resolution strategies of the appropriate performance issues will be again reviewed to determine if, in fact, the condition was adequately considered. This process continues until there is sufficient evidence to support either a positive or negative higher-level finding for every qualifying and disqualifying condition. As discussed previously, findings are not required for the favorable conditions or the potentially adverse conditions at this stage in the siting process. However, the DOE's analysis indicates that the concerns of these conditions are adequately addressed through the analyses of other issues.

The following discusses the qualifying condition of the preclosure system guideline and the qualifying and disqualifying conditions of the preclosure technical guidelines that are concerned with preclosure radiological safety. The ties of each condition to the NRC regulations are explained, and the preclosure performance issue resolution strategies that will be relied upon are identified. The information relevant to each guideline, which will be collected during site characterization and used in the resolution of the other issues, is also given.

System guideline qualifying condition

The qualifying condition pertaining to preclosure radiological safety is stated in 10 CFR 960.5-1(a)(1) as follows:

> Any projected radiological exposures of the general public and any projected releases of radioactive materials to restricted and unrestricted areas during repository operation and closure shall meet the applicable safety requirements set forth in 10 CFR Part 20, 10 CFR Part 60, and 40 CFR 191, Subpart A....

This qualifying condition is concerned with the amounts of radioactive material that may be released to the environment before and during permanent closure. The DOE distinguishes between the restricted and unrestricted environment. The restricted area is defined in 10 CFR 960.2 as "any area access to which is controlled by the DOE for purposes of protecting individuals from exposure to radiation and radioactive materials before repository closure...." The unrestricted area is everything outside the restricted area.

10 CFR 60.111 is the NRC performance objective that addresses the performance of the geologic repository operations area through permanent closure. Part (a) of this objective states that radiation exposures and radiation levels through permanent closure must be maintained within the limits specified in 10 CFR Part 20 and any that may be established by the EPA. Therefore, to make a higher-level finding for the qualifying condition

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of the system guideline for preclosure radiological safety, the ability of the site to comply with 10 CFR 60.111(a) must be determined.

The requirements of 10 CFR 60.111(a), 10 CFR Part 20, and 40 CFR 191, Subpart A, are addressed by two preclosure performance issues. Issue 2.1, which is discussed in Section 8.3.5.3, is concerned with projected releases to unrestricted areas and is stated as follows:

> During repository operation and closure, (a) will the expected average radiation dose to members of the public within any highly populated area be less than a small fraction of the allowable limits, and (b) will the expected radiation dose received by any member of the public in an unrestricted area be less than the allowable limits as required by 10 CFR 60.111, 40 CFR 191, Subpart A, and 10 CFR Part 20?

Issue 2.2, which is discussed in 8.3.5.4, is concerned with projected releases to restricted areas and is stated as follows:

> Can the repository be designed, constructed and operated in a manner that ensures the radiological safety of workers under normal operations as required by 10 CFR 60.111 and 10 CFR Part 20?

It is the judgment of the DOE that these two issues adequately cover the concerns of the system guideline for preclosure radiological safety. The information and analyses required to support resolution of these issues will thus provide sufficient evidence to support a higher-level finding for the qualifying condition of the system guideline. The details of the issue resolution strategies for these two issues and the information that will be collected during site characterization to resolve the issues are given in Sections 8.3.5.3 and 8.3.5.4. No information beyond that described in the two sections is expected to be required.

Population density and distribution technical guideline

The population density and distribution technical guideline has one qualifying condition and three disqualifying conditions.

Qualifying condition. The qualifying condition for the population density and distribution technical guideline is stated in 10 CFR 960.5-2-1(a) as follows:

> The site shall be located such that, during repository operation and closure, (1) the expected average radiation dose to members of the public within any highly populated area will not be likely to exceed a small fraction of the limits allowable under the requirements specified in 960.5-1(a)(1), and (2) the expected radiation dose to any member of the public in an unrestricted area will not be likely to exceed the limit allowable under the requirements specified in 960.5-1(a)(1).

This qualifying condition is concerned with radioactive releases to unrestricted areas and subsequent doses to members of the general public and

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is virtually identical to Issue 2.1 (public radiological exposures--normal conditions), which asks the following:

> During repository operation and closure, (a) will the expected average radiation dose to members of the public within any highly populated area be less than a small fraction of the allowable limits, and (b) will the expected radiation dose received by any member of the public in an unrestricted area be less than the allowable limits as required by 10 CFR 60.111, 40 CFR 191, Subpart A, and 10 CFR Part 20?

The resolution of Issue 2.1 will thus provide sufficient evidence to support a higher-level finding for the qualifying condition. The issue resolution strategy for Issue 2.1 and the information that will support resolution of Issue 2.1 are described in detail in Section 8.3.5.3, and this information is summarized in Table 8.3.5.6-4. No information beyond that described in Section 8.3.5.6 is expected to be required to support a higher-level finding.

Disqualifying condition 1. The first of the three disqualifying conditions of the population density and distribution technical guideline (10 CFR 960.5-2-1(d)(1)) states that a site will be disqualified if

> (1) Any surface facility of a repository would be located in a highly populated area.

The proximity of the site to highly populated areas greatly affects the extent of exposures of the general public during the preclosure period. It is desirable to locate a repository away from highly populated areas to limit doses to members of the public. A positive higher-level finding has been made for this condition. The finding and the supporting evidence are given in the Yucca Mountain environmental assessment (DOE, 1986b).

Disqualifying condition 2. The second of the three disqualifying conditions of the population density and distribution technical guideline (10 CFR 960.5-2-1(d)(2)) states that a site will be disqualified if

> (2) Any surface facility of a repository would be located adjacent to an area 1 mile by 1 mile having a population of not less than 1,000 individuals as enumerated by the most recent U.S. census.

As with the first disqualifying condition, a positive higher-level finding has been made for this condition. The finding and the supporting evidence are given in the Yucca Mountain environmental assessment (DOE, 1986b).

Disqualifying condition 3. The third of the three disqualifying conditions of the population density and distribution technical guideline (10 CFR 960.5-2-1(d)(3)) states that a site shall be disqualified if

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(3) The DOE could not develop an emergency preparedness program which meets the requirements specified in DOE Order 5500.3 (Reactor and Non-Reactor Facility Emergency Planning, Preparedness, and Response Program for Department of Energy Operations) and related guides or, when issued by the NRC, in 10 CFR Part 60, Subpart I, Emergency Planning Criteria.

The development of an emergency preparedness program is outside the scope of site characterization and the SCP. The development of such a plan will be discussed in Section 8.3.5.3.

Table 8.3.5.6-4. Information used in the resolution of Issue 2.1 (adapted from Table 8.3.5.3-2)

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Population density and distribution

- Distance from highly populated areas
- Population located in adjacent 1-mile by 1-mile area
- Population density of the region

Radionuclide concentration in environmental media and individual doses

- Bioaccumulation of radionuclides in terrestial flora
- Bioaccumulation of radionuclides in terrestial fauna
- Types and amounts of crops raised and consumed
- Types and amounts of animals raised and consumed
- Annual consumption of forage
- Forage storage time
- Grazing yield and period
- Radius of crop and animal area
- Volumetric flow of surface water to water bodies
- Population served by local drinking water
- Volumetric flow of local drinking water
- Recreational uses of water bodies

Meteorological information

- Windspeeds
- Wind direction
- Atmospheric stability
- Mixing layer depth
- Average ambient temperature
- Atmospheric moisture
- Precipitation: type, amount, intensity, etc.
- Barometric pressure
- Size and distance of topographic features from release points

Reference repository design and supporting analyses

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Offsite installations and operations

Location of nearby uranium fuel cycle facilities
Radionuclides normally released from nearby uranium cycle facilities

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Site ownership and control technical guideline

The site ownership and control technical guideline has one qualifying condition and no disqualifying conditions. The qualifying condition (10 CFR 960.5-2-2(a)) requires that

> The site shall be located on land for which the DOE can obtain, in accordance with the requirements of 10 CFR 60.121, ownership, surface and subsurface rights, and control of access that are required in order that surface and subsurface activities during repository operation and closure will not be likely to lead to radionuclide releases to an unrestricted area greater than those allowable under the requirements specified in 960.5-1(a)(1).

This qualifying condition is concerned with the ability of the DOE to control the use of the land within which a geologic repository system is located. Inability to control such use would affect the boundary of the unrestricted area, and therefore could affect releases to the unrestricted area. Lack of such control could also lead to a disruption of repository activities.

The ability of the DOE to obtain ownership, surface and subsurface rights, and control of access in accordance with the requirements of 10 CFR 60.121 is an institutional question that is outside the scope of the SCP. Instead, this subject will be addressed in future environmental program planning activities (see Section 8.3.1.11).

Meteorology technical guideline

The meteorology technical guideline has one qualifying condition and no disqualifying conditions. The qualifying condition (10 CFR 960.5-2-3(a)) requires that

> The site shall be located such that expected meteorological conditions during repository operation will not be likely to lead to radionuclide releases to an unrestricted area greater than those allowable under the requirements specified in 960.5-1(a)(1).

This qualifying condition is concerned with the effect of meteorological conditions only on releases to unrestricted areas. The releases expected at a site, given the meteorological conditions, must be within the limits set for releases to unrestricted areas.

The determination of whether releases to unrestricted areas are within allowable limits is addressed by Issue 2.1 (Section 8.3.5.3). The allowable limits are those referenced by 10 CFR 60.111 (10 CFR Part 20 and 40 CFR 191,

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Subpart A). To provide the information necessary to make a higher-level finding for this qualifying condition, the evaluation of releases to unrestricted areas must take into account the meteorological conditions expected at the site during the preclosure period. Table 8.3.5.6-4 lists the data identified through the resolution strategy for Issue 2.1, including meteorological data, that will be obtained during site characterization. Through the resolution of Issue 2.1, therefore, information is expected to be available to determine if expected meteorological conditions at the site will result in radiological releases to unrestricted areas greater than the allowable limits. This information will be sufficient to support a higher-level finding for this qualifying condition.

Offsite installations and operations

The offsite installations and operations technical guideline has one qualifying condition and one disqualifying condition.

Qualifying condition. The qualifying condition is stated in 10 CFR 960.5-2-4(a) as follows:

> The site shall be located such that present projected effects from nearby industrial, transportation, and military installations and operations, including atomic energy defense activities, (1) will not significantly affect repository siting, construction, operation, closure, or decommissioning or can be accommodated by engineering measures and (2), when considered together with emissions from repository operation and closure, will not be likely to lead to radionuclide releases to an unrestricted area greater than those allowable under the requirements specified in 960.5-1(a)(1).

Offsite installations and operations can affect required preclosure activities and the preclosure performance of a repository system in two ways: (1) the routine or anticipated activities associated with such operations or installations could interfere with or disrupt repository development, and (2) the offsite installations or operations could be releasing radioactive material to unrestricted areas that, when combined with the expected releases from repository operations, could result in total releases to unrestricted areas that are greater than the allowable limits. The first part of the qualifying condition is concerned with the potential for offsite installations and operations to significantly disrupt repository development and operations. The effects of offsite installations and operations on repository operations are being evaluated to establish the normal and anticipated conditions under which the repository will operate. For example, the effects of ground motion due to weapons testing at the Nevada Test Site will be investigated and the necessary measures taken to accommodate such motion. This investigation of normal and anticipated operating conditions is discussed in the resolution strategies of Issues 2.1 and 2.2 (public and worker radiological safety--normal conditions). The second part of the qualifying condition is concerned with total combined releases to unrestricted areas from offsite installations and operations. The combined total radionuclide releases to unrestricted areas under normal and anticipated operational conditions will be evaluated in resolving Issue 2.1. The NRC requires that, in calculating combined total releases, only releases from nuclear-fuel-cycle facilities need to be considered. However, through monitoring to establish

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background radiation levels at the site, releases from all other types of offsite installations and operations, such as the Nevada Test Site, will be determined. The evaluations and information obtained to resolve Issue 2.1 are therefore expected to be sufficient to support a higher-level finding for the qualifying condition of the offsite installations and operations technical guideline.

Disqualifying condition. The offsite installations and operations technical guideline disqualifying condition is stated in 10 CFR 960.5-2-4(d) as follows:

> A site shall be disqualified if atomic energy defense activities in proximity to the site are expected to conflict irreconcilably with repository siting, construction, operation, closure, and decommissioning.

This condition is the inverse of the first part of the qualifying condition. As discussed previously, the existence of offsite installations and operations, including those activities related to atomic energy defense, could conflict with or disrupt the activities required for repository development and operation, or they could result in total combined releases such that the applicable limits would be exceeded. Issue 2.1 will investigate the effects of offsite installations and operations on preclosure radiological safety (see the previous discussion of the qualifying condition). These investigations will provide the information necessary to support a higher-level finding for this disqualifying condition.

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8.3.5.7 Issue resolution strategy for Issue 4.1: Can the higher-level findings required by 10 CFR Part 960 be made for the qualifying condition of the preclosure system guideline and the disqualifying and qualifying conditions of the technical guidelines for surface characteristics, rock characteristics, hydrology, and tectonics?

Regulatory basis for the issue

The DOE has established a set of siting guidelines to be used as a basis for evaluating the suitability of potential repository sites during the site selection process.* These siting guidelines, which are set forth in 10 CFR Part 960, are separated into two categories: those that address postclosure conditions (10 CFR 960.4) and those that address preclosure conditions (10 CFR 960.5). The manner in which the siting guidelines must be addressed during the siting process is described in the DOE Implementation Guidelines (10 CFR 960.3).

DOE's preclosure system guideline and technical guidelines related to ease and cost of construction are the subject of this issue (4.1). These guidelines consist of a system guideline and four technical guidelines. The system guideline is concerned with the technical feasibility and relative cost of siting, constructing, operating, and closing a repository at a given site. Specific concerns are whether special engineering measures beyond the bounds of reasonably available technology may be necessary for repository construction, operation, and closure, and whether the cost of repository construction, operation, and closure may be unreasonable in comparison with the other repository siting options if a large number of special measures were necessary for these phases. The Nuclear Waste Policy Amendments Act of 1987 (NWPAA, 1987) identified the Yucca Mountain site as the only site to be characterized. As a consequence, the requirement for comparative evaluation of costs is no longer appropriate.

Each technical guideline is concerned with the effect of some specific aspect of site conditions on the concerns expressed in the system guideline. Each technical guideline has a qualifying condition that must be met for the site to be acceptable. In addition, three of the technical guidelines have a disqualifying condition. A site is unacceptable if any one of the disqualifying conditions is found to be present. The technical guidelines also identify favorable conditions and potentially adverse conditions that describe characteristics of the setting that, if present, could benefit or adversely affect the ease or cost of constructing, operating, or closing a repository.

The Implementation Guidelines require that the qualifying and disqualifying conditions of the system and technical guidelines be evaluated and that specific findings be made for each condition at principal decision points in the siting process. These findings are stated in 10 CFR Part 960, Appendix III, and are shown in Table 8.3.5.7-1.

*The passage of the Nuclear Waste Policy Amendments Act of 1987 (NWPAA, 1987) may impact the manner in which this process is implemented.

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Table 8.3.5.7-1. Findings for qualifying and disqualifying conditions

| | | |
|---|---|---|
| **Disqualifying condition--lower-level findings** | | |
| Level 1 | (a) | The evidence does not support a finding that the site is disqualified. |
| | (b) | The evidence supports a finding that the site is disqualified. |
| **Disqualifying condition--higher-level findings** | | |
| Level 2 | (a) | The evidence supports a finding that the site is not disqualified on the basis of that evidence and is not likely to be disqualified. |
| | (b) | The evidence supports a finding that the site is disqualified or is likely to be disqualified. |
| **Qualifying condition--lower-level findings** | | |
| Level 3 | (a) | The evidence does not support a finding that the site is not likely to meet the qualifying condition. |
| | (b) | The evidence supports a finding that the site is not likely to meet the qualifying condition, and therefore the site is disqualified. |
| **Qualifying condition--higher-level findings** | | |
| Level 4 | (a) | The evidence supports a finding that the site meets the qualifying condition and is likely to continue to meet the qualifying condition. |
| | (b) | The evidence supports a finding that the site cannot meet the qualifying condition or is unlikely to be able to meet the qualifying condition, and therefore the site is disqualified. |

There are four levels of findings--disqualifying and qualifying conditions both require a lower-level and higher-level finding. Lower-level findings must be made to determine if a site may be nominated as suitable for characterization or recommended as a candidate site for characterization. Higher-level findings, however, are the findings that must be made to determine if a site may be recommended for the development of a repository. Disqualifying conditions require Level 1 and Level 2 findings, and qualifying conditions require Level 3 and Level 4 findings. Each level has both a positive finding and a negative finding associated with it.

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Table 8.3.5.7-2 shows the findings previously made for the guideline qualifying and disqualifying conditions concerned with preclosure ease and cost of construction. These findings and the evidence supporting them are given in the Yucca Mountain Project environmental assessment (DOE, 1986b). The available evidence was sufficient to support positive lower-level findings for the qualifying and disqualifying conditions of the technical guidelines and the specified preclosure system guideline. To determine if the Yucca Mountain site is suitable for the development of a repository, therefore, higher-level findings must be made for the qualifying and disqualifying conditions of the system and technical guideline.

The DOE siting guidelines do not require any findings similar to lower-level or higher-level findings to be made for the favorable or potentially adverse conditions of the technical guidelines. As stated in the Supplementary Information (DOE, 1984b) for 10 CFR Part 960, Overview of the Guidelines, these conditions were intended to be used to predict the suitability of a site and provide a preliminary indication of system performance before the start of detailed site characterization studies. These conditions were considered and used in the identification of potentially acceptable sites, and in the nomination and recommendation of sites as suitable for characterization. By the completion of site characterization, however, sufficient data will be available to directly evaluate site performance and repository designs against the qualifying conditions of the system and technical guidelines. Therefore, the favorable and potentially adverse conditions will not be considered in specific terms as they were for the environmental assessment (DOE, 1986b).

Approach to resolving the issue

Key Issue 4 is basically concerned with design concepts, whereas Key Issue 1 and Key Issue 2 are concerned with postclosure and preclosure aspects of repository performance. Key Issue 1 is concerned with performance of the repository as compared with the postclosure release standard and other requirements as implemented in 10 CFR Part 60. Key Issue 2 is concerned with the preclosure performance of the repository as compared with the allowable release limits as specified in 10 CFR 60.111, 40 CFR 191 Part A, and 10 CFR Part 20. Key Issue 4, on the other hand, is concerned with the feasibility and availability of the technology needed to construct, operate, and close the repository, and with the reasonableness of the cost associated with the repository in comparison with the other sites under consideration. Since the passage of the NWPAA, the Yucca Mountain site is the only site under consideration and the requirement for the comparative evaluation of costs is no longer applicable (NWPAA, 1987). As noted, these are design topics. The reader should specifically note that the higher-level findings required for Issue 1.9 (higher-level findings--postclosure) and Issue 2.5 (higher-level findings--preclosure radiological safety) are concerned with repository performance by comparison with numerical standards. The strategies for resolving these two issues reference other related performance issues in outlining the information needed to make these findings. Conversely, the higher-level findings required for Issue 4.1 (this issue) are concerned with design questions of feasibility and safety for which there are no numerical standards, and the resolution strategy described below references related design issues to indicate the source of the information needed to make these findings.

Table 8.3.5.7-2. Preliminary findings for the qualifying and disqualifying conditions concerned with ease and cost of construction.[a]

| Preclosure guideline (10 CFR Part 960) | | Preliminary finding[b] |
|---|---|---|
| 960.5-1(a)(3) | System qualifying condition | Level 3(a) |
| 960.5-2-8 | Surface characteristics | |
| (a) | Qualifying condition | Level 3(a) |
| 960.5-2-9 | Rock characteristics | |
| (a) | Qualifying condition | Level 3(a) |
| (d) | Disqualifying condition | Level 3(a) |
| 960.5-2-10 | Hydrology | |
| (a) | Qualifying condition | Level 3(a) |
| (d) | Disqualifying condition | Level 3(a) |
| 960.5-2-11 | Tectonics | |
| (a) | Qualifying condition | Level 3(a) |
| (d) | Disqualifying condition | Level 3(a) |

[a]Preliminary findings from DOE (1986b).
[b]See Table 8.3.5.7-1 for an explanation of the finding levels.

To resolve Issue 4.1 sufficient evidence must be available to support either a positive or negative higher-level finding for each qualifying and disqualifying condition associated with the preclosure guideline on ease and cost of construction, operation and closure. Each of the qualifying conditions references the requirement for technical feasibility based on reasonably available technology. In making higher-level findings for the qualifying and disqualifying conditions, specific aspects of the geologic setting must be considered in the evaluation of this requirement.

Figure 8.3.5.7-1 shows the strategy for resolving Issue 4.1. The first step is to eliminate, if possible, from further consideration the qualifying and disqualifying conditions for which higher-level findings have already been made. In the group of technical guidelines subsumed by Issue 4.1, there are none that meet this condition. Next, for each condition, it is determined whether the evidence presently available is sufficient to support a higher-level finding. This evidence consists of the information presented in the Yucca Mountain Project environmental assessment (DOE, 1986b) and in Chapters 1 through 7 of the SCP. If the evidence is sufficient, the finding and the evidence are documented.

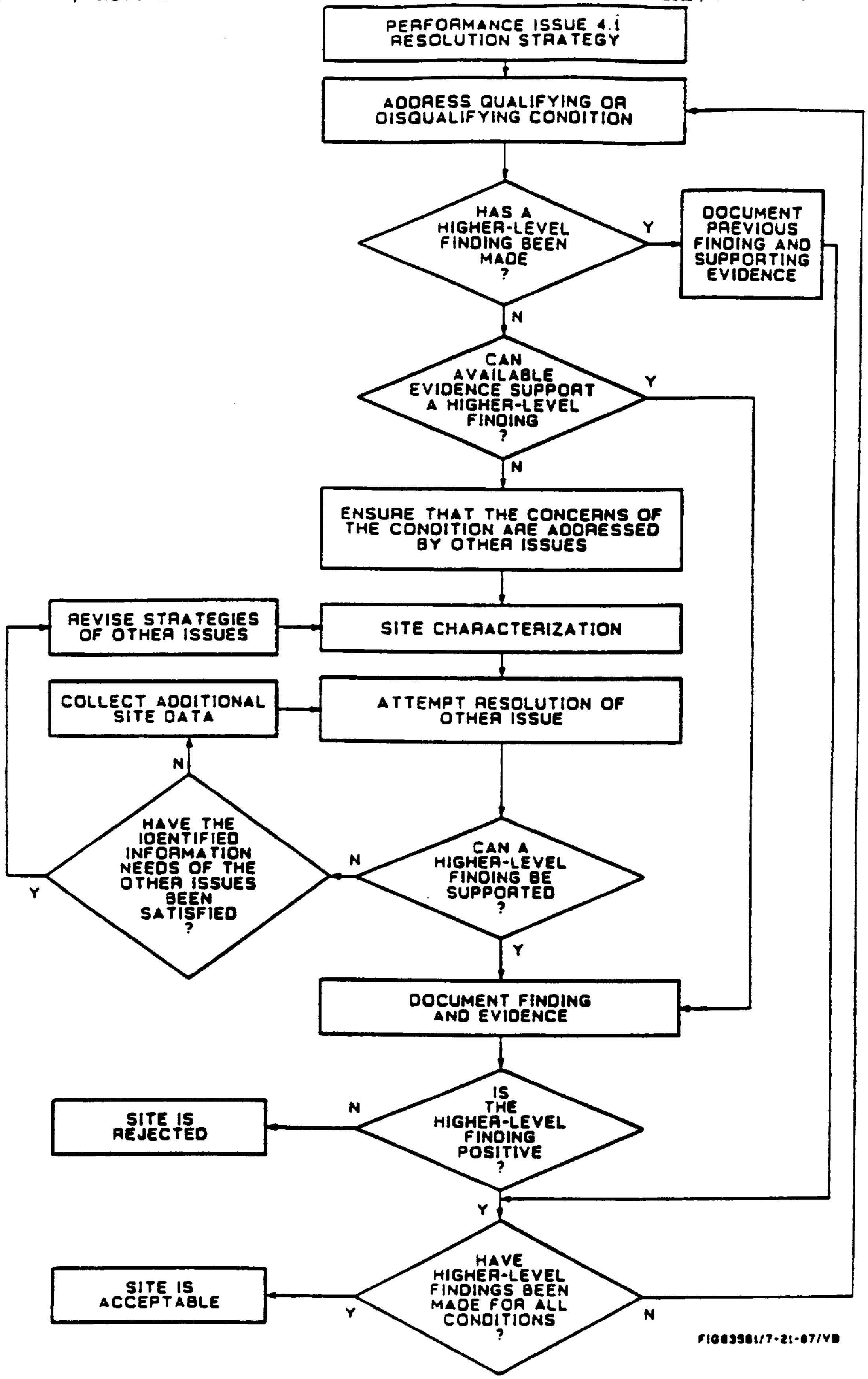


Figure 8.3.5.7-1. Issue resolution strategy for Issue 4.1 (higher-level findings——ease and cost of construction)

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For the qualifying and disqualifying conditions for which there is not adequate evidence available, the planned site characterization studies are reviewed to determine if the conditions will be investigated. This is accomplished by evaluating the resolution strategies for Issues 4.2 through 4.4, the preclosure design issues that are relevant to the evaluation of Performance Issue 4.1. These issue resolution strategies address design concerns in terms of the proposed technologies for construction, operation, and closure being reasonably available. Resolution of Issue 4.5, (total system costs), which was intended to provide the strategy for comparative evaluation of costs as called for by the performance issue, is no longer appropriate or necessary since the passage of the NWPAA (NWPAA, 1987). Therefore, if the concerns of the qualifying and disqualifying conditions are being considered in the resolution strategies of these issues, then the evidence to support higher-level findings will be made available through the information, analyses, and assessments that support resolution of these design issues. A correlation of the qualifying and disqualifying conditions and the issues that will supply the information is shown in Table 8.3.5.7-3.

After ensuring that the qualifying and disqualifying conditions will be investigated, the information necessary to assess compliance will be obtained during site characterization. Upon completion of the assessments, the results will be evaluated to determine if sufficient evidence is available to support higher-level findings. If the evidence is sufficient, the findings and the evidence will be documented. If the evidence shows that a negative higher-level finding must be made for any one of the conditions, i.e., that a disqualifying condition is present or that a qualifying condition is not present, then the site will be disqualified. This evaluation will continue until positive higher-level findings can be supported for all the conditions or until a negative higher-level finding must be made.

If, in evaluating the results of the assessments, insufficient information is found to support either a positive or a negative higher-level finding for a qualifying or disqualifying condition, additional data or analyses may be necessary to satisfy existing information needs. The resolution strategies of the appropriate design issues will be reviewed to determine if, in fact, the condition was adequately considered and the related information needs were satisfied. If not, the strategies for the design issues will be revised and new information needs will be identified as necessary, additional data will be collected, and compliance will be reassessed. This process continues until there is sufficient evidence to support either a positive or a negative higher-level finding for every qualifying and disqualifying condition. As discussed previously, findings are not required for the favorable conditions or the potentially adverse conditions at this stage in the siting process. However, the DOE's analyses indicate that the concerns of these conditions are adequately addressed through the data and analyses of other issues.

System Guideline Qualifying Condition

The preclosure system guideline qualifying condition on ease and cost of siting, construction, operation, and closure is stated in 10 CFR 960.5-1(a)(3) as follows:

Table 8.3.5.7-3. Preclosure design issues that address the concerns of the qualifying and disqualifying conditions of the preclosure guidelines on ease and cost of siting, constructing, operating, and closing a repository

| Guideline | Issue 4.2[a] (nonradiological health and safety) | Issue 4.3[a] (waste package production technologies) | Issue 4.4 (preclosure design and technical feasibility) | Issue 4.5[a,b] (total system costs) |
|---|---|---|---|---|
| System guideline | | | | |
| Qualifying condition | D[c] | D | D | NA[d] |
| Surface characteristics | | | | |
| Qualifying condition | I[e] | I | D | NA |
| Rock characteristics | | | | |
| Qualifying condition | I | I | D | NA |
| Disqualifying condition | I | I | D | NA |
| Hydrology | | | | |
| Qualifying condition | I | I | D | NA |
| Disqualifying condition | I | I | D | NA |
| Tectonics | | | | |
| Qualifying condition | I | I | D | NA |
| Disqualifying condition | I | I | D | NA |

[a]Issues 4.2, 4.3, and 4.5 do not directly require site information. Rather, they place design constraints upon Issue 4.4 or evaluate products prepared under Issue 4.4.

[b]Resolution of Issue 4.5 is not required as the Yucca Mountain site is the only site under consideration for development as a repository as designated by the Nuclear Waste Policy Amendments Act of 1987 (NWPAA, 1987).

[c]Information considered in resolving the issue directly (D) contributed to the higher-level finding for the specified guideline condition.

[d]NA = not applicable.

[e]Information considered in resolving the issue indirectly (I) contributed to the higher-level finding for the specified condition.

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> Repository siting, construction, operation, and closure shall be demonstrated to be technically feasible on the basis of reasonably available technology, and the associated costs shall be demonstrated to be reasonable relative to other available and comparable siting options.

This qualifying condition is concerned with the feasibility of a potential repository site from the perspectives of the relative reasonableness of the cost of siting, constructing, operating, and closing the facility compared with the other siting options. Since the passage of the NWPAA, this portion of the qualifying condition no longer needs to be addressed in making a finding on the system guideline (NWPAA, 1987). The condition is also concerned with the availability of the technology required to implement the design developed to meet the regulatory requirements expressed under Key Issues 1 and 2, as well as the other concerns addressed under Key Issue 4. To make the higher-level finding for this qualifying condition, the evidence must be available to (1) establish the properties of the host rock and the character of the site and to develop constitutive models, (2) develop and demonstrate site-specific equipment for packaging and handling the waste and to perform specific mining and drilling tasks, (3) identify site-specific seal requirements and develop site-specific materials, designs, and emplacement techniques for the seals, and (4) integrate the resulting information into an overall design that will meet the functional requirements and performance criteria established for the repository. The design task is an evolutionary and iterative process that includes (1) the formulation, testing, and refinement of concepts, (2) the combination of concepts into the design, (3) analyses of the design for technical validity, (4) comparisons of the design with criteria and requirements, and (5) the evaluation of costs to implement the design. This sequence is repeated and refined until the design meets the requirements established for performance, efficiency, and cost effectiveness.

The information that will be used to support the higher-level finding for this qualifying condition derives primarily from the design and cost evaluation of the facility as addressed under Issues 4.4 and 4.5. Therefore, the site characteristic information used in the development of the design is included. An effort has been made to centralize repository design activities for the Yucca Mountain site under Issue 4.4 (preclosure design and technical feasibility). Thus, although postclosure facility design and design requirements for preclosure radiological safety are not explicitly addressed under Key Issue 4, the design prepared under Key Issue 4 incorporates these concerns in addition to those expressed in Issue 4.1 (see Section 8.3.2.1).

Surface characteristics

There is one qualifying condition for this technical guideline for which higher-level findings must be made.

Qualifying condition. The qualifying condition for the technical guideline on surface characteristics (10 CFR 960.5-2-8(a)) is as follows:

> The site shall be located such that, considering the surface characteristics and conditions of the site and surrounding area including surface water systems and the terrain, the requirements specified in

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10 CFR 960.5-1(a)(3) can be met during repository siting, construction, operation, and closure.

The qualifying condition is concerned with the potential for surface conditions of the site and surrounding area that could impact the ability of the site to meet the cost and technical feasibility requirements specified in the system guideline. Assurance that the preclosure system can be constructed and operated under the surface conditions present or credibly expected to be encountered must be provided. A determination of the surface characteristics and conditions, as well as credible events, is required for the evaluation needed to determine compliance with the system guideline and to make the higher-level finding required for this qualifying condition.

The impact of surface characteristics on repository preclosure performance will be evaluated in support of the resolution of Issue 4.4 (Section 8.3.2.5). These evaluations will also serve as the basis for making a higher-level finding for the qualifying condition of the guideline on surface characteristics under Issue 4.1. No additional information outside the information needs identified as being needed for resolution of Issue 4.4 is required.

The link between the information required for making a higher-level finding on the qualifying condition for the technical guideline on surface characteristics and the information needs identified to support resolution of preclosure design Issue 4.4 is identified in Table 8.3.5.7-4.

Rock characteristics

There are three qualifying and one disqualifying conditions for this technical guideline for which higher-level findings must be made.

Qualifying conditions. The qualifying conditions for the technical guideline on rock characteristics (10 CFR 960.5-2-9(a)) are as follows:

The site shall be located such that:

(1) the thickness and lateral extent and the characteristics and composition of the host rock will be suitable for accommodation of the underground facility;

(2) repository construction, operation, and closure will not cause undue hazard to personnel; and

(3) the requirements specified in 10 CFR 960.5-1(a)(3) can be met.

The qualifying conditions for preclosure rock characteristics require that the host rock must be capable of safely accommodating the construction, operation, and closure of the underground facility using reasonably available technology. A determination of the characteristics and properties for the geologic setting in which construction activities are proposed is required to determine compliance with the system guideline and to make a higher-level finding for this qualifying condition.

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Table 8.3.5.7-4. Surface characteristics information considered in making the higher-level finding for the qualifying condition of the surface characteristics guideline, and issues for which the information will be obtained

| Issue | Information |
|---|---|
| 4.2 | No surface characteristics information required |
| 4.3 | No surface characteristics information needed |
| 4.4 | Surface topography at facility locations<br>Surface topography at candidate mined material storage area<br>Surface topography at underground access locations<br>Surface topography of surface facility sites<br>Surface topography on access routes<br>Surface topography at facility locations<br>Allowable foundation bearing load pressure for soil considering shear failure and settlement (total and differential)<br>Allowable foundation bearing load pressure for rock considering shear failure and settlement (total and differential)<br>Active and passive soil pressures for flexible and rigid structural walls<br>Active and passive rock pressure for flexible and rigid structural walls<br>Factor of safety for an identified mechanism of potential slope failure in soil for static and dynamic loading conditions<br>Factor of safety for an identified mechanism of potential slope failure in rock for static and dynamic loading conditions<br>Magnitude and rate of time dependent settlement in soils below earthfills<br>Magnitude and rate of swell in subgrade soils below roads<br>Magnitude of soil collapse below surface facilities (foundations, earthfills, and roads) due to saturation and/or loading<br>Soil liquefaction potential for saturated low density soils under dynamic loading conditions |

The characteristics and properties of the host rock must be determined in support of evaluations made for resolution of design Issue 4.4. Evaluations of these characteristics and properties will serve as the basis for making the higher-level finding for the qualifying condition of the technical guideline on rock characteristics under Issue 4.1. Other than the information needs identified for the design issue just cited, no additional information is required.

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The link between the information required for making a higher-level finding on the qualifying condition for the technical guideline on rock characteristics and the information needs identified to support resolution of preclosure design issues is identified in Table 8.3.5.7-5.

Disqualifying condition. The disqualifying condition for the technical guideline on rock characteristics (10 CFR 960.5-2-9(d)) is in Table 8.3.5.7-5. The site shall be disqualified if the rock characteristics are such that the activities associated with repository construction, operation, or closure are predicted to cause significant risk to the health and safety of personnel, taking into account mitigating measures that use reasonably available technology.

The information identified in Table 8.3.5.7-5 will also support the evaluation necessary to reach the required higher-level finding for this disqualifying condition.

Hydrology

There are three qualifying and one disqualifying conditions for this technical guideline for which higher-level findings must be made.

Qualifying conditions. The qualifying conditions for the technical guideline on hydrology (10 CFR 960.5-2-10(a)) are as follows:

> The site shall be located such that the geohydrologic setting of the site will
>
> (1) be compatible with the activities required for repository construction, operation, and closure;
>
> (2) not compromise the intended functions of the shaft liners and seals; and
>
> (3) permit the requirements specified in 10 CFR 960.5-1(a)(3) to be met.

These qualifying conditions require that the present and expected characteristics of the geohydrologic setting be compatible with the safe construction, operation, and closure of the repository using reasonably available technology as required by the system guideline. A determination of the hydrologic characteristics and properties within the geologic setting is required for the evaluations needed to determine compliance with the system guideline and to make a higher-level finding for these qualifying conditions.

Evaluations of the geohydrologic setting and of the resulting impact on repository preclosure performance will be performed in support of the resolution of Issue 4.4. These evaluations will serve as the basis for making a higher-level finding for the qualifying condition of the geohydrology guideline under Issue 4.1. Other than the information needs identified for the design issue just cited, no additional information is required.

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Table 8.3.5.7-5. Rock characteristics information considered in making the higher-level finding for the qualifying condition of the rock characteristics guideline, and issues for which the information will be obtained

| Issue | Information |
|---|---|
| 4.2 | No site rock characteristics information is requested directly by this issue |
| 4.3 | No site rock characteristics information is requested directly by this issue |
| 4.4 | Description and frequency of abnormal conditions in rock mass<br>Initial formation temperature<br>Thermal conductivity of rock<br>Heat capacity of rock<br>Rock properties in primary area<br>    Poisson's ratio (intact rock)<br>    In situ stress (rock mass)<br>    Coefficient of thermal expansion (rock mass)<br>    Thermal conductivity (rock mass)<br>    Young's modulus (intact rock)<br>    Deformation modulus (rock mass)<br>    Heat capacity<br>    Unconfined compressive strength (intact rock)<br>    Cohesion of rock and angle of internal friction--intact rock (compressive strength as a function of confining pressure)<br>    Joint normal and shear stiffness properties (fractures)<br>    Joint wall compressive strength (fracture surfaces)<br>    Joint roughness coefficient (fracture surfaces)<br>    Cohesion and coefficient of friction (fractures)<br>    Joint frequency and spacing<br>    Joint orientation<br>    Number of joint sets<br>    Joint roughness and condition of joints<br>    Rock quality designation<br>    Joint alteration<br>    Construction method<br>    Presence of swelling or squeezing ground<br>    Water inflow<br>    Expected seismic loading<br>Stratigraphic features<br>    Depth, thickness, and lateral extent of host rock<br>    Stratigraphy and structural features |

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The link between the information required for making a higher-level finding on the qualifying condition for the technical guideline on hydrology and the information needs identified to support resolution of preclosure design issues is identified in Table 8.3.5.7-6.

Disqualifying condition. The disqualifying condition for the technical guideline on hydrology (10 CFR 960.5-2-10(d)) is as follows:

> A site shall be disqualified if, based on expected ground water conditions, it is likely that engineering measures that are beyond reasonably available technology will be required for exploratory shaft construction or for repository construction, operation, or closure.

The information identified in Table 8.3.5.7-6 will also support the evaluation necessary to reach the required higher-level finding for this disqualifying condition.

Tectonics

There is one qualifying and one disqualifying condition for this technical guideline for which a higher-level finding must be made.

Qualifying condition. The qualifying condition for the technical guideline on tectonics (10 CFR 960.5-2-11(a)) is as follows:

> The site shall be located in a geologic setting in which any projected effects of expected tectonic phenomena or igneous activity on repository construction, operation, or closure will be such that the requirements specified in 10 CFR 960.5-1(a)(3) can be met.

The characteristics and probability of occurrence of tectonic and igneous processes and events must be determined to identify the potentially disruptive scenarios that may affect the ability of the site to meet the preclosure requirements on ease and cost of construction, operation, and closure as specified in the system guideline and to make a higher-level finding for this qualifying condition. An evaluation of these same processes, events, and scenarios is also required to support the resolution of Issue 4.4. The information identified as being needed to resolve this design issue will serve as the basis for the required higher-level finding for the qualifying condition for tectonics under Issue 4.1. No new information needs are required for the higher-level finding for this qualifying condition.

The link between the information required for making a higher-level finding on the technical guideline for tectonics and the information needs identified to support the resolution of other preclosure issues is identified in Table 8.3.5.7-7.

Disqualifying condition. The disqualifying condition for the technical guideline on tectonics (10 CFR 960.5-2-11(d)) is as follows:

Table 8.3.5.7-6. Hydrologic information considered in making the higher-level finding for the qualifying condition of the hydrology guideline, and issues for which the information will be obtained

| Issue | Information |
|---|---|
| 4.2 | No site hydrologic characteristics are requested directly by this issue |
| 4.3 | No site hydrologic characteristics are requested directly by this issue |
| 4.4 | Surface<br>Surface hydrology for 5-, 25-, 50-, 100-, 500-year flood and the probable maximum flood (PMF)<br>Area of inundation<br>Surface water systems, stream flow rate, quantities and durations, channel morphology<br>Subsurface<br>Aquifer locations<br>Aquifer characteristics<br>Sustained yield of pumped water source for operational water |

A site shall be disqualified if, based on the expected nature and rates of fault movement or other ground motion, it is likely that engineering measures that are beyond reasonably available technology will be required for exploratory shaft construction or for repository construction, operation, or closure.

The information identified in Table 8.3.5.7-7 will also support the required higher-level finding for this disqualifying condition of the technical guideline on tectonics.

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Table 8.3.5.7-7. Tectonics information considered in making the higher-level finding for the qualifying condition of the tectonics guideline, and issues for which the information will be obtained

| Issue | Information |
|---|---|
| 4.2 | No site tectonic information is requested directly by this issue |
| 4.3 | No site tectonic information is requested directly by this issue |
| 4.4 | Surface<br>Identification and characterization of late Quaternary faults in the repository block. If determined to exist, establish location, orientation, and probability of exceeding 7 cm displacement in areas of waste emplacement<br>Design basis ground motion time histories and corresponding response spectra at underground facility locations<br>Combined potential for vibratory ground motion at underground facility locations<br>Probability of volcanic eruption through area of waste emplacement<br>Stratigraphic contacts for top and bottom of the TSw2 formation within candidate areas for repository<br>Identification of any fault within 100 m of facilities important to safety (FITS) with greater than 1 chance in 100 of producing more than 5 cm of surface displacement in 100 years. If determined to exist, establish location at surface, orientation at surface, and probability of exceeding 5 cm displacement under FITS<br>Design basis ground motion time histories and corresponding response spectra<br>Potential for exceeding design basis ground motion at FITS<br>Probability vs. peak ground acceleration, peak ground velocity, and peak velocity response at selected frequencies at surface fits locations<br>Probability of volcanic eruption that would disrupt surface facilities<br>Design basis ash fall thickness<br>Soil-structure interaction considering displacements and degree of yielding in soil beneath the base of the building<br>Soil-structure interaction considering displacements and degree of yielding in soil adjacent to retaining walls<br>Rock-structure interaction considering displacements and degree of yielding in rock beneath the base of the building<br>Rock-structural interaction considering displacements and degree of yielding in rock adjacent to retaining walls<br>Subsurface<br>Fault properties<br>Location |

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Table 8.3.5.7-7. Tectonics information considered in making the higher-level finding for the qualifying condition of the tectonics guideline, and issues for which the information will be obtained (continued)

| Issue | Information |
|---|---|
| 4.4 (continued) | |
| | Subsurface, fault properties (continued)<br>  Orientation<br>    Physical, thermal, and mechanical properties of major faults |

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## 8.3.5.8 Strategy for postclosure performance assessment

As explained in the introduction to Section 8.3.5, assessments of the performance of a repository at Yucca Mountain are required for resolving the performance issues in the issues hierarchy; a major part of the performance-assessment program will examine the postclosure behavior of the repository. The detailed plans for the assessment of postclosure behavior are described as part of the issue-resolution strategies in Sections 8.3.5.9 through 8.3.5.18. The principal presentations of these plans are in Sections 8.3.5.9 and 8.3.5.10 for assessments of the waste package and in Sections 8.3.5.12 and 8.3.5.13 for assessments of the site. In addition, waste-package performance assessment is reviewed in Section 7.4.5.

This section describes strategic aspects of the performance-assessment program that are common to all those detailed plans for assessing postclosure performance. The first part of this section, a brief overview of the performance-assessment strategy, begins by explaining the relationships among the performance issues. The overview then describes the major steps in the iterative process by which final performance is assessed and performance issues resolved. At several points in the iterative process, the DOE must decide whether the available data are sufficient for carrying out the assessments; the overview emphasizes these steps because many of the needed data will be supplied by the site characterization program. The second part of this section reviews the conceptual models of a Yucca Mountain repository that have been used in the preliminary work underlying the detailed performance-assessment plans.

### Overview of strategy

The primary objective of the Yucca Mountain Project postclosure performance assessment program is to resolve Key Issue 1 in the issues hierarchy, which is

> Will the mined geologic disposal system at Yucca Mountain isolate the radioactive waste from the accessible environment after closure in accordance with the requirements set forth in 40 CFR Part 191, 10 CFR Part 60, and 10 CFR Part 960?

The performance issues under Key Issue 1 parallel the regulatory criteria in 10 CFR Part 60 and 10 CFR Part 960. Each issue either asks whether specific performance objectives can be met or asks for analyses and qualitative judgments of the expected future conditions at Yucca Mountain after the repository at the site has been closed and decommissioned. These performance issues are the following:

| Issue | Issue statement | SCP section |
|---|---|---|
| 1.1 | Will the mined geologic disposal system meet the system performance objective for limiting radionuclide releases to the accessible environment as required by 10 CFR 60.112 and 40 CFR 191.13? | 8.3.5.13 |

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| | | |
|---|---|---|
| 1.2 | Will the mined geologic disposal system meet the requirements for limiting individual doses in the accessible environment as required by 40 CFR 191.15? | 8.3.5.14 |
| 1.3 | Will the mined geologic disposal system meet the requirements for the protection of special sources of ground water as required by 40 CFR 191.16? | 8.3.5.15 |
| 1.4 | Will the waste package meet the performance objective for containment as required by 10 CFR 60.113? | 8.3.5.9 |
| 1.5 | Will the waste package and repository engineered barrier systems meet the performance objective for radionuclide release rates as required by 10 CFR 60.113? | 8.3.5.10 |
| 1.6 | Will the site meet the performance objective for prewaste emplacement ground-water travel time as required by 10 CFR 60.113? | 8.3.5.12 |
| 1.7 | Will the performance-confirmation program meet the requirements of 10 CFR 60.137? | 8.3.5.16 |
| 1.8 | Can the demonstrations for favorable and potentially adverse conditions be made as required by 10 CFR 60.122? | 8.3.5.17 |
| 1.9 | (a) Can the higher-level findings required by 10 CFR Part 960 be made for the qualifying condition of the postclosure system guideline and the disqualifying and qualifying conditions on the technical guidelines for geohydrology, geochemistry, rock characteristics, climate changes, erosion, dissolution, tectonics, and human interference; and (b) can the comparative evaluations required by 10 CFR 960.3-1-5 be made? | 8.3.5.18 |

The flow of information among Issues 1.1 through 1.6 is depicted schematically in Figure 8.3.5.8-1, which also shows, in simplified form, the exchange of information with the group of three design issues under Key Issue 1. Even though the diagram indicates only one-way flow, some information flows backwards along the lines shown in the figure. This backward flow conveys the results of sensitivity analyses, which are carried out in each issue as part of its treatment of uncertainty. These sensitivity analyses reveal whether the information supplied to an issue is sufficient for its needs, and their results may, therefore, be conveyed from an issue back to the issue that supplied the information.

The connections among issues shown in Figure 8.3.5.8-1 achieve an important synergism. A single series of analyses may often answer questions that arise in solving more than one issue. Because of these close connections, the results of analyses performed in one issue are available to guide the work in other issues.

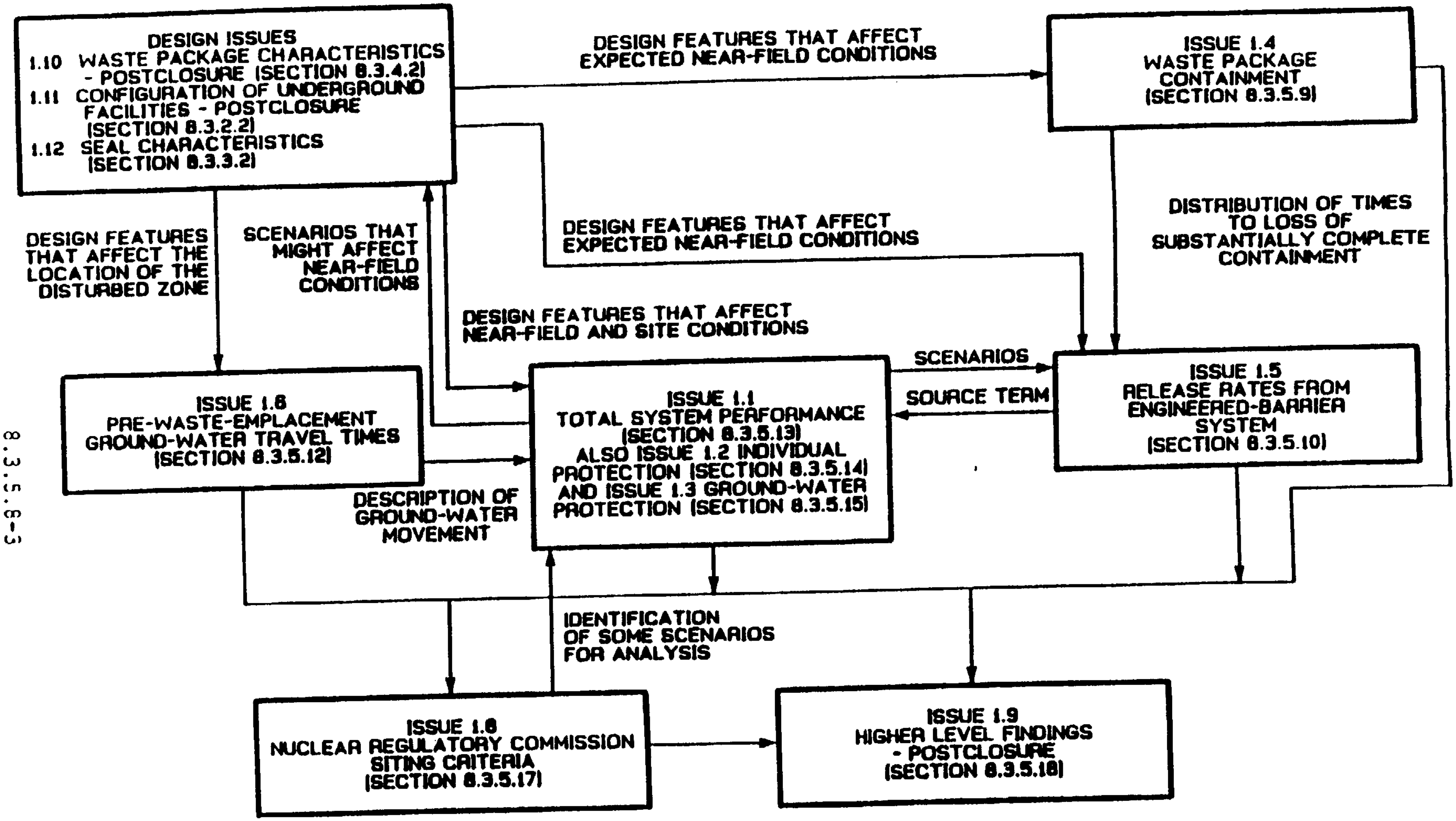


**Figure 8.3.5.8-1.** Simplified information flow among postclosure performance issues and their interaction with design issues.



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The figure also shows an information flow path between the collective results of Issues 1.1 through 1.6 and Issues 1.8 and 1.9. This path is present because the insight and tools developed as a result of providing quantitative answers to Issues 1.1 through 1.6 will contribute to the evaluation of whether the waste-disposal system can meet the regulatory criteria addressed in Issues 1.8 and 1.9. Further discussion of this link appears in Sections 8.3.5.17 and 8.3.5.18.

The figure does not show Issue 1.7, which addresses the need to establish a performance confirmation program. The need for a performance confirmation will be identified from the performance assessment that will be conducted during site characterization. The approach to deciding what aspects of the program need to be confirmed after license application is discussed in Section 8.3.5.16.

Under each of the Yucca Mountain Project issues is a set of information needs. The information needs under each postclosure performance issue (presented in detail in Sections 8.3.5.9 through 8.3.5.18) are structured to reflect the iterative application of the general issue-resolution strategy described in Section 8.1.2. The next few paragraphs explain the structure and the iterations, shown schematically in Figure 8.3.5.8-2.

The figure presents five major steps in assessing postclosure performance. In actual practice, of course, many of the steps take place simultaneously and not necessarily in the strict order implied by arrows in the figure. For example, preliminary calculations are performed while models are being developed and tested and before scenarios have been completely identified. As the arrows on the right-hand side of the figure suggest, progress made in one step may indicate a need for further development in a step that is higher in the figure. For example, an attempt, in the fourth step, to calculate values for performance measures may point out a deficiency in a conceptual model developed in the third step; further work in model development would then be called for. Iterations also occur as data become available, and the following discussion describes three points at which the sufficiency of the available data can logically be judged in terms of the needs of performance assessment for doing the next step.

The first step in this process is the compilation of the relevant existing site and design information. The first information need under each issue is, therefore, a summary of the parameters for which data are needed. The information currently available is described in Chapters 1 through 7, but eventually, this information, augmented by the results of the data-gathering programs described in this site characterization plan, will be provided primarily through the reference information base (RIB). The RIB will be a compilation of the current best information to be used in design and performance analyses. This common source of information will help to ensure uniformity among the analyses carried out in separate issues.

The available information is used in the next step to develop conceptual models and scenarios including the sets of hypothetical events and processes that must be examined to resolve the issue and to develop boundary conditions for calculations. After the existing data have been compiled, the question is asked: "Are the data sufficient to continue with the next step?" In the early iterations through the process, the data may be sufficient if there is

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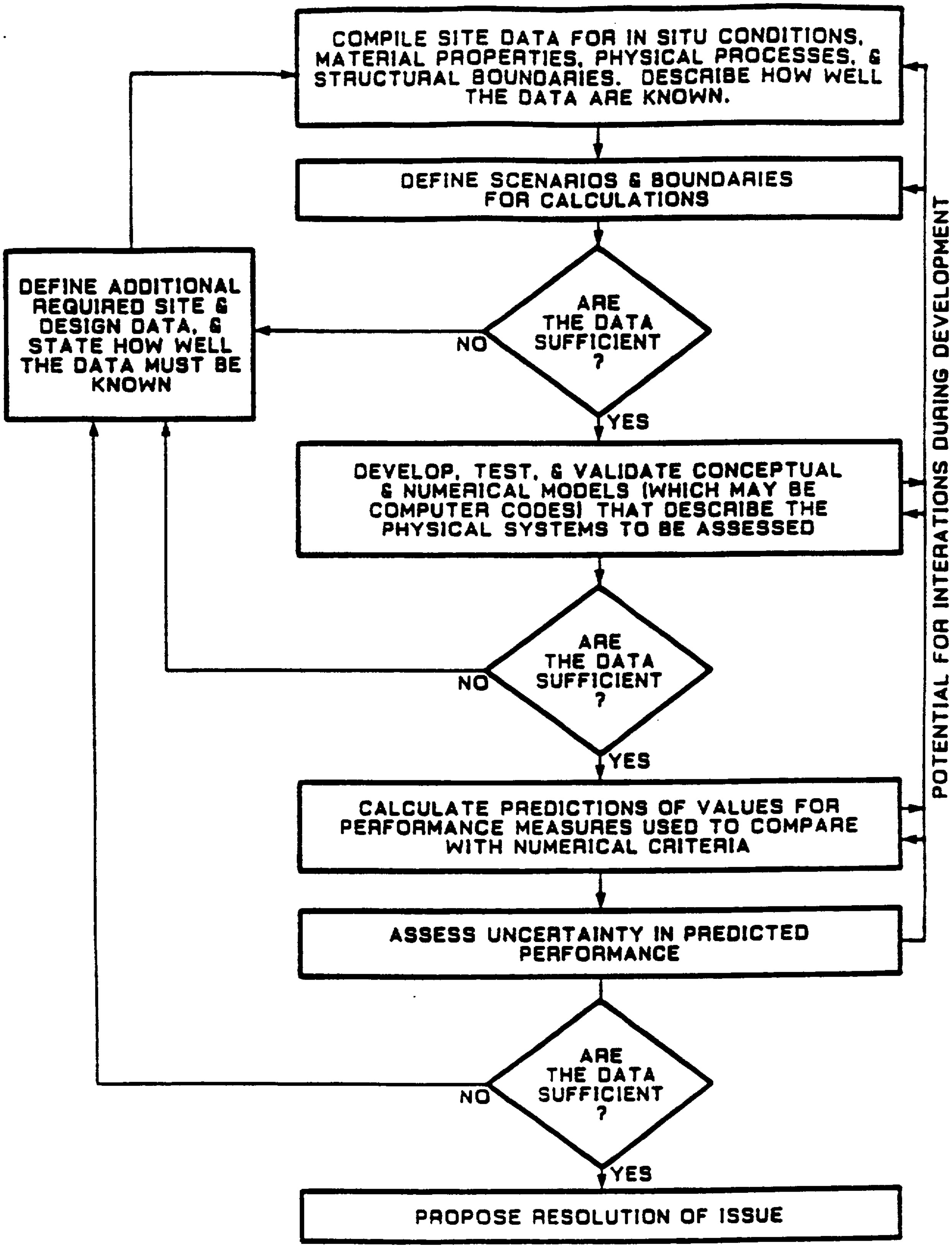


**Figure 8.3.5.8-2.** Steps in performance assessment for post-closure performance issues.

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at least a bounding value to use for every parameter that must be input for the analysis and for the selection of scenarios. In later iterations, the answer depends on whether the data provide usefully realistic values for those parameters. If the data are judged not sufficient, the performance assessors must call for additional data, as Figure 8.3.5.8-2 shows.

The third step shown in the figure includes the validation that must be attempted for the calculational models used to predict the values of the performance measures; this validation provides reasonable assurance in the predicted values of the performance measures. In addition, further model development may be necessary to modify or expand the existing conceptual models of the system or subsystem behavior. This development consists of describing the conceptual models in terms of mathematical equations and of constructing algorithms to solve the equations. The calculational-model development often must proceed in parallel with the scenario development, because details of a calculational model may depend on the particular scenario to be analyzed. Plans for validation of conceptual models of site characteristics are described in Section 8.3.1. Plans for validation of analytic techniques to be used in the performance assessments are summarized in Section 8.3.5.20.

Again the question is asked: "Are the data sufficient?" At this point, the data requirements are more stringent because the fourth step requires predictions for comparison with numerical criteria. The data must be certain enough to allow the assessors to draw conclusions about the events and processes being examined.

When sufficient confidence in the models has been attained, values for the performance measures are calculated to assess whether the performance goals are met with the desired confidence. (Explanations of these terms and of their role in issue resolution are in Section 8.1.2.) The uncertainty in the predictions is assessed, and the question is asked again: "Are the data sufficient?" The requirements for sufficiency are most stringent at this point. The data must allow the heterogeneity of the system to be realistically assessed and the effect of future conditions on the models and the material properties to be satisfactorily accounted for. As part of this process, the sensitivity of the performance measure to various parameters and conditions must also be assessed. In some scenarios, the uncertainty in a parameter may be shown insignificant because the behavior of the system under assessment is insensitive to the parameter; the requirements on data for such a parameter would accordingly be less stringent.

The process shown in Figure 8.3.5.8-2 requires numerous applications of judgment. Each decision on whether data are sufficient requires such judgment. The need for iterations and further developments will be decided through judgments of whether the work has provided a basis on which the NRC may find the "reasonable assurance" called for by 10 CFR Part 60. These decisions may involve the routine use of expert judgment, the formal use of expert judgment, or the use of peer review as defined in Altman et al. (1988). The DOE will subject the licensing assessment work to rigorous peer review, using experts from its repository programs as well as from the outside technical community. Review by the NRC will also take place continually throughout site characterization and the development of a repository. The final licensing decisions by the NRC are based on their review. The pro-

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cesses of consultation with affected states and Indian tribes will also furnish technical review of these decisions. The use of subjective methods involving judgment through peer review is an important process in all the activities shown in Figure 8.3.5.8-2. The general role of subjective methods (i.e., use of expert judgment) in site characterization is discussed in Section 8.1.

The specific work for resolving each performance issue is explained in the individual information need discussions in Sections 8.3.5.9 through 8.3.5.18. They summarize the site and design data that are needed, the scenarios and models that will be used, the predictive analyses that will be performed, the performance measures, goals, and confidences that have been allocated, and the quantitative analyses and qualitative judgments that will be used to establish the degree of certainty in the results.

Summary of conceptual models that have been used for performance assessment

The current strategy for postclosure performance assessment and the identification of information needs are partially determined by the current conceptual models of the repository system and the evaluations to date of how this system is predicted to behave with respect to the performance objectives in 10 CFR Part 60. The bulk of the work in developing the preliminary conceptual models and the evaluation of the system based upon these models has been done for the Yucca Mountain environmental assessment (DOE, 1986b); for example, Oversby and McCright (1984), Montazer and Wilson (1984), Sinnock et al. (1984a), and Klavetter and Peters (1986).

The conceptual models that were developed in the preliminary work are summarized in the following paragraphs. These preliminary conceptual models contain assumptions that simplify the conceptual models described in Chapters 1 through 7, and the following description gives the simplifying assumptions and boundary conditions that have been used to date in performance assessments. Details of scenarios based upon these conceptual models that will ultimately be considered are being developed. Plans to further develop these scenarios are described under Issue 1.1 (Section 8.3.5.13).

The most important concept used in the performance-assessment models summarized here concerns the existing hydrogeologic conditions (i.e., flow paths and water fluxes). In addition, the models must account for the bounds on the natural geochemical and future hydrologic conditions, the possible repository-induced effects on existing hydrogeologic and geochemical conditions, and future tectonic and climatic conditions.

The most probable water flow path from the repository to the accessible environment is currently thought to be vertically downward through the unsaturated Topopah Spring, Calico Hills, and Crater Flat units to the water table, and then horizontal below the water table. Because of capillarity in unsaturated rocks and the low percolation rates in the unsaturated units, the steady-state water flow between the repository location and the water table occurs in the rock matrix (for instance, Montazer and Wilson, 1984; Klavetter and Peters, 1986). As discussed in Section 3.9.1, however, water flow in some of the fractures in the Tiva Canyon, the Topopah Spring, and the zeolitized Calico Hills units may also occur and could affect radionuclide release and transport. Furthermore, water could flow laterally at some interfaces

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between rock units. The hypothesis that water movement in the Topopah Spring welded unit is dominated by evaporative vapor flux upward (Montazer and Wilson, 1984) is not currently used. This concept, if shown to be probable, would predict smaller releases of radioactivity than current models predict, because very little waste could ever be dissolved or transported in this concept. The preliminary performance-assessment models have assumed that all release of waste from the repository would be by dissolution in the ground water that flows through the Topopah Spring densely welded unit. The transport of the dissolved radionuclides, according to these models, would occur through the unsaturated zone and the saturated zone to the accessible environment. Current models also consider transport of gaseous radionuclides both by ground water and vertically upward through the unsaturated rock to the surface.

The amount and chemistry of water that contacts the waste will limit radionuclide releases. This contact water is limited by the flux that percolates through the Topopah Spring densely welded unit and by the geometry of the emplaced waste packages. The amount of contact water and time of contact may be reduced because of dry-out and changes in fracture apertures in the vicinity of the waste package. Water chemistry may also be influenced by thermally affected rock-water interactions. These thermal, mechanical, and chemical effects will be included in the analyses of the performance of the waste package and the engineered-barrier system.

With a few exceptions, it is thought that the release of radionuclides from spent fuel and glass waste form will be controlled by secondary phases bearing radionuclides and by the waste-form degradation. Some radionuclides, such as cesium-137, may never reach saturation and will be controlled by waste-form degradation and water flow. Some exceptions to these assumptions are the carbon-14 released from metal components and the mobile cesium, technetium, and iodine-129 that collect in gaps within the fuel and between the fuel pellets and the fuel cladding in spent-fuel rods.

The geochemical conditions that affect release rates are included in the analyses of the engineered-barrier system. Current waste-package-release models assume that the release from the waste package is controlled by water influx and waste-form release. Near the boundary of the engineered-barrier system current transport models assume that transport is driven by the water flowing near the package and by processes such as diffusion, dry-out, and resaturation in the near field.

The cumulative release of radionuclides is calculated at the accessible environment. Currently, the condition considered most probable, on the basis of data presented in Chapter 3, is that the percolation flux through any of the unsaturated units is less than the saturated conductivity of the rock matrix, resulting in one-dimensional water flow and radionuclide transport through the matrix. The effects of alternative conditions are as follows: for percolation fluxes higher than the saturated conductivity of the rock matrix, it is believed that flow would occur in the fracture system. The resulting paths and speeds of radionuclide transport might then be controlled by diffusion of the radionuclides from the water in the fractures into the water in the matrix.

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For intermediate fluxes, close to but not exceeding the saturated conductivity of the matrix, transport by diffusion would probably be on the same order as advective transport by convection of the water in the matrix. It is not clear whether mechanical dispersion, which is related to water velocity, would be a significant contributor to transport of radionuclides. For fluxes greater than the saturated conductivity of the matrix, mechanical dispersion in the fractures could contribute to radionuclide transport because of the higher velocities that may occur; however, the duration of the flow would probably be very short, so that dispersion in the fractures might not be an active mechanism for any significant length of time. The relative contributions of diffusion and dispersion to the transport of radionuclides in both the saturated and unsaturated zones will be studied in activities described in Section 8.3.1.2 (geohydrology program) and 8.3.1.3 (geochemistry program). The radionuclides are assumed to be retarded by the combined effects of sorption, diffusion from fractures into the matrix, mineral precipitation, and ion exchange. These effects, modeled by a bulk retardation factor and a concentration limit, are assumed to be operative in both the Topopah Spring welded unit and the Calico Hills nonwelded unit.

The conceptual models just described formed most of the bases for the performance allocation that has been done for postclosure performance Issues 1.1 through 1.9 and is described in the subsequent sections of this document. As explained in Section 8.1.2, performance allocation establishes a basis for planning site characterization work. It requires that the planners set specially defined "performance measures," "goals," and "indications of desired confidence." The "goals" are not criteria that the site must meet; they simply serve as guidance for a detailed derivation of the site characterization data needed for use in licensing a repository. As new data and a fuller understanding of the site are acquired, a new planning basis may well become appropriate, and some of the performance allocation will be revised.

One reason for reallocating performance could arise from the evaluation of the conceptual models used in the original allocation. An objective of the site characterization program will be the validation of these models. If, during site characterization, the experimental results indicate that the conceptual models that have been used are not valid, the allocation of goals and confidences to certain performance measures will be reconsidered. In addition, the performance measures themselves may have to be changed.

A second reason for reallocating performance measures, goals, and confidences will arise if new data show that (1) the ranges of values for the physical parameters are different from the ranges that have been assumed to date and (2) the measured ranges do not allow the performance goals to be met with the desired confidences.

Since considerable conservatism has been used in the performance allocation, future reassignments of goals and desired confidences, if any, are not expected to drastically change the kinds of data to be sought in site characterization. As the design and site characterization processes continue, it could, however, become necessary to call for additional tests to broaden the data base and ensure that predictions of values for performance measures are based on values characteristic of the entire site.

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8.3.5.9 Issue resolution strategy for Issue 1.4: Will the waste package meet the performance objective for containment as required by 10 CFR 60.113?

Regulatory basis for the issue

The NRC regulations set a performance objective for the waste packages to provide containment of the high-level waste (HLW) during the period after closure of the repository when the temperatures and radiation levels are highest. The performance objective for containment (10 CFR 60.113 (a)(1)(ii)) is

> the engineered barrier system shall be designed, assuming anticipated processes and events, so that: (A) Containment of HLW within the waste packages will be substantially complete for a period to be determined by the Commission taking into account factors specified in 60.113(b) provided that such period shall not be less than 300 years nor more than 1,000 years after permanent closure of the geologic repository

For the purposes of this discussion, the waste package is defined as in 10 CFR 60.2 as

> the waste form and any containers, shielding, packing and other absorbent materials immediately surrounding an individual waste container.

Graphic representations of the spent fuel and high level waste glass containers are given in Figure 8.3.5.9-1. The design configurations for both vertical and horizontal emplacement are shown in Figure 8.3.5.9-2.

Technical interpretation

The DOE understands substantially complete containment to mean that the set of waste packages will fully contain the total radionuclide inventory for a period of 300 to 1,000 years following permanent repository closure, allowing for recognized technological limitations. Implementation of this understanding will be based solely on reliance on the waste package as the major component of the engineered barrier system. The container is the primary barrier of the multiple barrier system for the purpose of containment of radionuclides. The waste package will be designed to be resistant to the degrading effects of the repository environment under anticipated processes and events. Containment will be based on the ability of the waste package, by virtue of its intrinsic properties and design, to maintain a continuous, sealed barrier around the waste.

The DOE intends to design the waste packages to provide total containment of radionuclides for a period of 300 to 1,000 yr after permanent closure of the repository. In a practical sense, however, considering the large number of waste packages, the large area of the repository horizon, and the long time period involved, it is not possible to precisely predict or demonstrate the endurance of an individual waste package. It is also reasonable to expect that some small number of packages will prematurely lose containment. The DOE will develop and conduct a test program to collect the

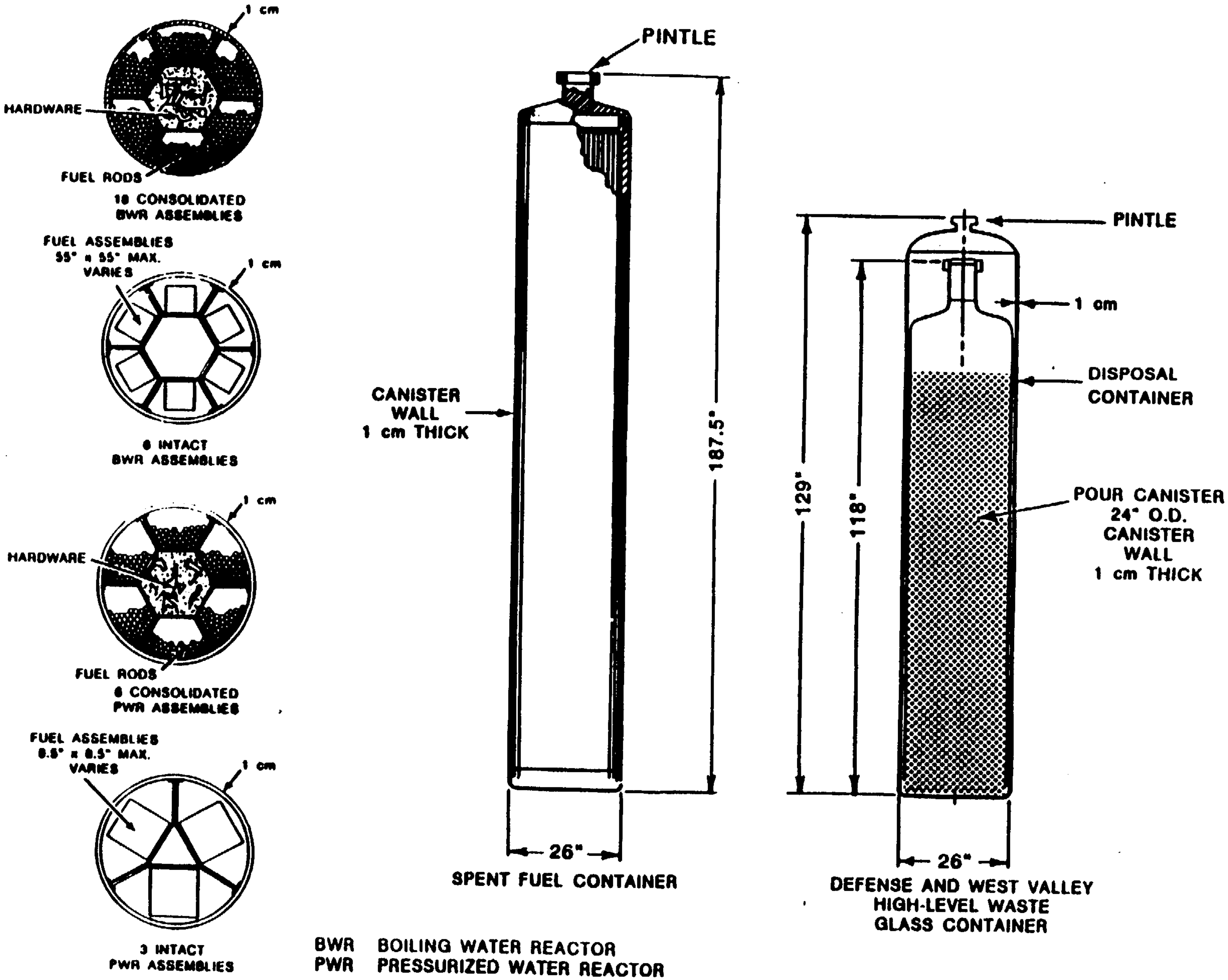


**Figure 8.3.5.9-1.** Spent fuel and high-level waste glass containers.



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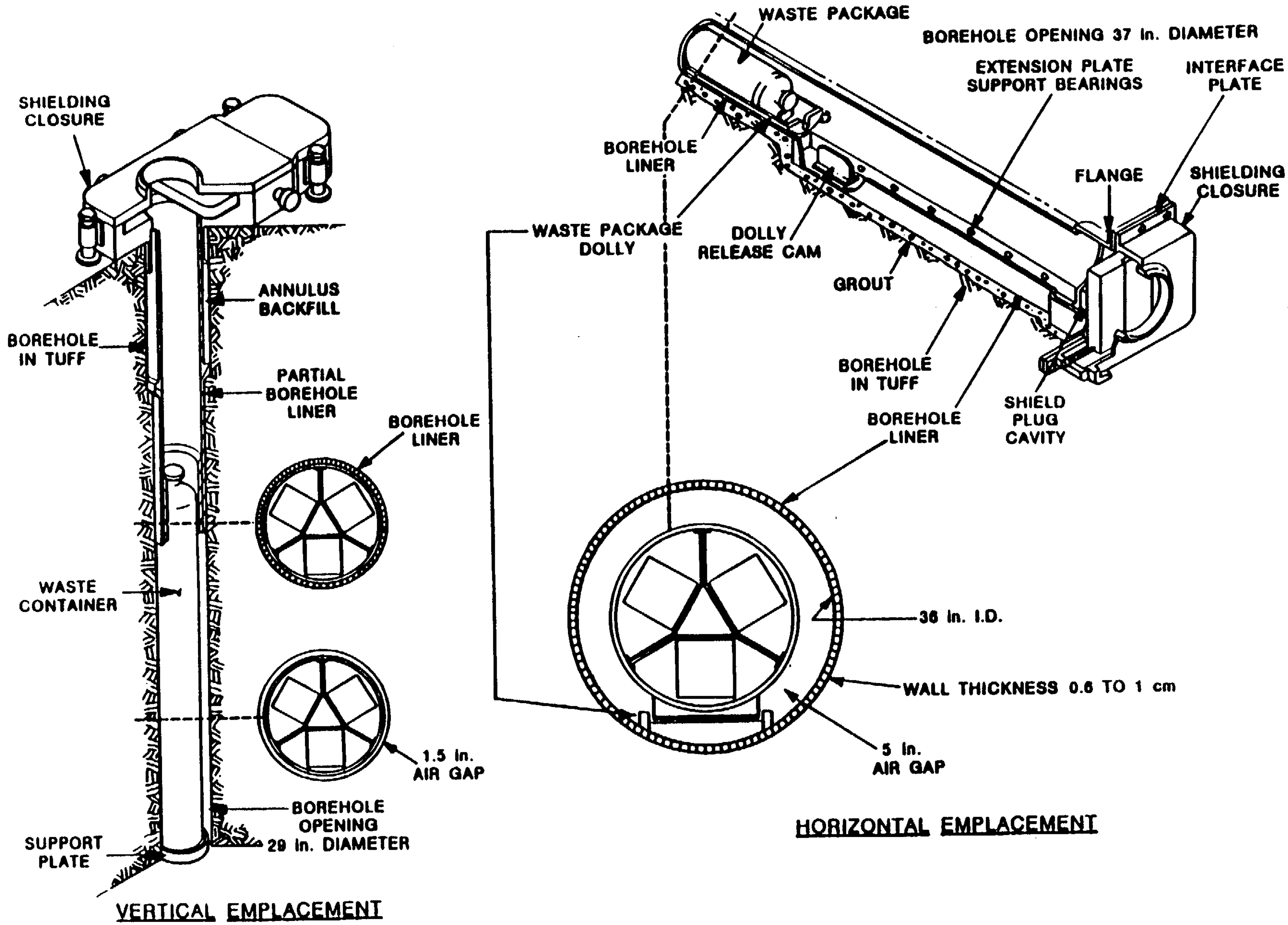


**Figure 8.3.5.9-2.** Design configurations for vertical and horizontal emplacement.

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necessary information that will enable the designers to select materials and design the waste packages in a manner that will reduce the incidence of failure during the containment period to a reasonable minimum.

The DOE expects that the performance of the waste package during the containment period will be best achieved by minimizing the residual uncertainties. The residual uncertainties in predicting performance are due to several factors: (1) the inherent limitations associated with manufacturing, handling, and emplacement operations, (2) the uncertainty in developing a complete understanding of the behavior of waste package materials, and (3) the uncertainty in predicting the future environment of each waste package. These factors are recognized in the NRC Staff Analysis of Public Comments on Proposed Rule 10 CFR Part 60 (NUREG-0804), where it is stated that

> the staff does not intend that the containment time requirement be achieved absolutely for all of the waste (i.e., absolute proof of zero release for 1,000 years is not required). It is expected that containment of the waste will be substantially complete, with release during the containment time limited to a small fraction of the inventory present. It is intended that the waste package design have a high reliability, taking into account anticipated processes and events that would affect package performance. It is realized that a small fraction of the approximately 100,000 packages will be breached before 1,000 years due to variations in materials manufacturing processes, etc., that can only be estimated using statistical procedures. Similarly, a significant fraction of the packages may remain intact for much longer than 1,000 years.

More specifically, these uncertainties can be divided into preclosure and postclosure considerations. During the preclosure repository operation, the DOE will manufacture waste packages in accordance with detailed design specifications. Waste packages will be loaded, sealed, inspected, and moved through the repository surface and subsurface facilities, and be emplaced into boreholes for final disposition, using detailed operating procedures. The DOE will have in place a quality assurance program, including quality control (QC) procedures, that will ensure that emplaced waste packages meet detailed material, fabrication, closure, surface finish, and handling specifications. Even with a fully qualified QC program, however, it cannot be ensured with absolute certainty that packages with undetected flaws will not be emplaced. Throughout the preclosure period, appropriate monitoring will be conducted as part of the performance confirmation program to ensure that the waste packages "are functioning as intended and anticipated."

During the postclosure period, the performance of any waste package cannot be accurately predicted over the long time period of the performance objective because of (1) the problems associated with demonstrating the mechanisms of all possible material degradation modes under the range of future environmental conditions and (2) the difficulties in extrapolating short-term experimental data to predict long-term performance. Therefore, it is the goal of the waste package program to provide for complete containment, allowing for only residual uncertainties. The DOE will minimize the uncertainties associated with the technical limitations for the postclosure period through a defense-in-depth concept. This concept introduces conservatism in

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demonstrating waste package performance through bounding assumptions, using multiple barriers to limit container degradation and waste form releases, and evaluating alternative materials and designs.

Design objective

The DOE will design the waste packages to provide total containment of the enclosed waste for the containment period under the full range of anticipated repository conditions. In addition, the DOE will use design features of the waste package to ensure, for any waste packages that prematurely fail, that (1) a large fraction of the radioactivity will be contained within the set of waste packages for the duration of the containment period and (2) any radioactivity released from the ensemble of waste packages will be released at a very low rate, relative to the total inventory. The waste packages, therefore, will be designed to provide a reasonable expectation that, should any individual waste package fail at any time following permanent closure, releases of radioactivity from the engineered barrier system will occur at very low rates.

The DOE has developed a performance allocation process that is the basis for the testing program. The process is designed to reduce uncertainties in demonstrating waste package containment through a comprehensive in situ and laboratory testing program. The performance allocation process identifies the system elements that contribute to the demonstration of substantially complete containment and that provide assurance that releases of HLW occur at very low rates. These elements include the engineered environment, the waste containers, and the waste forms. The performance allocation process also establishes the sensitivity allowed in testing parameters and explains the needed evaluations and assessments to show that uncertainties are minimized. Finally, the process considers possible material or design alternatives that may be used to supplement or replace the reference design. These alternatives include selection of various container materials and the use of alternative designs such as inner liners to contain significant radioactive gases and diffusion barriers to limit the inflow of water and the egress of radionuclides. For the purposes of the test program, however, the duration of the containment period, the fraction of the radioactivity that can be retained within the set of waste packages, the number of waste packages that can be reasonably expected to provide total containment, and the rate of release from any failed waste packages during this period cannot be reasonably determined until the site is sufficiently characterized and additional information is available regarding the performance of waste packages subject to the conditions of the site.

Testing program

In recognition of the limitations and uncertainties that prevent achieving complete containment, design and materials testing activities have been developed to quantify the expected performance of the waste packages. In order to build a comprehensive testing program, the DOE has developed quantitative estimates of system performance as a first step in the testing, design, and performance assessment process. It is important to note that these estimates are tentative. Their sole purpose is to allocate importance to each of the system elements and thus enable the DOE to develop an accept-

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able testing program. The detailed allocations to each of the system elements is discussed later in this section.

The technological limitations inherent in package fabrication, closure, and inspection are addressed in the process reliability assessments that will be conducted in support of resolving Issue 4.3 (Section 8.3.4.4).

Additional limitations associated with the repository handling and emplacement operations that may have an effect on subsequent containment performance are discussed in conjunction with Issues 1.11 (Section 8.3.2.2) and 4.4 (Section 8.3.2.5).

The waste package materials testing activities are designed to aid in evaluating the uncertainties in the behavior of the materials under anticipated repository conditions. Those activities associated with the container materials are discussed in this section. The waste form testing activities are described under Issue 1.5 (Section 8.3.5.10). Similarly, uncertainties will exist in the characterization of the near-field environment. The activities aimed at quantifying the remaining uncertainties are described under Issue 1.10 (Section 8.3.4.2).

Inherent in the resolution of the containment issue is the requirement to predict the performance of the waste packages over the entire duration of the containment period. This requirement will necessitate predictive models that cannot be fully validated and will therefore contain additional residual uncertainties. The models that support predictions of the container performance are discussed in this section. Waste form and overall waste package performance assessment models, including sensitivity analyses, are described under Issue 1.5 (Section 8.3.5.10)

Figure 8.3.5.9-3 shows the hierarchy of models to be developed and employed in resolution of the issues relating to design and performance of the waste packages. To avoid duplication in the SCP of description of the development of the numerical models and the testing activities that provide their bases, the discussion in this section is limited to the models and submodels that are highlighted in the figure. These are the models that supply the simulations of the performance of the containers. The various other models needed to complete the predictions for containment are described under Issues 1.5 (Section 8.3.5.10) and 1.10 (Section 8.3.4.2). The testing and design activities described in this section are tentative and are subject to change. Any such change will be reported in semiannual progress reports.

This issue, as stated, is restricted to assessing waste package performance under anticipated processes and events, and only for the period up to 1,000 yr following closure of the repository. This is based on a performance allocation approach described below. Figure 8.3.5.9-4 shows the performance allocation approach to resolving this issue. The performance measures and goals are shown in Table 8.3.5.9-1. However, the performance of the waste packages during the containment period is intimately linked to the performance required thereafter by the engineered barrier system in controlling radionuclide releases in Issue 1.5 (Section 8.3.5.10). The level of performance needed during the containment period to establish conditions that will provide the required release rate control thereafter may require different goals than those used to resolve this issue. Other issues need information

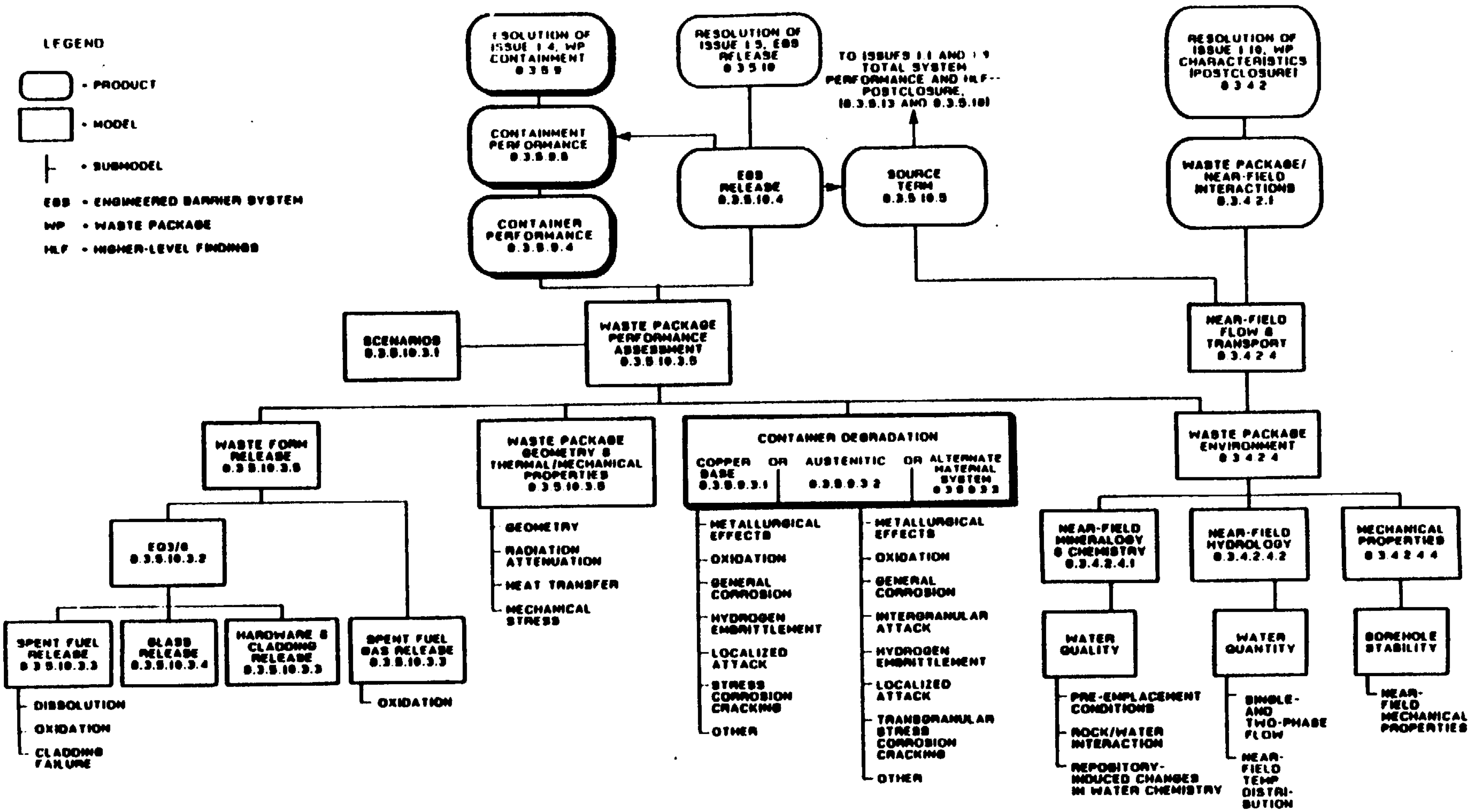


**Figure 8.3.5.9-3.** Model hierarchy for Issue 1.4 (containment by waste package).

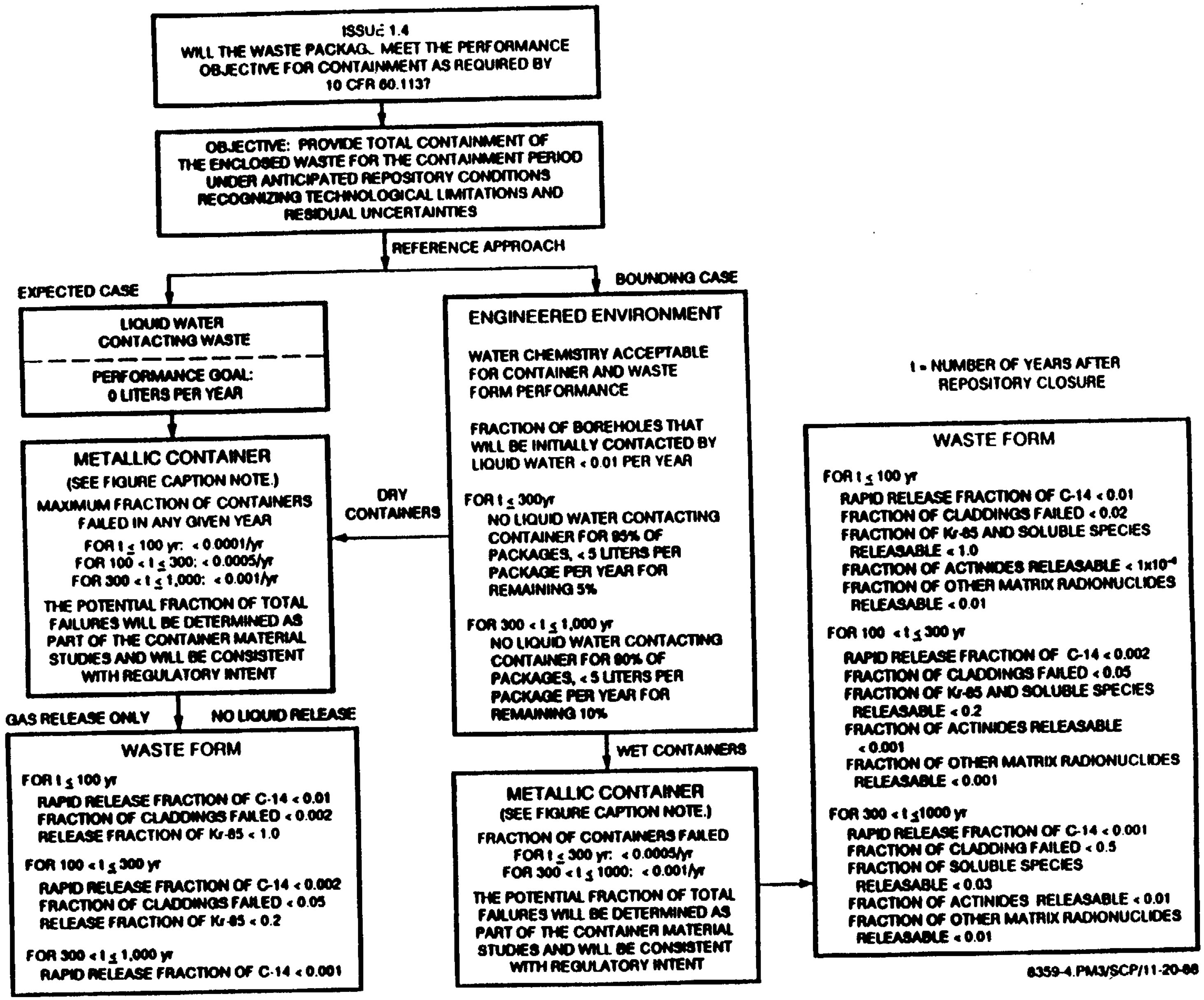


**Figure 8.3.5.9-4.** Reference approach for Issue 1.4 (containment by waste package). For metallic container failures, simply multiplying the fraction of failed container values by the time durations does not correctly estimate the total (cumulative) number of container failures that can be tolerated during the containment period (see page 8.3.5.9-25)



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Table 8.3.5.9-1. Performance measures and goals for Issue 1.4 (containment by waste package)

| System element | Performance measure | Tentative goal[a] | Needed confidence |
|---|---|---|---|
| Engineered environment[b] | Quantity of liquid water that can contact the container | For t ≤300: No liquid water contacting the container for 95% of packages, <5 L per package per year for the remaining 5% | High |
| | | and | |
| | | <1.0%/yr of the total number of emplacement hole walls will be initially contacted by liquid water | High |
| | | For 300 < t ≤ 1000: No liquid water contacting the container for 90% of packages, <5 L per package per year for the remaining 10% | High |
| | | and | |
| | | <1.0%/yr of the total number of emplacement hole walls will be initially contacted by liquid water | High |
| | Quality of liquid water that can contact the container | Constrain water chemistry to acceptable levels for performance of container and waste form | High |
| | Rock-induced load on waste package | Load less than design basis (see Table 8.3.4.2-3) | High |

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Table 8.3.5.9-1. Performance measures and goals for Issue 1.4 (containment by waste package) (continued)

| System element | Performance measure | Tentative goal[a] | Needed confidence |
|---|---|---|---|
| Container | Maximum fraction of containers that failed in any given year[c] | For containers with no liquid water contact: | |
| | | For $t \leq 100$: $< 0.0001$/yr | High |
| | | For $100 < t \leq 300$: $< 0.0005$/yr | High |
| | | For $300 < t \leq 1{,}000$: $< 0.001$/yr | High |
| | | For containers with liquid water contact: | |
| | | For $t \leq 300$: $< 0.0005$/yr | High |
| | | For $300 < t \leq 1{,}000$: $< 0.001$/yr | High |
| Waste form | Cumulative release of radionuclides from the ensemble of breached packages | For $t \leq 300$ yr: $<2.0 \times 10^{-2}$ of the total curie inventory of the ensemble of breached packages | High |
| | | For $300 < t \leq 1000$: $<1 \times 10^{-2}$ of the total curie inventory of the ensemble of breached packages | High |

[a]t = years after repository closure.

[b]Envelope for anticipated processes and events.

[c]Failure is defined as a breach allowing air flow of $1 \times 10^{-4}$ atm-cm$^3$/s. A value for the limit of cumulative failures will be determined as part of the container material studies and will be consistent with regulatory intent.

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on the performance of the containers for longer time periods and under both anticipated and lower probability scenarios. These other issues are as follows:

1. Issue 1.1: This system performance issue needs information on predicted time to loss of containment by the waste packages for times up to 10,000 yr after closure due to both anticipated and unanticipated processes and events.

2. Issue 1.5: This issue addresses the release rates of radionuclides from the engineered barrier system, assuming anticipated processes and events for 1,000 to 10,000 yr after closure. The condition of the waste forms and containers will affect those release rates.

3. Issue 1.9: This issue addresses the higher-level findings that support site selection. Calculations of predicted releases to the accessible environment for 100,000 yr are required. These calculations will use release rate information from the engineered barrier system that is affected by the condition of the containers.

These issues are addressed in Sections 8.3.5.13 (Issue 1.1), 8.3.5.10 (Issue 1.5), and 8.3.5.18 (Issue 1.9).

<u>Approach to resolving the issue</u>

To resolve this issue, the DOE will use the following approach to the development of the engineered barrier system:

1. Enhance the natural features of the unsaturated zone repository by engineering the local environment to conditions favorable to waste package integrity.

2. Evaluate waste package container design to provide a highly reliable sealed containment barrier around the waste for at least 1,000 yr over the full range of repository conditions.

3. Evaluate alternative design concepts and materials and select a final design based on a comparison of waste isolation capabilities and other relevant factors.

4. Execute a thorough testing, evaluation, and characterization program (following approved quality assurance procedures) to evaluate waste package designs and estimate their expected performance in the repository.

5. Fabricate and close waste package containers using detailed specifications and procedures including stringent quality controls, to ensure high reliability in postclosure performance.

6. Identify uncertainties that influence performance predictions through performance assessment, quantify or bound the uncertainties, and then reduce them to a practical minimum through testing and performance confirmation.

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7. Utilize the characteristics of the waste form in conjunction with the other engineered waste package components and the unsaturated zone environment to ensure that any releases that may occur during the containment period occur at low rates.

The DOE considers that the activities just outlined will result in an engineered barrier system design that, through its many complementary and redundant characteristics, will satisfy all the criteria of 10 CFR 60.113, and in doing so, will resolve this issue with a high degree of assurance.

Engineered environment enhancement

As discussed in Section 8.4.1.1, the unsaturated zone environment is naturally dry, and it is likely to remain that way for 10,000 yr and more. This is expected to be confirmed by site characterization. Moreover, the characteristics of the tuff rock in the repository horizon are such that movement of water occurs generally by matrix flow rather than by flow in fractures. Since most waste package degradation modes and waste transport modes depend upon the presence and movement of ground water in the vicinity of the waste, these features create a very favorable environment for waste disposal.

The DOE plans to incorporate several additional features into the engineered barrier design to further enhance the natural characteristics of the unsaturated zone environment:

1. Construction and operation of the repository will further "dry out" the repository host rock by entrainment of the moisture in the air moved through the repository by the ventilation system.

2. The decay heat produced by the high-level waste will be used by designing the arrangement of emplacement locations to raise the temperature of the host rock above the local boiling point of water and to maintain it above that point for hundreds of years for most of the waste packages.

3. Other features, such as an air gap between the host rock and the waste containers, may be used to further militate against the contact of water with the waste packages.

4. Precautions will be taken to minimize changes to the water quality that would be deleterious to postclosure performance. Performance parameters and goals for water quality are given in Table 8.3.5.9-2.

Activities and performance allocation related to these factors are discussed in Section 8.3.4.2 under Issue 1.10.

Sealed containment barrier

The DOE has established, as a design basis, that a sealed barrier, a container, will be maintained around the waste for 1,000 yr following repository closure. This sealed barrier will be designed to survive without breach over the full range of expected repository environmental conditions.

Table 8.3.5.9-2. Water quality performance parameters and goals for Issue 1.4 (containment by waste package) (page 1 of 2)

| Performance measure | Performance parameter | Tentative goals[a] | Needed confidence | Current estimated range | Current confidence |
|---|---|---|---|---|---|
| Quality of liquid water that can contact the container | pH | 5.5 to 9 | High | 6.1 to 7.7 | Medium |
| | $Cl^-$ | <20 ppm | High | <10 ppm | Medium |
| | $F^-$ | <6 ppm | High | <5.4 ppm | Medium |
| | $NO_3^-$ | <15 ppm | High | 0 to 11 ppm | Medium |
| | $SO_4^{2-}$ | <50 ppm | High | 15 to 35 ppm | Medium |
| | $CO_3^{2-}$, $HCO_3^-$ | <200 ppm | Medium | 90 to 160 ppm | Medium |
| | Total anions | <220 ppm | Medium | 110 to 160 ppm | Medium |
| | Organics | TBD[b] | TBD | NA[c] | NA |
| | Colloids | TBD | TBD | NA | NA |
| | $O_2$ | 0.1 to 8 ppm | High | <6.5 ppm | Medium |
| | $NH_3$ | <1 ppm | High | <1 ppm | Low |
| | $Si^{4+}$ | >20 ppm | High | 20 to 550 ppm | Medium |
| | $Na^+$ | <100 ppm | High | 30 to 80 ppm | Medium |
| | $K^+$ | <50 ppm | High | 1 to 30 ppm | Medium |
| | Na/Ca | >1 | High | >2 | Medium |



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Table 8.3.5.9-2. Water quality performance parameters and goals for Issue 1.4 (containment by waste package) (page 2 of 2)

| Performance measure | Performance parameter | Tentative goals[a] | Needed confidence | Current estimated range | Current confidence |
|---|---|---|---|---|---|
| | Total heavy metals[d] | <2 ppm | High | NA | Low |
| | Total other cations | <50 ppm | High | <30 ppm | Low |

[a]Not all combination of the limits on the goals given will result in acceptable water chemistries; see Section 8.3.4.2.
[b]TBD = to be determined.
[c]NA = not available.
[d]Atomic number >Fe.



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In practice, because of uncertainties associated with the long time spans and technological limitations, not all containers will remain unbreached for 1,000 yr. But the actual fraction of containers that may breach will be a small number. Much of the waste package testing program activities will be aimed at determining the actual number with high confidence, and reducing that number to the lowest practical level.

Alternative designs

To be consistent with the requirements (10 CFR 60.21(c)(1)(ii)(D)), the DOE will evaluate alternative waste package and engineered barrier system designs, including material selections. Each design will provide a highly reliable containment barrier over the full range of repository conditions. Comparative evaluations of alternatives will be made at various points in the development process, with selections being made among them based on their relative waste isolation capabilities and other relevant factors. It is intended that these alternative designs will enhance the capability of waste isolation by reducing the sensitivity to residual uncertainties in the service environment.

Testing program

Performance is allocated to the container, waste form, and engineered environment as part of an overall strategy to ensure compliance with the substantially complete containment requirement. However, the emphasis on providing containment is placed on the waste container. The testing program under Issue 1.4 involves both literature and laboratory studies directed at bounding the uncertainties on container performance. As described earlier, these uncertainties include (1) preemplacement limitations, such as fabrication and handling, (2) the inability to definitively quantify waste package material performance, and (3) uncertainties in the near-field environment surrounding each waste package. The conceptual design of the waste package is based on the current understanding of the anticipated repository conditions (see Chapter 7). The test program is part of the iterative test, assessment, and design process that may be modified as more and better information is obtained during site characterization. As part of the development of detailed plans for testing, the DOE will determine a statistical basis for the number and types of tests conducted under appropriate activities to the extent practical. Information from other areas of investigation, such as Issues 1.5 (Section 8.3.5.10), 1.10 (Section 8.3.4.2), 2.6 (Section 8.3.4.3), and 4.3 (Section 8.3.4.4), will all be considered in the final waste package design. A full appreciation of the design process can only be gained by understanding all waste package performance and design issues. In addition, analyzed data from Issue 1.4 will be used to resolve other issues, including Issue 1.1 (Section 8.3.5.13), which will demonstrate total system performance over 10,000 yr, and Issue 1.5, the gradual release requirement of 10 CFR 60.113.

The current reference container design is based on a corrosion-resistant container fabricated from one of six possible metals, three iron-based or high-nickel austenitic alloys, and three copper or copper alloys. In addition, alternative materials and concepts are being evaluated, including other metal systems, ceramics, coatings, and fillers. Because of the possible range in the postemplacement repository environment and the preemplacement

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(fabrication, assembly, and handling) conditions affecting postemplacement processes, it is important to fully understand those features of the waste package container affecting performance. The mechanical, physical, and microstructural properties of the container base metal, welded materials, and exposed surfaces will be described. General and localized corrosion will also be evaluated. These results will aid in the understanding of the performance of the as-emplaced container and will be important inputs for the container material selection process, along with the characterization of the likely modes of container breach, and the modeling of container performance. These models will be used to assess (1) the rate of container degradation in the repository environment, under both anticipated and unanticipated processes and events, and (2) the failure rate of the containers over time, using both deterministic and probabilistic approaches.

The results of this work will be used to determine compliance with the substantially complete containment requirement and as input into the activities addressing gradual releases of radionuclides over 10,000 yr under Issue 1.5. Analyses will be conducted in order to understand and reduce the uncertainties associated with different waste package designs. Also, the DOE believes that releases from any failed containers will occur at a very low rate and, therefore, will meet the substantially complete containment requirement. To address this, the DOE will combine the results of studies, from Issue 1.5, that will predict the performance of the spent fuel and glass waste forms over all times up to 10,000 yr, with information from Issue 1.4, which emphasizes container performance.

Waste package design and fabrication program

The objective of the waste package design and fabrication program is to provide waste package containers of high quality that can be fabricated, closed, and inspected using available and accepted practices. This program includes the following activities: parametric studies to aid material selection, evaluation and selection of manufacturing processes, setting appropriate specifications for these materials and processes, and developing inspection techniques. These activities will be conducted under a sound quality control program. Other activities include fabricating full-scale prototypes and designing and implementing a program for monitoring the performance of representative waste packages during the repository preclosure period as a part of the performance confirmation program. Thermal parametric studies will also be conducted to evaluate the effects of variations in thermal properties, emplacement configuration, and heat transfer characteristics on waste form, container, and near-field rock temperatures. These studies will provide a basis for designs that are consistent with the postclosure containment strategy.

The container fabrication process development consists of multiyear multiphase activities to assess alternatives and to recommend and demonstrate a method for fabrication of containers through production of full-scale prototypes. The container final closure development activity also involves a multiyear effort to recommend and demonstrate joining methods. Emphasis will be placed on a simple, reliable, maintainable system that (1) will provide the required throughput to support the projected disposal container production schedule, (2) is capable of operation in the repository hot-cell, and

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(3) will produce a defect-free closure that has a microstructure suitable for nondestructive evaluation (NDE).

Ultrasonic and dye penetrant techniques have been tentatively selected for NDE of the metallic container final closure. Similar techniques will be selected for alternates as appropriate. Techniques, such as mechanical testing or metallographic inspection, are destructive in nature and will be used on statistically sampled containers at the point of manufacture. Others, such as ultrasonic inspection for defects, are nondestructive and will be comprehensively performed on all the waste packages.

Container materials and prototypes will be procured for testing activities in accordance with detailed specifications. To ensure a high level of quality, materials and prototype container evaluations will include mechanical properties, chemical composition, microstructures, surface finish and cleanliness, closure quality, structural integrity, and physical dimensions. Other measurements may be required as the waste package designs evolve.

Provision will be made for transfer and rework or repackaging of the contents of completed packages that fail final inspection. Containers suspected of being damaged in handling at the repository after final inspection is completed would be reinspected and, if appropriate, disassembled, and the contents would be transferred to new containers.

These activities will ensure that the final products will perform as intended and serve their function to contain the radionuclides during the containment period.

Assessment and reduction of uncertainties

The stated containment performance goal, in effect, is that containment will be total, recognizing practical technological limitations. This goal requires (1) the calculation of the degree of containment, (2) the identification of sources of uncertainties, and (3) the quantification, to the extent practicable, of the contributions of each source of uncertainty to the overall uncertainty. In addition, once the contributing sources of uncertainty have been evaluated, the site characterization and experimental programs need to be reevaluated to determine what can reasonably be done to reduce those significant sources of uncertainties.

One methodology for analytically addressing containment is reliability analysis. This methodology has been suggested as an acceptable approach for addressing the regulatory containment requirement by the NRC staff in their "Generic Technical Position on Waste Package Reliability Analysis" (NRC, 1985). Reliability is the probability that a system or component, when operating under stated environmental conditions, will perform its intended function adequately for a specified interval of time. The NRC staff is, therefore, suggesting a probabilistic approach be taken to address the deterministic containment requirement. In Chapter 7 of this SCP, the DOE stated its intent to use an appropriate reliability analysis approach.

The analysis of uncertainty supporting the determination of waste package reliability will follow a systematic approach as recommended in the

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conclusions of the Nuclear Energy Agency's workshop on uncertainty analysis (NEA, 1987). This means that the analysis of reliability, in terms of predicting containment time and evaluating the uncertainty in that prediction, will include the following aspects:

1. Combining deterministic modeling, probabilistic analyses, and uncertainty analyses to determine waste package reliability.

2. Using combinations of quantitative methods, recognizing the uses and limitations of each method and its results.

3. Minimizing the use of nonquantitative uncertainty analysis methods.

4. Using sensitivity analyses to identify important sources of uncertainty in parameters.

5. Treating correlations between parameters as part of the uncertainty analysis.

6. Systematically documenting and properly identifying the input data and process used to create subjective probability density functions describing parameter uncertainty.

7. Using quality assurance procedures to document and verify codes and to validate models to the extent practicable, taking into account their use and relative importance in demonstrating regulatory compliance.

In terms of minimizing uncertainties, combining uncertainty and sensitivity analyses appropriately will allow the modeling effort to feed information back to the design and testing effort regarding priorities in reducing those uncertainties that can be experimentally addressed. The iterative nature of this issue resolution strategy becomes evident in the identification of those sources of important uncertainties that may be amenable to reduction through experimental or design changes. It is expected that, at the time of licensing, this iterative approach will have determined that the preferred waste package design will meet the established performance goals. Given the ongoing in situ testing program that can help define expected conditions more closely, and the ongoing performance confirmation program with its continuing monitoring and testing, it would be expected that the uncertainties in the application for a license amendment for final closure of the repository would present waste-package containment-performance estimates with significantly reduced uncertainties.

Tentative goals for releases from the waste packages

As noted earlier, despite the best efforts during design, fabrication, handling, and emplacement, a small number of containers may be expected to breach during the containment period. Thus, a demonstration of substantially complete containment must inevitably address possible releases during the containment period, as well as possible alterations of the waste form that may have an effect on subsequent releases after the containment period.

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In 10 CFR Part 60, the NRC requires that any releases from the engineered barrier system be at a low rate. However, the NRC provided a numerical criterion only for releases following the end of the containment period. From the requirement for substantially complete containment, it is clear that the NRC intended that releases during the containment period also be low.

The DOE considers it appropriate to require that releases of isotopes with long half-lives from the waste packages be controlled at a stricter standard during the containment period than during the post-containment period. Thus, to guide the testing program, the DOE has established the tentative criterion that release of these isotopes from the waste packages will be controlled such that their annual rates of release are each less than 1 part in 1,000,000 for those isotopes present in sufficient quantity in the 1,000-yr inventory. The isotopes for which this criterion applies are listed in Table 8.3.5.10-3b. In establishing the requirement for substantially complete containment for 300 to 1,000 yr, the NRC indicated that this was intended to provide a period of greater isolation when concentrations of fission products were at their highest levels. However, no specific quantitative guidance was provided for releases during the containment period. Consistent with public health safety requirements, the DOE has tentatively elected to limit releases of all other radioactive isotopes to an annual release rate of less than 1 part in 100,000 of the current inventory of that isotope in the ensemble of all the waste packages. The performance parameters related to this performance measure on the waste form are given in Table 8.3.5.9-3.

Performance allocation

Performance is allocated to the engineered environment to provide a situation favorable to the performance of both the container and the waste form. The reference approach includes branches for both the expected case, in which no water contacts the container, and a bounding case, in which increasingly more containers could be exposed to an increasing but still limited amount of water with time. The in situ conditions provide a host rock only partially saturated with water and at atmospheric pressure, and a very low downward flux of water. These conditions are expected to apply over the range of all anticipated processes and events. For these conditions, the thermal field developed by the waste package thermal loading and the repository emplacement configuration will raise the temperature of the near-field rock above the boiling point, drying it out, and retarding the return of liquid water. Because of this combination of natural and engineered features, performance goals are set for the amount of liquid water per year that can contact the container, for the rate at which conditions permitting liquid water to contact packages is established, and for the chemical quality of the water. After cooling below the boiling point, most waste packages are not expected to be exposed to liquid water because of the limited water flux available in the host rock, the heat generation from the packages, and the air gap over most of the interface between the packages and the host rock. To provide bounding assumptions to control performance allocation to other system elements, bounding values of 5 L of water per package per year contacting 5 percent of the packages during the first 300 yr after closure, and 5 L of water per package per yr contacting 10 percent of the waste packages for the period from 300 to 1,000 yr after closure is assumed. This

Table 8.3.5.9-3. Waste form performance parameters and goals for Issue 1.4 (containment by waste package) (page 1 of 3)

| Performance measure | Performance parameter | Tentative goals | Needed confidence | Current estimated range | Current confidence |
|---|---|---|---|---|---|
| | | GLASS WASTE FORM | | | |
| Release rate from the ensemble of breached packages | Annual fraction of the radionuclide inventory in effluent solution from failed containers | For t ≤ 300 yr: $<2 \times 10^{-4}$ per yr | High | $<1.0 \times 10^{-4}$ per yr | Medium |
| | | For 300 < t ≤ 1,000 yr: $<1 \times 10^{-4}$ per yr | High | $<1.0 \times 10^{-5}$ per yr | Medium |
| | | SPENT FUEL WASTE FORM | | | |
| | Fraction of cladding failed in failed containers[a] | For t ≤ 100 yr: <0.02 failed | High | 0.001 to 0.02 | Medium |
| | | For 100 < t ≤ 300; <0.05 failed overall | High | 0.001 to 0.10 | Medium |
| | | <0.02 failed while dry | High | 0.001 to 0.10 | Low |
| | | For 300 < t ≤ 1,000: <0.5 failed overall | High | 0.1 to <0.9 | Low |



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Table 8.3.5.9-3. Waste form performance parameters and goals for Issue 1.4 (containment by waste package) (page 2 of 3)

| Performance measure | Performance parameter | Tentative goals | Needed confidence | Current estimated range | Current confidence |
|---|---|---|---|---|---|
| | **SPENT FUEL WASTE FORM** (continued) | | | | |
| | Fraction of cladding failed in failed containers[a] (continued) | <0.02 failed while dry | High | 0.001 to 0.10 | Low |
| | Fraction of total inventory of gap and grain boundary elements available for rapid release from unoxidized fuel[b] | <0.02 | High | 0.005 to 0.04 | Medium |
| | Fraction of C-14 inventory available for rapid release as a gas | <0.01 | High | 0.002 to 0.02 | Low |
| | Solubility of U, Pu, and Am | For $t \leq 300$: $<2 \times 10^{-4}$ of a package inventory of these elements per L of water | High | $<1 \times 10^{-6}$ per L | High |




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Table 8.3.5.9-3. Waste form performance parameters and goals for Issue 1.4 (containment by waste package) (page 3 of 3)

| Performance measure | Performance parameter | Tentative goals | Needed confidence | Current estimated range | Current confidence |
|---|---|---|---|---|---|
| | **SPENT FUEL WASTE FORM (continued)** | | | | |
| | Solubility of U, Pu, and Am (continued) | For 300 < t ≤ 1,000: <5 x $10^{-4}$ of a package inventory of these elements per L of water | High | <1 x $10^{-4}$ per L | High |
| | Fractional release from failed containers of all other radionuclides not in the rapidly released gap and grain boundary inventory | For t ≤ 300: <2 x $10^{-2}$ | High | <$10^{-2}$ | Medium |
| | | For 300 < t ≤ 1,000: <1 x $10^{-2}$ | High | <$10^{-3}$ | Medium |

[a]Numerical definition of cladding failure is to be determined.

[b]Fraction of total inventory of gap and grain boundary elements available for rapid release from oxidized fuel will depend on the degree of oxidation and other fuel conditions. For the purpose of the performance allocation a conservative value of 1.0 is used.

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goal is consistent with that set in Issue 1.10 (Section 8.3.4.2), where the basis for selecting the goal and the performance parameters and model inputs that will be used to achieve this goal are discussed in more detail. A characterization goal is set for the mode of water flow into the borehole, to ensure that processes connected with fracture flow and concentration of the salts carried in low concentrations by the ground water do not upset the simple bounding process described (see discussion of the design envelope in Section 8.3.4.2). Performance is allocated to limit the rock-induced load to an amount accommodated by the waste package design. The loads may arise from block movement due to the rock responding to gravitational forces and the thermal cycle.

Performance is allocated to the waste package container to meet the design objectives. The containers will be designed with a design life goal that is consistent with the duration of the containment period. However, it is recognized that some preclosure container breaches will escape detection and that a very small fraction of containers will breach during the containment period. These breaches may not constitute failure since failure is defined as a breach large enough to allow significant air flow ($1 \times 10^{-4}$ atm-$cm^3$/s) into the container. The values given in Table 8.3.5.9-1 represent those that are conservative compared with those that are presently attainable given today's state of technology. This test is a general standard accepted by the nuclear industry.

Performance is allocated to the waste form, including the cladding of spent fuel, to aid in retaining radioactivity inside the waste packages and limiting radioactivity release rate from the engineered barrier system. Glass waste forms can release radionuclides only through alteration and transport by liquid water. The glass waste form, when exposed in failed packages, is allocated performance limiting the rate of release from the failed package; this rate is less stringent than the performance goal set (in Issue 1.5) for the controlled release period and is expected to be achieved.

Spent fuel has several potential modes of waste release; hence, performance parameter goals have to be set to limit the fractions of the total radioactivity available for these release modes to meet the waste form performance goal. The performance allocations change over time since the proportions of different radionuclides in the total inventory change over time. The fuel cladding is also allocated performance for several purposes. During the first 100 yr after closure, when there is still a significant amount of Kr-85 gas in the spent fuel, the intact cladding can help contain this nuclide. During the first 300 yr after closure, the fraction of intact cladding under liquid exposure conditions can help limit the release rate of Co-137 and Sr-90, the major components of the radioactivity inventory during that period. For time periods when the fuel is still hot enough to oxidize appreciably if exposed to air (this temperature range is well above the boiling point of water), the intact cladding can prevent exposure of the fuel matrix to air. The performance assigned to the cladding while still dry is improved by the absence of liquid-based corrosion modes.

The reasons these performance measures and their goal values were selected can be clarified by examining the strategy for satisfying the substantially complete containment requirement of this issue.

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The design objective, limiting the rate of radioactivity release from the waste package, relies on performance allocated to the engineered environment, the waste package container, and the waste form. The performance allocation differs for three time periods during the first 1,000 yr of the postclosure period as the proportions of radionuclides of different types change, the environmental conditions change, and the container and fuel cladding are exposed for longer times to potential failure modes. The specific performance allocations for the three time periods are discussed in detail below.

The bounding values for many of the parameters discussed in this and the following sections are not expected to occur. Insufficient information is available to select more realistic values at this time, but it is expected that data gathered during the site characterization program and by the testing programs on performance of waste forms and container materials will provide the basis for determining the bounds for anticipated processes and events.

Rationale for division of the containment period into segments. As noted in the preceding sections, the up to 1,000-yr containment period will be one of continuously changing environmental conditions and rapidly changing radionuclide inventory. One of the most significant environmental aspects of this period will be an early high temperature peak during the first 100 yr after emplacement of the waste, followed by a much more gradual decline in temperature. The types and quantities of radionuclides that contribute to the total radioactivity also undergo major changes during this period, with early times dominated by relatively short-lived fission products and the late times dominated by long-lived actinides. Because the environmental conditions play a large role in determining the performance of the various components of the waste package, and the changing makeup of the radionuclide inventory imposes different demands on the containment barriers as a function of time, the containment period has been divided into three subperiods: 0 to 100-yr postclosure; 100- to 300-yr postclosure; and 300- to 1,000-yr postclosure. The rationale for selecting these divisions and the performance allocated to key system elements during each subperiod are discussed below. This is followed in subsequent sections by more detailed discussions of the performance allocated to each system element and the justification for the goals set for these allocations.

The reference performance allocation case given in Table 8.3.5.9-1 and Figure 8.3.5.9-4 uses bounding conditions based on the present understanding of the repository emplacement environment, the expected performance of the waste forms in that environment, and the data available on the performance of metals in similar environments. In setting the performance goals, allowance has been made for the uncertainties in the site and materials properties data.

Section 8.3.4.2 describes activities to establish the waste package environment. In order to establish the performance allocations discussed herein, certain assumptions about the repository environment were made based on current understanding of the site. The "expected case" describes the environment that would exist if site characterization activities confirm this understanding. The "bounding case" is believed to approximate the most limiting, adverse conditions that are consistent with the repository horizon

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remaining in the unsaturated zone. The bounding case will be used as the initial design basis.

The DOE intends to design, produce, and emplace the waste packages to ensure that only a small number of containers will fail during the containment period, and recognizes that a very low average failure rate (with many years of zero failures) will be necessary to achieve this. The "maximum failures in any given year" shown in Figure 8.3.5.9-4 and Table 8.3.5.9-1 are not related to this average failure rate. Instead, they are intended to limit spike releases for atypically grouped failures, which would not be repeated on an ongoing basis. These values represent the maximum allowable container failure rates in any given year based on "rapid release" fractions of some radionuclides, which may occur without exceeding the release rate goals assumed above for the containment period. It is not correct to simply multiply these values by the time durations listed to derive an estimate of the total (cumulative) number of container failures that can be tolerated during the containment period. Additional studies will be conducted to establish a bounding value for the cumulative number of container failures allowable. Alternative designs will be evaluated with the selection made on the basis of their relative waste isolation capabilities, and other relevant factors.

0- to 100-yr postclosure. During this time period, the waste packages and near-field rock will experience the highest temperatures achieved during the postclosure period. The temperature of the borehole wall is expected to be well in excess of the boiling point of water for a large majority of packages. The effect of this thermal pulse in the environment will be to dry out the surrounding rock and thus preclude the possibility of liquid water contacting the majority of the waste packages. In the absence of liquid water, there are few credible mechanisms for producing failure of the containers and no mechanisms for the release of radionuclides other than those that can exist in a gas phase.

The thermal pulse is a direct result of the high radioactivity of the waste during this period. The dominant contribution to the total radioactivity comes from the nuclides Cs-137, Sr-90, and their very short-lived daughter products, Ba-137m and Y-90. These four nuclides alone account for about 85 percent of the total activity at the start of this period, declining to about 50 percent of the total by 100 yr after closure. Cesium and, to a lesser extent, strontium can migrate during reactor operation to grain boundaries and the pellet-cladding gap in the fuel where they are readily accessible for release when contacted by water. Less than 2 percent of the inventory of those readily soluble "gap and grain boundary" elements is expected to be in this form in unoxidized fuel.

The fuel in pins with failed cladding within failed containers will be contacted by oxygen in the repository air. Because of the high temperatures expected during this period, such conditions may result in the oxidation of the $UO_2$ fuel to higher oxidation states. This has two effects: (1) to increase the fraction of gap and grain boundary elements (i.e, cesium and strontium) that is available for subsequent rapid release in water and (2) to allow all the Kr-85 inventory in such oxidized fuel to be released rapidly as a gas. Fuel that might oxidize in the first 100 yr but does not contact water is assumed to have its entire inventory of gap and grain boundary

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elements available for rapid release at some later time. The conditions under which the gap and grain boundary elements can be released (conditions under which liquid water can enter a waste package, contact the waste form, and then exit) and the conditions under which the fuel can oxidize are mutually exclusive; if liquid water can contact the waste form, it will be too cold to oxidize the fuel significantly. Nevertheless, oxidation of the fuel early in the history of the repository will affect the performance of the spent fuel waste form at later times by increasing the size of the gap and grain boundary inventory of readily soluble elements like cesium, iodine, and technetium.

In addition, during this period, there are significant quantities of the radionuclides Kr-85 and H-3 (approximately 300 and 40 parts in 100,000 of the total inventory, respectively) present in the spent fuel waste form. In unoxidized spent fuel, about 1 to 2 percent of the Kr-85 may be present in the pellet-cladding gap as a gas and is available for immediate release without the mediation of liquid water. H-3 is thought to be fixed by the cladding and is unavailable for rapid gaseous release.

The requirements driving the performance goals set for the first 100 yr after closure are as follows:

1. Limit the quantity of fuel that can oxidize during this period to 2 percent of the total inventory of the failed containers.

2. Control the annual release of Cs-137, Sr-90, and their daughter products (as well as other gap and grain boundary elements).

3. Control the annual release of gaseous radionuclides (e.g., Kr-85).

The first requirement is met by the number of allowed cladding failures (less than 2 percent).

The second of these requirements is met by the combination of the goals for allowed container failure rates (0.0005 per yr), limited water availability (less than 5 percent of the packages being wet), number of cladding failures, and the total number of packages allowed to be initially contacted by liquid water in a single year (less than 1 percent per year). The need to control the release of the remaining fraction of these elements that is in the $UO_2$ matrix rather than in the rapidly released gap and grain boundary inventory requires that an additional goal for the fractional release for elements in the matrix be set at $2 \times 10^{-2}$. It is believed that this is achievable.

The final requirement is met by a combination of the allowed number of container failures per year, allowed number of cladding failures, limited quantity of Kr-85 available for rapid release from unoxidized fuel (less than 2 percent), and the first requirement to limit the amount of fuel that can oxidize.

From the foregoing, it can be seen that the container and the cladding are important both in limiting radionuclide releases and in preventing oxidation of the fuel during this period.

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<u>100- to 300-yr postclosure</u>. The environmental conditions in the period 100 to 300 yr after closure are expected to be characterized by borehole temperatures considerably lower than in the first 100 yr, though still well in excess of the boiling temperature of liquid water for most of the packages. As in the first 100 yr, this will preclude the possibility of liquid water contacting the majority of the waste packages.

By 100 yr after closure, the inventory of Kr-85 will have decayed to an insignificant level and the inventory of gap and grain boundary radionuclides will have decayed to 50 percent of the total activity. By 300 yr, these elements will contribute less than 5 percent to the total radioactivity. The percentage of the inventory accounted for by the actinides rises from approximately 50 percent at year 100 to more than 95 percent at year 300, with approximately 90 percent of the total activity due to isotopes of americium and plutonium alone.

As in the first 100 yr, fuel exposed to the repository air has the potential to oxidize and redistribute radionuclides to locations where they are readily accessible for rapid release upon contact with water. Though the oxidation would proceed more slowly because of the lower temperatures, fuel exposed to air during this period may oxidize on a time scale of tens to hundreds of years. Thus, the cladding and containers must continue to protect the majority of the fuel from oxidation to avoid increasing the fraction of gap and grain boundary elements available for rapid release.

Because of the rapidly changing radionuclide inventory during this period, the requirements of containment are different at the start of the period than at the end. The requirements driving the performance goals set for this period can be summarized as follows:

1. Limit the quantity of fuel that can oxidize to less than 2 percent of the total inventory in the failed containers.

2. Limit the annual release of gap and grain boundary elements early in this period.

3. Limit the annual release of actinides, specifically plutonium and americium.

The first requirement is met by the number of cladding failures allowed to occur while the fuel is dry and therefore hot enough to oxidize significantly (less than 2 percent).

The second requirement is met by the combination of the goals for allowed container failures, limited quantity of water (less than 5 percent of the packages being wet), number of total cladding failures (less than 5 percent), and the total number of packages that are initially contacted by liquid water in a single year (less than 1 percent). Note that overall, a goal of 5 percent failed cladding is set but only 2 percent of the cladding is allowed to fail while dry. This reflects the fact that once the waste has cooled sufficiently to allow liquid water to contact it, the cladding has fulfilled its primary function of preventing oxidation of the $UO_2$. Since the inventory of gap and grain boundary elements decays to a minor fraction of the total inventory during this period, the cladding is not as important in

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controlling the rapid release of these elements as it is in the first 100 yr after closure.

The release of the fraction of gap and grain boundary elements that are actually in the $UO_2$ matrix is controlled by the goal of a fractional release of $1 \times 10^{-3}$ in addition to the goals for container failure, cladding failure, and water availability. It is believed that this is achievable.

The third requirement, control of the release of actinides, is met by the combination of goals for allowed container failures, the limited quantity of water, and the low solubility of these elements in ground water of the expected composition.

300- to 1,000-yr postclosure. The environmental conditions during the years 300 through 1,000 after closure are expected to be characterized by slow cooling of the repository. A substantial fraction of the waste packages are expected to remain above the boiling point of water throughout this period. Though the expected conditions are that no liquid water will contact any of the waste packages, a goal is set that allows a limited quantity of water (no liquid water contacting the container for 90 percent of the packages, and less than 5 L per package per year for remaining 10 percent of the packages) to contact all the packages in the repository in a time-distributed manner.

The radionuclide inventory of the waste at 300 yr after closure is dominated by the actinides, which account for about 95 percent of the total radioactivity. At 1,000 yr after closure, the total inventory is one-half that at 300 yr. The contribution of americium and plutonium isotopes to the total rises from about 93 percent at 300 yr to about 97 percent at 1,000 yr. By 300 yr after closure, the makeup of the dominant radionuclides in the gap and grain boundary inventory has changed significantly from earlier times; the contribution of Tc-99 to the fraction available for rapid release is comparable to that of cesium and strontium at 300 yr, and becomes the dominant radioactivity in this fraction by 1,000 yr at which time it comprises about 750 parts in 100,000 of the total inventory.

Fuel temperatures are expected to drop to values at which oxidation of $UO_2$ proceeds quite slowly. Nevertheless, significant oxidation may occur in fuel that experiences cladding failure at temperatures well above the boiling point of water during this 700-yr period.

Unlike the other radionuclides present in significant quantity between 300 and 1,000 yr after closure, C-14 can be released as a $^{14}CO_2$ gas without requiring liquid water to contact the waste. The available data, however, indicate that less than 1 percent of the C-14 inventory is readily available for rapid release in this manner at elevated temperatures, and smaller releases are expected at lower temperatures.

The requirements driving the performance goal set for this period can be summarized as follows:

1. Limit the annual release of actinides, particularly americium and plutonium.

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2. Limit the annual release of gap and grain boundary elements (e.g., Tc).

3. Limit the annual release of C-14 as $^{14}CO_2$.

4. Limit the amount of oxidized fuel to less than 2 percent of the total amount of fuel.

The first requirement is met by a combination of the goal for container failure (less than 0.1 percent per year), the goal for the quantity of water that is allowed to contact a waste package (less than 5 L per year), and the low solubility of these elements in ground water of the expected pH and composition.

The second requirement is met by a combination of the goal for container failure, the goal for the fraction of these elements available for rapid release from unoxidized fuel, the number of packages initially contacted by liquid water, and the limit on the amount of oxidized fuel given in the fourth requirement. Release of the fraction of these elements not located in the pellet-cladding gap or on grain boundaries is controlled by the goal of less than 1 x $10^{-2}$ fractional release for the gap and grain boundary elements that are located in the $UO_2$ matrix and the above noted goals. It is believed that this achievable.

The third requirement is met by the combined goals for container failure and the quantity of C-14 that is available for rapid gaseous release.

The final requirement is met, as in previous time periods, by the goal for the fraction of cladding that is allowed to fail at high temperature (dry).

Performance parameter goals for the containment period. The following sections present more details on the performance allocated to each system element and the justification for the goals set for these allocations. In instances where the supporting information or activities for an allocation are drawn from another issue, that information is not repeated here. Instead, a brief summary of that material is given together with a reference to the appropriate section of the SCP.

Performance parameters goals for the engineered environment. As indicated in Table 8.3.5.9-1, performance measures and goals are set for both the quantity and quality of water than can contact a waste package during the containment period. Performance parameters and goals for water quality are given in Table 8.3.5.9-2.

The quantity of water that contacts a waste package will affect the degradation rates of both the container and the spent fuel cladding. In addition, with the exception of Kr-85 and C-14, significant release of radionuclides from a package requires the mediation of liquid water. The expected case under anticipated conditions is that no liquid water will contact the waste packages during the entire containment period and beyond. Nevertheless, the goal for the quality of water that can contact a waste package is set to be none for 95 percent of the packages and less than 5 L per package per year for the remaining 5 percent of the packages during the

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first 300 yr after closure. The corresponding goal for years 300 to 1,000 after closure is less than 5 L per package per year for 10 percent of the packages and no water for the remaining 90 percent. These goals parallel those set in Issue 1.10, waste package characteristics (postclosure) (see discussion of design envelope in Section 8.3.4.2), where the basis for selecting the goal and the parameters and models that will be used to demonstrate that the goal has been met are discussed in more detail.

In addition to goals for the total quantity of water that can contact the waste packages, goals have been set for the rate at which the ensemble of packages is initially contacted by liquid water. As stated in Table 8.3.5.9-1, the goal for this process is to allow no more than 1 percent of all the emplacement hole walls in the repository to fall below the boiling temperature of water in a single year, subject to the restrictions on the total number of packages allowed to be contacted by water at a given time. This goal is selected in order to spread out in time the potential release of the readily soluble, gap and grain boundary radionuclides. The assumption that release of radioactivity from a failed container via aqueous transport could occur as soon as the package falls below the boiling point is extremely conservative. This assumption does not consider the fact that though the borehole wall might be below the boiling point, the container and the waste within the container might not necessarily be below the boiling point. Thus, even though liquid water might exist at the borehole wall, it is not necessarily available to contact or enter the container. In addition, any water that enters the container might be vaporized and would then not be available for liquid transport of radioactivity. Further, once the temperature of the waste falls below the boiling point, it might take a considerable amount of time for water to accumulate within a container to the level of the breach. Alternatively, if the breach is in a position to allow immediate drainage, the water would have limited contact time with the waste.

The chemistry of the water that can contact either the container or the waste can have a large effect on the performance of these materials. For instance, as is discussed in a later section, the corrosion behavior of the austenitic alloys under consideration is sensitive to the chloride content of the water with which they come in contact. Thus, goals are set for the composition of the water contacting the waste packages so that the water will be similar to that currently thought to exist within the undisturbed environment in the unsaturated Topopah Spring tuff at Yucca Mountain. The detailed constraints are given in Table 8.3.5.9-2, and the characterization goals for water chemistry to be achieved during site characterization are given in Issue 1.10 (Section 8.3.4.2). Section 8.3.4.2 also provides the rationale for the selection of the goals listed in Table 8.3.5.9-2. The test and analyses to provide for the characterization of the water will be done under Characterization Program 8.3.1.3 (geochemistry) and Design Issue 1.10 (Section 8.3.4.2) and are not repeated here.

The method by which water is delivered to a waste package can affect both the corrosion rate and mechanism. Water that drips from a fracture onto a hot container surface might evaporate, leaving behind a residue of salts. These salts might accumulate and be dissolved in a later water flow, thereby creating small volumes of solutions with higher ionic strength than that given in Table 8.3.5.9-2 and discussed in the preceding paragraph. A characterization goal for the water flow mechanism has thus been set to determine

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whether dripping of water from fractures is likely under anticipated conditions. If it is likely, then the fraction of waste packages for which it will occur will be estimated. The activities dealing with this flow mechanism are described in Issue 1.10 (Section 8.3.4.2.4.3).

Performance parameter goals for the waste forms. As indicated in Table 8.3.5.9-1, a performance measure and goal has been set for the waste form during the containment period. The performance parameters and goals for this measure are given in Table 8.3.5.9-3.

The performance measure is based on the design objective of controlling the release of radionuclides from the ensemble of waste packages during the containment period to a small fraction of the radionuclide inventory present. The goal for the measure is expressed in terms of the allowed release from the ensemble of failed waste packages for the periods 0 to 300 yr and 300 to 1,000 yr after closure. The different numerical goals for these two periods reflect the increasing uncertainty in predicting the number of failed containers and the changing environmental conditions as a function of time.

Because of the difference in behavior of the glass and spent fuel waste forms, different performance parameters are assigned to them (Table 8.3.5.9-3). Because only one mechanism exists for release from the glass waste form (aqueous dissolution of the waste glass), only one parameter is given for the glass waste form: the fraction of the inventory of a glass-containing waste package in the effluent from such a package per year. A larger number of parameters are assigned to the spent fuel waste form because of the larger number of release modes possible for it. The different release modes possible for the spent fuel waste form arise from the fact that it is a heterogeneous material, consisting of several radionuclide-bearing components (cladding, assembly hardware, fuel, etc.). In addition, different radionuclides may be released from a single component by different mechanisms (e.g., gaseous release of Kr-85, rapid release of the gap and grain boundary inventory and release via dissolution of the $UO_2$ matrix). In contrast to this, the glass waste form is a relatively homogeneous material. It must be noted that the complexity of the description of the spent fuel waste form implies neither the superiority nor inferiority of the material in terms of the ultimate performance that will be demonstrated.

Glass waste form. The numerical goals for the glass waste form performance parameter (Table 8.3.5.9-3) are set so that the glass waste is not allowed to release more than its pro rata fraction of the repository inventory of radionuclides. The goals differ for the pre-300-yr and post-300-yr time periods because of the goal for a smaller number of wetted containers and a smaller number of failed containers in the first 300 yr after closure.

The glass performance parameter goal is very similar to the performance parameter goal set for the controlled release period under Issue 1.5 (Section 8.3.5.10). However, the small number of failed containers, because of the primary emphasis on design to attain total containment for the duration of the containment period, results in a net release rate substantially lower than the performance goal for the postcontainment period.

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The tests, analyses, models, and model inputs that will be used to show that the goals for the glass waste have been met are the same as those used to resolve Issue 1.5 and are not repeated here. Detailed discussion of these items may be found under Issue 1.5 (Section 8.3.5.10).

Spent fuel waste form. Performance parameters are defined and goals set for several components of the spent fuel waste form in Table 8.3.5.9-3. Specific performance is assigned to the cladding, the gaseous release behavior of C-14, the fraction of the inventory of gap and grain boundary radionuclides available for rapid release, the solubility of actinides, and the reaction rate of the $UO_2$ matrix. Each of these are discussed in the following paragraphs.

Performance measure for cladding. Performance is assigned to the cladding in order to limit the oxidation of the $UO_2$ at high temperatures throughout the containment period, and to control the release of gaseous Kr-85 during the first 100 yr of the containment period. Cladding is also assigned performance in order to limit the release of gap and grain boundary radionuclides in the first 300 yr after closure. Different parameter goals for the allowed fraction of failed cladding have been set for the three different time periods of the containment period. A goal of less than 2 percent failed cladding in failed containers is assigned to the first 100 yr after closure. A small fraction of the cladding will have failed during reactor service or during storage and handling before emplacement in the repository. It is expected that less than 0.5 percent of the cladding will fall into this category. The remaining fraction of failed cladding allows for the unavoidable uncertainty in the fraction of as-received failed cladding and allows for the occurrence of additional failures after emplacement. As previously discussed, the majority of the waste packages are expected to remain dry during the first 100 yr after closure. In the absence of liquid water, the only mechanism for causing cladding failure is that of stress rupture. If the cladding on a fuel rod is to fail by this mechanism it will most likely do so at early times, when the fuel temperatures are highest and therefore the internal pressure in the fuel rod is highest. The available data on this failure mechanism suggest that it will not be an important factor in limiting the life of cladding provided the design goals on peak cladding temperature are met (Section 7.2.1.3.3). Hydride reorientation in the cladding has the potential for reducing cladding strength and thus decreasing its ability to resist stress rupture; however, data on the extent to which this process occurs imply that it will not be a significant factor in causing cladding failure in the repository. The goal of less than 2 percent cladding failures in the first 100 yr is thus judged to be achievable.

Goals of less than 5 percent total failed cladding and less than 2 percent "dry" cladding failures (in failed containers) are set for the period 100 to 300 yr after closure. A distinction is made between "dry" and "wet" cladding for two reasons: (1) once the fuel has cooled below the boiling point of water, it will no longer oxidize at a rate sufficient to degrade its performance significantly, and the cladding has therefore fulfilled its primary purpose; and (2) once the cladding comes in contact with liquid water, additional mechanisms for failure, such as stress corrosion cracking, become possible. The limit on 2 percent dry cladding failures in failed containers limits the amount of oxidized fuel, a limit that is necessary to ensure the performance of the fuel at later times. Since few additional

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cladding failures by stress rupture are expected to occur after the temperature peak in the first 100 yr after closure (or during the preclosure period), the goal of 2 percent dry failures at 300 yr should be achievable. The limit of 5 percent on the total number of failed fuel rods at 300 yr was chosen to help limit the release of gap and grain boundary radionuclides during the first 300 yr after closure. The larger allowance given to the total number of failures versus dry failures reflects the possibility that some of the cladding may be contacted by liquid water during this period. Note that the packages whose temperatures fall below the boiling point and have the potential to become wet will be those that have a low radionuclide inventory, specifically the short-lived fission products Cs-137, Sr-90, and their daughter products. Since a primary concern during this time is to control the release of these radionuclides, the assumption that the cladding failures and the inventory in wet packages are randomly distributed is a conservative one.

After 300 yr postclosure, the performance requirement of less than 50 percent total failures is placed on the cladding; subject to the constraint that no more than 2 percent of the cladding is allowed to have failed when it is above the boiling point of water. This constraint, as discussed above, is imposed to ensure that no more than 1 percent of the fuel in the repository becomes significantly oxidized.

The tests, analyses, models, and model inputs that will be used to demonstrate that the goals on cladding failures have been met are discussed in detail under Issue 1.5 (Section 8.3.5.10), where they are grouped together with the other waste form characterization, testing, and modeling activities. That section also discusses the work planned for characterizing the oxidation rate of spent fuel and the effect of oxidation on the release of radionuclides. The reader is referred to that section for further information on these topics.

Performance measure for gap and grain boundary inventory rapid release. During reactor operation a fraction of certain volatile fission products that are not soluble in the the $UO_2$ matrix can migrate to the pellet-cladding gap or grain boundaries, where they are available for rapid release upon contact with water. Included in this group are the elements cesium, iodine, and to a lesser extent, strontium and technetium. As discussed in a preceding section, these nuclides dominate the radioactivity inventory of the fuel during the first 200 to 300 yr after closure. Control over their release is therefore a primary concern if the containment design objectives are to be met. In unoxidized fuel, the fraction of the inventory of these radionuclides that is in the gap and grain boundary ( as opposed to remaining within the $UO_2$ matrix) appears to be approximately equal to the fraction of fission gases released from the fuel (the fission gas release). It is expected that, on average, the fission gas release of the fuel emplaced in the repository will be less than 1 to 2 percent; hence, a performance parameter goal has been set that specifies that less than 2 percent of the inventory of these radionuclides will be available for rapid release in unoxidized fuel. This goal applies to both solid and gaseous radionuclides present in the pellet-cladding gap or on grain boundaries. For the purpose of setting goals, it has been assumed that once the fuel becomes oxidized, the entire inventory of fission gas and gap and grain boundary elements is available for rapid release. This is a conservative assumption because

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oxidation of the fuel is not a simple one-step process. During oxidation, $UO_2$ progresses through several intermediate phases ($U_4O_9$, $U_3O_7$, $U_3O_8$, then $UO_3$) and it is expected that much of the fuel will not become fully oxidized. It appears that gross redistribution of the fission products does not occur until the $U_3O_8$ stage of oxidation is reached; therefore, only in fuel that is oxidized to this stage will the entire inventory of gap and grain boundary elements be available for rapid release. Demonstration that the goal set for the rapid release fraction of the gap and grain boundary elements has been achieved will be done under Issue 1.5 (Section 8.3.5.10).

Performance measure for carbon-14 rapid release. Carbon-14 is present in the spent fuel waste form both in the fuel and on or near the exterior surfaces of the fuel cladding and assembly hardware. A fraction of C-14 can be released rapidly as $^{14}CO_2$ when air contacts the waste form at elevated temperatures. The presence of liquid water is not necessary for this release to occur. A goal has been set that would limit the release of C-14 in this way to less than 1 percent of the inventory of this radionuclide in spent fuel at elevated temperatures. The limited data in hand suggest that less than 0.3 percent of the C-14 is available for rapid release as $^{14}CO_2$. Release at lower temperatures is expected to be smaller. Demonstration that these goals have been met will be done under Issue 1.5 (Section 8.3.5.10).

Performance measure for actinide solubility. Goals have been set for the solubility of the elements plutonium, americium, and uranium. These elements constitute about 95 percent of the radioactivity in the spent fuel waste form after 300 yr postclosure. Goals were chosen to limit the release of these radionuclides based on very conservative values of solubility obtained from spent fuel dissolution experiments. The numeric values given in Table 8.3.5.9-3 are expressed in terms of a package inventory per liter of water and take into account both the goals on water quality and quantity and the number of container failures. Though the intermediate goal of limiting release of these elements remains the same, different numeric values are given for the time periods before and after 300 yr postclosure. The difference arises because of the different number of container failures and amount of liquid water available during these time periods. The concentrations of americium and plutonium will be limited to extremely low levels in ground water of the expected composition by the precipitation of phases containing these elements and current data indicate that this goal can be achieved. The tests, analyses, models, and model inputs used to demonstrate that the goal has been met will be conducted under Issue 1.5 and are not repeated here. The reader is referred to Section 8.3.5.10 for a detailed discussion of these topics.

Performance measure for the release of other radionuclides. The final performance parameter for the spent fuel waste form is the fractional release of all radionuclides that are not accounted for by the other parameters. This category includes the remaining fraction of gap and grain boundary elements that are present in the $UO_2$ matrix, other fission products, activation products that are present in the fuel, cladding and hardware, and the other actinides and their intermediate decay products. The solution concentration of some of these elements is expected to be limited by their solubility (e.g., zirconium, tin, nickel) but a significant portion of the inventory will not be so limited (e.g., technetium). As before, different

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numerical goals are assigned for the periods 0 to 300 yr and 300 to 1,000 yr after closure because of the changing goals on water availability and number of container failures, as well as the changing composition of the radionuclide inventory.

The release of other radionuclides from the fuel itself will be governed by the reaction rate of the $UO_2$ matrix and the availability of water. For those nuclides that are not limited by their solubility to solution concentrations corresponding to fractional releases lower than the goals set for this parameter, it will be demonstrated that their release is controlled to the specified limits by a combination of the reaction rate of the $UO_2$ and the limits on the availability of water.

Release of radionuclides other than C-14 from nonfuel components (cladding and assembly hardware) will be governed by the generalized corrosion rate of the materials involved. As in the instance of radionuclides released from the fuel itself, many of the elements released from these sources will be limited by their solubility. Those that are not will be shown to have release fractions lower than that specified by the parameter goals.

The tests, analyses, models, and model inputs used to demonstrate that the goal for the release of other radionuclides from the spent fuel waste form has been met will be conducted under Issue 1.5. The reader is referred to Section 8.3.5.10 for a detailed discussion of these topics.

Performance parameter goals for the container. The performance measure allocated to the container in Table 8.3.5.9-1 is the fraction of containers that have failed. The performance goal is divided into two time intervals as follows:

1. For the first 300 yr after repository closure, less than 0.05 percent per year of the total population of emplaced containers will fail. A failed container is defined as one with a defect sufficiently large to sustain an air flow of $1 \times 10^{-4}$ atm-$cm^3$/s. This test is a general standard accepted by the nuclear industry. (This flow rate is the same numerical value as the ASME leak tightness test described in ASME Section V, Article 10, Appendix IV, 1986 Edition.)

2. For the interval from 300 to 1,000 yr after repository closure, less than 0.1 percent per year of the total population of emplaced containers will fail. The same definition of a failed container applies in this time period.

This performance measure must be further divided to assign meaningful performance parameters and goals for those parameters. The division is along two lines: container material type and degradation modes. The container material has not yet been selected. Materials from two separate alloy families are under consideration. This is reflected in Figure 8.3.5.9-3 (model hierarchy) by the division of the container degradation model into the copper-based and austenitic alloy families. An alternate materials/concepts effort is being pursued concurrently. The reader is referred to Section 8.3.5.9.1 for a detailed discussion of these topics. A performance

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measure has not been assigned to the alternative materials. The performance parameters are divided into the two alloy families because the degradation behavior is substantially different between families and substantially similar within families. The performance measure is also divided into "submeasures" by degradation mode because different modes have different controlling parameters.

Table 8.3.5.9-4 lists the detailed degradation mode submeasures. One or more performance parameters are identified for each mode. In each case, these performance parameters were selected because they were regarded as the key measures for predicting the container degradation. Some failure modes have more than one parameter identified because a combination of these may be employed to establish the performance. Performance parameter goals are also listed. Table 8.3.5.9-5 lists the model inputs for each degradation mode. A brief explanation of the performance parameters follows. More detailed discussion of the performance parameters and explanation of the models is deferred to Information Needs 1.4.1, 1.4.2, and 1.4.3. (Sections 8.3.5.9.1 through 8.3.5.9.3).

In Table 8.3.5.9-4, a tentative goal is established for each performance parameter; this goal is established on the basis of what chemical, metallurgical, physical, or mechanical features of the container or the environment appear to be the key features in determining performance. The current estimated value, or range of values, is based in some instances on measurements that have been performed in the Yucca Mountain Project-sponsored work and discussed in Section 7.4.2. In other instances, the current estimated value is based on information from the technical literature. In all instances, the values indicated as performance parameter goals are estimates of points where discernible differences in performance of the metal container occur. Much of the work outlined under Information Needs 1.4.1 through 1.4.3 is concerned with establishing "critical" values of environmental, metallurgical, and mechanical parameters where the degradation behavior of the metal barrier will change significantly and to relate these critical values to the range of conditions that will occur in the Yucca Mountain repository. As discussed under Information Need 1.4.2, six candidate materials in two major alloy families are currently being evaluated for the container. A material selection process is outlined in Information Need 1.4.2. Part of the input to the selection process is determining which degradation modes are the most important and how the resistance of each candidate material to these degradation modes should be weighted in the selection process. A further consideration is the ability to model the various degradation modes. How accurately the values of these critical environmental, metallurgical, or mechanical parameters can be determined may limit the utility of the models.

Many of the degradation modes have a time factor associated with them, because certain conditions must exist before the particular degradation mode can occur. For example, the aqueous corrosion degradation modes require the presence of an electrolyte on the metal surface. The return of the unconfined water boiling point isotherm will occur over a span of time so that there will be a distribution of the initiation of aqueous corrosion modes. Similarly, the performance parameter goals that are related to microstructural features in the metal (e.g., formation of brittle phases, degree of sensitization) are most often dependent on time-at-temperature to form the microstructural feature.

Table 8.3.5.9-4. Performance parameters and goals for containers subdivided by alloy family and degradation mode[a] (page 1 of 4)

| Performance measure | Degradation modes | Performance parameter | Tentative goals[b] | Needed confidence | Current estimated range | Current confidence |
|---|---|---|---|---|---|---|
| | | COPPER BASED ALLOYS | | | | |
| Fraction of containers that have failed | Metallurgical and mechanical effects | Brittle phase fraction | Phase fraction <0.01 | High | Phase fraction <0.01 | Medium |
| | | Reduction in fracture toughness | J(emb)/J <0.7 | High | To be determined | NA[c] |
| | Low temperature oxidation | Oxidation rate (R) | Average rate ≤0.1 d per 1,000 yr | High | R ≈ 0.03 to 3 μm/yr | Medium |
| | General aqueous corrosion | General corrosion rate (R) | Average rate ≤0.1 d per 1,000 yr | High | R ≈ 0.4 to 5 μm/yr | Medium |
| | Hydrogen effects | H content | [H] <0.1 [H(crit)] | Medium | To be determined | NA |
| | | Oxide inclusion phase fraction | Phase fraction <0.01 | High | Phase fraction <0.01 | Medium |
| | Localized attack | Critical potential for initiation | E(crit) - E(corr) >100 mV | High | E(crit) - E(corr) ≈ (100 to 800) mV | Low |



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Table 8.3.5.9-4. Performance parameters and goals for containers subdivided by alloy family and degradation mode[a] (page 2 of 4)

| Performance measure | Degradation modes | Performance parameter | Tentative goals[b] | Needed confidence | Current estimated range | Current confidence |
|---|---|---|---|---|---|---|
| | Stress corrosion cracking (SCC) | Critical potential | E(critSCC) - E(corr) > 100 mV | High | To be determined | Low |
| | | Ammonia ($NH_3$) concentration | $[NH_3]$ <2 ppm | High | $[NH_3]$[d] < det. ~ 2 ppm | Medium |
| | | Stress intensity (K) | K < K(SCC) | Medium | K ~ (0.1 to 3) K(SCC) | Low |
| | Other effects | To be determined | To be determined | To be determined | NA | NA |
| | | | AUSTENITIC ALLOYS | | | |
| | Metallurgical and mechanical effects | Brittle phase fraction | Phase fraction <0.01 | High | Fraction ~ 0 to 0.03 | Medium |
| | | Reduction in fracture toughness | J(emb)/J <0.7 | High | J(emb)/J ~ 0.5 to 1.0 | Medium |
| | Low temperature oxidation | Oxidation rate (R) | Average rate ≤0.1 d per 1,000 yr | High | R ~ 0.02 to 0.1 μm/yr | High |



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Table 8.3.5.9-4. Performance parameters and goals for containers subdivided by alloy family and degradation mode[a] (page 3 of 4)

| Performance measure | Degradation modes | Performance parameter | Tentative goals[b] | Needed confidence | Current estimated range | Current confidence |
|---|---|---|---|---|---|---|
| | | AUSTENITIC ALLOYS (continued) | | | | |
| | General aqueous corrosion | General corrosion rate (R) | Average rate ≤0.1 d per 1,000 yr | High | R ~ 0.04 to 0.3 μm/yr | Medium |
| | Intergranular attack and intergranular stress corrosion cracking (IGSCC) | Degree of sensitization, R(A) (activation ratio in EPR[e] test) | R(A) <5% | High | R(A) ~ 0 to 20% | Medium |
| | | Stress intensity, K | K < K(IGSCC) | Medium | K ~ (0.1 to 3) K(IGSCC) | Low |
| | Hydrogen effects | H content | [H] <0.1 [H(crit)] | Medium | To be determined | NA |
| | | Martensite fraction, M | M <0.01 by volume | High | M <0.01 by volume | Medium |
| | Localized attack | Critical potential | E(crit) - E(corr) >100 mV | High | E(crit) - E(corr) ~ (0 to 900) mV | Low |
| | | Chloride ion content | [Cl⁻] <100 ppm | High | [CL⁻] ~ 5 to 150 ppm | Medium |





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Table 8.3.5.9-4. Performance parameters and goals for containers subdivided by alloy family and degradation mode[a] (page 4 of 4)

| Performance measure | Degradation modes | Performance parameter | Tentative goals[b] | Needed confidence | Current estimated range | Current confidence |
|---|---|---|---|---|---|---|
| | Transgranular stress corrosion cracking (TGSCC) | Critical potential | E(critTGSCC) - E(corr) > 100 mV | High | To be determined | NA |
| | | Chloride ion content | $[Cl^-]$ <50 ppm | High | $[Cl^-]$ ≈ 5 to 150 ppm | Medium |
| | | Stress intensity | K < K(TGSCC) | Medium | K ≈ (0.1 to 3) K(TGSCC) | Low |
| | Other effects | To be determined | To be determined | To be determined | NA | NA |

[a]See text discussion for explanations of degradation modes. Section 8.3.5.9.3 contains additional material explaining some of the interactions between the chemical, physical, metallurgical, or mechanical properties.

[b]Parameters not defined in table are as follows: J(emb) = impact strength of the embrittled material; J = normal impact strength; d = container wall thickness, 1 cm < d < 3 cm; H(crit) = critical hydrogen; E(crit) = critical potential; E(corr) = corrosion potential; K = stress intensity factor; K(SCC) = critical value of K at which stress corrosion cracking takes place; K(IGSCC) = critical value of K at which intergranular stress corrosion cracking takes place; E(critTGSCC) = critical potential with respect to transgranular stress corrosion; K(TGSCC) = critical value of K at which transgranular stress corrosion cracking takes place.

[c]NA = not applicable.

[d]det. = detection limit.

[e]EPR = electrochemical potentiokinetic reactivation.



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Table 8.3.5.9-5. Container degradation model inputs

| Model | Model inputs | Needed confidence | SCP section |
|---|---|---|---|
| | COPPER-BASED ALLOY FAILURE MODELS | | |
| Metallurgical aging and phase stability | Temperature-time projections | High | 8.3.5.9.3.1.1 |
| | Quantity of phase segregation | High | 8.3.5.9.3.1.1 |
| | Mechanical properties of the segregation products | Medium | 8.3.5.9.3.1.1 |
| | Electrochemical differences between segregation products and base metal | Medium | 8.3.5.9.3.1.1 |
| | Strain in the container body material and in the heat affected zone around the closure | Medium | 8.3.5.9.3.1.1 |
| | Residual stress | Medium | 8.3.5.9.3.1.1 |
| Low temperature oxidation | Oxidation rate | High | 8.3.5.9.3.1.2 |
| | Temperature | High | 8.3.5.9.3.1.2 |
| | Radiation field intensity | High | 8.3.5.9.3.1.2 |
| | Identification and quantity of radiolysis products | Medium | 8.3.5.9.3.1.2 |
| General aqueous | General corrosion rate | High | 8.3.5.9.3.1.3 |
| | Composition of water | High | 8.3.5.9.3.1.3 |
| | Composition of corrosion product layers | Medium | 8.3.5.9.3.1.3 |
| | Identification and quantity of radiolysis products | Medium | 8.3.5.9.3.1.3 |
| Hydrogen entry and embrittlement | Hydrogen production rate by radiolysis and corrosion | Medium | 8.3.5.9.3.1.4 |
| | Hydrogen recombination rate | Medium | 8.3.5.9.3.1.4 |
| | Rate of hydrogen entry into the alloy | Medium | 8.3.5.9.3.1.4 |
| | Concentration of hydrogen in the alloy | High | 8.3.5.9.3.1.4 |
| | Phase structure of the alloy | High | 8.3.5.9.3.1.4 |
| | Mechanical property changes from hydrogen degradation | High | 8.3.5.9.3.1.4 |
| Pitting, crevice, and other localized attacks | Critical concentration of ions known to favor these modes of attack | High | 8.3.5.9.3.1.5 |
| | Temperature | High | 8.3.5.9.3.1.5 |
| | Solution pH | High | 8.3.5.9.3.1.5 |

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Table 8.3.5.9-5. Container degradation model inputs (continued)

| Model | Model inputs | Needed confidence | SCP section |
|---|---|---|---|
| Pitting, crevice, and other localized attacks (continued) | Metal microstructure | High | 8.3.5.9.3.1.5 |
| | Corrosion potential | High | 8.3.5.9.3.1.5 |
| | Pitting (and other critical potentials) | High | 8.3.5.9.3.1.5 |
| Stress corrosion cracking | Concentration of ammonia (and other species) known to favor stress corrosion cracking | High | 8.3.5.9.3.1.6 |
| | Temperature | High | 8.3.5.9.3.1.6 |
| | Stress (and stress cracking) | Medium | 8.3.5.9.3.1.6 |
| | Alloy segrations | Medium | 8.3.5.9.3.1.6 |
| | Corrosion potential | High | 8.3.5.9.3.1.6 |
| | Critical potential for crack initiation | High | 8.3.5.9.3.1.6 |
| Other potential degradation modes | To be determined | Not applicable | 8.3.5.9.3.1.7 |
| | AUSTENITIC ALLOY FAILURE MODELS | | |
| Metallurgical aging and phase transformation | Temperature-time projections | High | 8.3.5.9.3.2.1 |
| | Kinetics of phase transformation reactions | High | 8.3.5.9.3.2.1 |
| | Mechanical properties of the transformation products | Medium | 8.3.5.9.3.2.1 |
| | Alloy composition of the base metal and the weld metal | High | 8.3.5.9.3.2.1 |
| | Strain in the container body material and in the heat affected zone around the closure | Medium | 8.3.5.9.3.2.1 |
| | Residual stress | Medium | 8.3.5.9.3.2.1 |
| Low temperature oxidation | Oxidation rate loss or gain tests under relevant conditions | High | 8.3.5.9.3.2.2 |
| | Temperature | High | 8.3.5.9.3.2.2 |
| | Radiation field intensity | High | 8.3.5.9.3.2.2 |
| | Identification and quantity of radiolysis products | Medium | 8.3.5.9.3.2.2 |

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Table 8.3.5.9-5. Container degradation model inputs (continued)

| Model | Model inputs | Needed confidence | SCP section |
|---|---|---|---|
| General aqueous corrosion | General corrosion rate | High | 8.3.5.9.3.2.3 |
| | Composition of water | High | 8.3.5.9.3.2.3 |
| | Composition of corrosion product layers | Medium | 8.3.5.9.3.2.3 |
| | Identification and quantity of radiolysis products | Medium | 8.3.5.9.3.2.3 |
| Intergranular attack and intergranular stress corrosion cracking | Temperature-time projections | High | 8.3.5.9.3.2.4 |
| | Diffusion rate of chromium in the metal as a function of temperature | High | 8.3.5.9.3.2.4 |
| | Diffusion mechanism for chromium in the metal | Medium | 8.3.5.9.3.2.4 |
| | Strain | Medium | 8.3.5.9.3.2.4 |
| | Alloy composition | High | 8.3.5.9.3.2.4 |
| | Effects of transformation products on diffusion rates | Medium | 8.3.5.9.3.2.4 |
| | Composition of carbide precipitates formed | Medium | 8.3.5.9.3.2.4 |
| | Amounts of sigma and chi phases | Medium | 8.3.5.9.3.2.4 |
| Hydrogen entry and embrittlement | Hydrogen production rate by radiolysis and corrosion | Medium | 8.3.5.9.3.2.5 |
| | Hydrogen recombination rate | Medium | 8.3.5.9.3.2.5 |
| | Rate of hydrogen entry into the alloy | Medium | 8.3.5.9.3.2.5 |
| | Concentration of hydrogen in the alloy | High | 8.3.5.9.3.2.5 |
| | Phase structure of the alloy | High | 8.3.5.9.3.2.5 |
| | Mechanical property changes from hydrogen degradation | High | 8.3.5.9.3.2.5 |
| Pitting, crevice, and other localized attack | Critical concentration of ions known to favor these modes of attack | High | 8.3.5.9.3.2.6 |
| | Temperature | High | 8.3.5.9.3.2.6 |
| | Solution pH | High | 8.3.5.9.3.2.6 |
| | Metal microstructure | Medium | 8.3.5.9.3.2.6 |
| | Corrosion potential | High | 8.3.5.9.3.2.6 |
| | Pitting potential | High | 8.3.5.9.3.2.6 |

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Table 8.3.5.9-5. Container degradation model inputs (continued)

| Model | Model inputs | Needed confidence | SCP section |
|---|---|---|---|
| Transgranular stress corrosion cracking | Chloride concentrations of water | High | 8.3.5.9.3.2.7 |
| | Temperature | High | 8.3.5.9.3.2.7 |
| | Stress | Medium | 8.3.5.9.3.2.7 |
| | Alloy constituents | Medium | 8.3.5.9.3.2.7 |
| | Other ions in solutions | Medium | 8.3.5.9.3.2.7 |
| | Corrosion potential | High | 8.3.5.9.3.2.7 |
| Other potential degradation modes | To be determined | Not applicable | 8.3.5.9.3.2.8 |

Some abbreviated notations are used in Table 8.3.5.9-4 for simplicity in the entries. These are briefly explained below, along with some remarks on their interpretation. The reader should refer to the full discussion on models of degradation modes (Information Need 1.4.3) for additional material that explains some of the interactions between the chemical, physical, metallurgical, or mechanical factors. The following discussion of the performance parameters is arranged by failure mode. In some cases, the discussion applies to both alloy families; in other cases, the remark is specific to only one family (and sometimes just to one metal or alloy in that family).

Metallurgical and mechanical effects. Under these effects in Table 8.3.5.9-4, for both the copper-based and austenitic materials, reduction in fracture toughness is indicated by the ratio J(emb)/J, where J is the normal impact strength and J(emb) is the impact strength of the embrittled material. Other indices of degraded mechanical properties affecting ductility or toughness may also be applied. For the austenitic materials, formation of sigma phase was used as the standard for establishing a critical value for loss in fracture toughness. A similar value is specified for copper-based materials. The likely embrittling species in copper and copper-based alloys are residual impurities, such as arsenic, selenium, and lead, that precipitate at grain boundaries. In some cases, oxide or other inclusions could be the source of the embrittlement. These effects may be addressed by determining an appropriate specification on these residuals and inspection of the container material. Similarly for the austenitic materials, sigma (and other brittle) phase formation during fabrication can be detected as part of the container acceptance criteria. However, the concern here is formation of sigma phase if the appropriate metallurgical, strain, and time-at-temperature conditions are present after emplacement.

Oxidation and general aqueous corrosion. The entries in Table 8.3.5.9-4 on oxidation and general aqueous corrosion for both alloy systems express the time-average rate over a 1,000-yr period being such that

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90 percent or more of the initial container thickness remains at all times up to 1,000 yr. The container thickness is considered a variable with the range of approximately one to three centimeters. For the purpose of this issue and information need, this range of container thicknesses allows (1) some options in the waste package design, (2) some options in processes for fabricating containers, (3) lower strength materials (such as high purity copper) to be accommodated, and (4) a somewhat higher oxidation-general corrosion rate for copper under some environmental conditions.

Hydrogen effects. In Table 8.3.5.9-4, these effects are indicated by the performance parameters relating to the amount of hydrogen absorbed by the metal. In most cases, hydrogen is preferentially absorbed and "trapped" by or associated with a particular microstructural constituent. In the instance of the metastable austenitic stainless steels, austenite transformation to martensite is the key to inducing a condition that may lead to hydrogen embrittlement. In the case of high purity copper (CDA 102), oxygen pickup during welding or hot forming may form copper oxide inclusions that are unstable in a hydrogen-containing environment, resulting in blistering of the copper. The parameter goal is set such that the hydrogen content in the metal should be less than 0.1 of the "critical" hydrogen content, but in both alloy families that amount is not yet determined.

Intergranular attack and intergranular stress corrosion cracking. For these effects on austenitic materials, the performance parameter in Table 8.3.5.9-4 is the degree of sensitization. (There is no corresponding sensitization phenomena in copper-based materials). There are a number of ways to define "degree of sensitization", but the one chosen here relates to the activation ratio as determined in an electrochemical potentiokinetic reactivation (EPR) test. The activation ratio relates to the electrochemical current required to "activate" a previously passivated specimen. The activation ratio is proportional to the degree of sensitization; the EPR test is particularly useful in discerning degrees of sensitization in low-carbon austenitic materials where only a fraction of the grain boundaries are attacked electrochemically. For intergranular stress corrosion cracking of the austenitic materials, the stress intensity factor (K) can also be used as an additional performance parameter. However, there are some difficulties associated with using K for this (and all the other stress corrosion cracking) modes(s). First, all of the candidate materials are normally very ductile materials so that there is the question of how to incorporate the plasticity contribution. Second, the parameter goal that K be below the K(SCC) (the critical value of K at which stress corrosion cracking takes place) is sometimes difficult to show experimentally because stresses vary widely over small distances between the weld and the base metal and because of the uncertainty in detecting the magnitude and distribution of all flaws. Hence, the current estimated range of K is quite large.

Localized attack. For localized attack in both alloy systems, one performance parameter is expressed in Table 8.3.5.9-4 as the difference between the "critical" potential, E(crit)and the corrosion potential, E(corr). The "critical" potential varies according to several physical, chemical, and metallurgical quantities (as does the corrosion potential). Also, different critical potentials exist according to the particular localized corrosion phenomena being studied. In the instance of copper-based materials, pitting corrosion and selective leaching (for the alloys only)

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need to be considered. The 100 mV difference has been set as the parameter goal; this value is based, in part, on previous work discussed in Chapter 7 and in other Yucca Mountain Project supported work, and, in part, what seems to be a reasonably conservative value from the literature in comparable environmental settings. It should be noted that the critical potential will include effects of microstructural features (inclusions, second phases, etc.). No particular chemical species are expected to be present naturally in the ground waters associated with the Yucca Mountain repository that are especially important in causing localized attack on copper-based materials. For the austenitic materials, pitting and crevice attack need to be considered as types of localized corrosion (selective leaching is not known in these materials). However, in the instance of the austenitic materials, chloride (and to a lesser extent, fluoride) ion is present in the natural environment and is of paramount concern in setting one of the parameter goals. Therefore a chloride ion content is set in the parameter goal for failure of the container by localized corrosion (pitting or crevice attack on these materials) and the value set is 100 ppm. This goal is tentative and must be viewed in the light of some controversy because the chloride ion threshold for the initiation of localized attack will depend on other chemical species present in the environment. The work proposed in Information Need 1.4.2 has the purpose of establishing the value of critical concentrations of causative ions for localized attack on the selected container material.

Transgranular stress corrosion cracking. For this effect in the austenitic materials, a difference between the critical and corrosion potential is set as the performance measure (Table 8.3.5.9-4). This approach has been demonstrated in concentrated chloride solutions, but has not yet been shown to be valid in dilute chlorides, hence the "to be determined" entry for the current estimated value. The 100 mV difference has been taken as the parameter goal from analogy to the localized corrosion failure mode. As in localized corrosion, chloride ion is the outstanding example of the causative species for initiating transgranular stress corrosion cracking (TGSCC), and a critical parameter goal is set at 50 ppm chloride. This parameter must also be considered as tentative for the same reasons given above, and the actual threshold will depend on several other factors (pH, temperature, other ions present). This threshold is established with the most susceptible candidate material in mind (AISI 304L). This threshold will be less controversial for the more resistant candidate material (alloy 825) in this alloy family. As in the discussion of the intergranular stress corrosion cracking (IGSCC) for the austenitic materials, a stress intensity factor (K) is set as the third performance parameter. The comments made in that discussion also apply here.

Stress corrosion cracking. For copper-based materials, ammonia is the outstanding causative species for stress corrosion cracking (Table 8.3.5.9-4). (The crack propagation path is not distinguished as a fundamental characteristic of the degradation mode as it is for the austenitic alloys). Therefore, an ammonia concentration limit is set as one of the performance parameters. Because ammonia is not present in the natural environment (it would form because of radiolysis) and small amounts of ammonia cause stress corrosion cracking (SCC) in the most vulnerable candidate alloys in this family (CDA 102 and CDA 613), a very low threshold is set as the parameter goal. This level is believed to be the detection limit (2 ppm).

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Models and model inputs

Models, model inputs, their needed confidence, and forward references to the information needs are given in Table 8.3.5.9-5 for the different failure modes in each alloy system. The model inputs are quantities that are measurable and quantifiable. The needed confidence is determined by considering how measurable the quantities are and how important they are in establishing the model.

Interrelationships of information needs

Information Needs 1.4.1 and 1.4.2 (Sections 8.3.5.9.1 and 8.3.5.9.2) will provide data to be used in the analysis of container performance under repository conditions. The conditions to be considered will include low probability scenarios (not required for resolution of this issue but needed for input to Issue 1.1, Section 8.3.5.13), as well as anticipated processes and events. Models will be developed under Information Need 1.4.3 (Section 8.3.5.9.3) to allow extrapolation of the laboratory data to long times. The models and data will be combined in analyses to be done under Information Need 1.4.4 (Section 8.3.5.9.4) to provide a description of the condition of the container under anticipated processes and events for 10,000 yr (10 CFR 60.112 and 40 CFR 191.13), for low probability cases for 10,000 yr, and for expected conditions for 100,000 yr (10 CFR 960.3-1-5).

The issue will be resolved under Information Need 1.4.5 (Section 8.3.5.9.5), where the analyses from Information Needs 1.4.4 (Section 8.3.5.9.4) and 1.5.4 (Section 8.3.5.10.4) will be compared with the interpretation of substantially complete containment.

## 8.3.5.9.1 Information Need 1.4.1: Waste package design features that affect the performance of the container

Technical basis for addressing the information need

This information need addresses the important features of the waste package design that affect the performance of the container. Under this information need, the as-fabricated and as-assembled waste package is first characterized with respect to ensuring the integrity of the as-emplaced container. A close relationship also exists between certain design parameters, the manufacturing processes by which the container is fabricated and closed, and the ultimate performance of the container in the preclosure and postclosure repository environment. Some decisions on design details will, therefore, depend on which metal of the several candidate materials is selected and how the waste package is fabricated, assembled, and emplaced in the repository. Characterization of the properties of the as-emplaced container is an important part of resolving this issue because many of these properties influence the behavior of the container during the containment and postcontainment periods. This information need addresses those features of both the reference and the alternative design that characterize the as-

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emplaced package and that influence the behavior in later periods. The reference design is a metal container, and the alternative design is to be chosen from one of the following: ceramic-metal systems, bimetallic/single metal systems, and coatings and filler systems.

Link to the technical data chapters and applicable support documents

The characteristics of the waste package are discussed in Chapter 7. Characterization and description of the waste form contents of the package are given in Section 7.4.3. The six candidate waste package container materials are introduced and discussed in Section 7.4.2 on the metal barriers. Representative mechanical properties and the metallurgical industry standard composition ranges are given in Section 7.3 for each candidate material. The waste package design and a brief discussion on fabrication and welding (or other closure) processes for producing the waste package are given in Section 7.3. Some aspects of the repository description and layout design influence this information need; these are found in Chapter 6. No previous work has been performed by the Yucca Mountain Project on other materials systems as container materials; thus, no discussion of other materials systems is presented in Chapter 7.

Parameters

Information needed from other information needs includes

1. The reference and alternative waste package designs, from Information Need 1.10.2 (Section 8.3.4.2.2).

2. The temperature at the container surface and projections of the change in temperature with time. This comes from Information Need 1.10.4 (Section 8.3.4.2.4) and is based on the thermal power load per container (Information Need 1.5.1, Section 8.3.5.10.1) and the areas power load (Information Need 1.10.3, Section 8.3.4.2.3).

3. The radiation field intensity in the near-package environment and projections of its change with time, from Information Need 1.10.2 (Section 8.3.4.2.2).

4. Emplacement configuration (horizontal or vertical) in the repository, from Information Need 1.10.3.

5. Thickness of the metal container, from Information Need 1.10.2.

6. Design configuration of the container-system developed under the alternate barriers investigations (Information Need 1.10.2).

7. The process history of the container body and other assembly components (e.g., bottom and top lids, weld filler metal) used in the assembled and closed waste package container, from Information Need 1.10.2.

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Data for the following parameters are to be obtained:

1. The candidate container materials. The six candidate materials are classified into two broad alloy groups: (a) copper and copper-based alloys and (b) austenitic materials (iron- and nickel-based alloys). The specific candidates in the first group are oxygen-free high-conductivity copper CDA 102 (UNS C10200), aluminum bronze CDA 613 (UNS C61300), and 70/30 copper-nickel CDA 715 (UNS C71500). The candidates in the second groups are austenitic stainless steel AISI type 304L (UNS S30403), austenitic stainless steel AISI type 316L (UNS S31603), and nickel-based austenitic alloy 825 (UNS N08825).

2. Candidate container material systems being evaluated under the alternate barriers investigations include ceramic-metal systems, bimetallic/single metal systems, and coating and filler systems.

3. The mechanical properties of the container material in the as-emplaced condition, from which the relative projected changes of these are established for the repository preclosure, the containment, and the postcontainment periods.

4. The microstructural characteristics of the container material in the as-emplaced condition. Projections of any changes in the microstructure of the container after emplacement (Information Needs 1.4.2 and 1.4.3) are based on characterization of the as-emplaced microstructural condition.

5. Certain physical properties of the container material that are relevant to the waste package design analysis.

6. The state of stress (nature, magnitude, and distribution) that exists in the container at the time of emplacement and projections of the changes in the state of stress after emplacement.

7. The integrity of the assembled and closed waste package container as it is emplaced into the repository. The integrity of the closure weld or other closure process is of special importance.

8. The surface condition of the assembled, closed, and emplaced waste package container.

9. For the option of a container system being developed under the alternate barriers investigations, a similar set of properties and fabrication characteristics.

10. The effect of large-scale fabrication on the metallurgical condition and resultant performance.

Logic

The container temperature, radiation field, and state of stress and the expected range and variation of these parameters during the containment and postcontainment periods are used in establishing the test conditions that are part of the study areas more fully discussed under Information Needs 1.4.2

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and 1.4.3. The emphasis of the present information need is characterization of the condition of the container as it is emplaced in the repository. In some instances, it will be appropriate to use standardized test methods and procedures governed and issued by the American Society for Testing and Materials (ASTM). The next two Information Needs (1.4.2 and 1.4.3) use this information for predicting the characteristics of the container in the later periods. In some instances, certain design features bear on the selection of the container material. Much of the information on mechanical and physical properties of the candidate container materials is available from published sources. These properties are not environmentally dependent and so are not site-specific to Yucca Mountain; therefore, compilation of existing information should suffice.

For the option included in the alternate barriers investigations, i.e., ceramic-metal systems, bimetallic/single metal systems, and coatings and filler systems, the important physical and mechanical properties are closely linked with the proposed feasibility study on this design option. These properties will be discussed in Activity 1.4.1.2 (Section 8.3.5.9.1.2).

Two activities are included in this Information Need (1.4.1). The first activity is based on using a container fabricated from one of the six candidate metallic materials. This includes both copper-based and austenitic alloys. Some activities are common to both alloy groups, and some are specific to (or are more emphasized in) one group, as indicated in the following descriptions.

The second activity is based on a waste package concept that will evolve from the alternate barriers investigations and be chosen from one of the three following groups: ceramic-metal systems, bimetallic/single metal systems, and coating and filler systems. This activity will introduce some unique features not found in the first activity and is less well-defined because the feasibility of producing such a waste package must first be evaluated.

Activities to be pursued to completion depend on which waste package concept, together with the appropriate materials, is eventually selected and the outcome of the feasibility study on the alternate barriers investigations. The activities that support the selection process for the metallic container materials are explained in the next Information Need (1.4.2).

The presently available information on the items discussed in this information need is probably adequate to serve the needs of material selection, except possibly in the area of welding effects. The Project intends to evaluate the existing information during the time up to material selection and to undertake only those laboratory measurements needed to support material selection. After the final container material is chosen, a test plan will be developed for the selected material to supply the data needed to support the repository license application.

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8.3.5.9.1.1 Activity 1.4.1.1: Integrate design and materials information (metal container)

The following subactivities support this activity.

8.3.5.9.1.1.1 Subactivity 1.4.1.1.1: Mechanical properties

Objectives

The objective of this subactivity is to compile available data on the mechanical properties of the candidate materials over the temperature range of interest (approximately room temperature to 300°C).

Parameters

The principal mechanical properties of interest are the following:

1. Yield strength.

2. Ultimate tensile strength.

3. Elongation (or other measure of ductility, such as reduction in area).

4. Modulus of elasticity.

5. Impact strength (or other measure of fracture toughness).

Knowledge of the effect of metal fabrication processing and interrelationships between mechanical properties and microstructural properties is also required. This includes the effect of such factors as phase distribution, grain size, inclusion content, and previous plastic deformation. The effect of the strain rate on the mechanical properties is also needed. While individual mechanical properties were just listed, the entire stress-strain relationship merits attention to enable the evaluation of the toughness of the material when subjected either to low strain rate or to high strain rate processes that can later develop in the containment period.

Description

Depending on the results of the compilation, experimental determination of any inadequately known mechanical properties will be performed. Extended time at temperature may change the values of the mechanical properties, and this effect will be considered in the compilation. For the austenitic materials (including alloy 825), there will be little need for additional experimental work for this activity because of the extensive published information on this subject. However, some experimental work may be required to determine the properties of the welded-austenitic material because of the inherently more complex structure of the weld and its dependence on many process variables that will be determined in the future. Because the data on mechanical properties of the copper-based materials at the higher end of the repository-relevant temperature range is not as extensive as that for the

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austenitic materials, some experimental work may be needed to fill the information gaps.

The low strength of high-purity copper (CDA 102) suggests that a long-term, low-temperature creep phenomenon may lead to a degradation mode that would be most important for the "retrieval period" following emplacement of the container in the repository. The somewhat thicker container sections (approximately 2-3 cm) that likely will be required for a high-purity copper waste package because of its lower yield strength will likely impart greater creep strength as well. But this supposition will need to be supported by analysis of stresses (and strains) that will develop in the postemplacement period and by a comparison of the results with available creep rupture data for this material over the temperature range of interest. Creep appears to be a less significant potential degradation mode for the solid-solution hardened copper-based alloys and the austenitic materials.

### 8.3.5.9.1.1.2 Subactivity 1.4.1.1.2: Microstructural properties

Objectives

The objective of this subactivity is to compile available information and characterization of the microstructures of the candidate copper-based and austenitic materials to predict the microstructural properties of the as-emplaced container. Predictions of microstructural properties are compared with examinations of microstructures in prototype containers. The characteristics of the as-emplaced container microstructure serve as a basis for predicting what microstructural changes will occur in the postemplacement time periods.

Parameters

Because the microstructure is intimately related to fabrication process variables and, in some instances, to relatively small compositional variations, this dependence will be documented. The microstructures of the fusion zone and heat-affected zones around the weld must also be characterized; characterization of these microstructures depends strongly on the welding process variables and, in some welding processes, on the composition of the filler materials. The microstructural features of importance include the following:

1. Primary phases present and their distribution.

2. Secondary phases, their distribution, and evidence of precipitation reactions.

3. Segregation effects.

4. Grain size and distribution of grain size.

5. Evidence of preferred orientation.

6. Identification and distribution of nonmetallic inclusions.

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The time at elevated temperature (during the container fabrication and closure process) is influential in determining these features.

Description

The work in this subactivity is primarily concerned with the microstructure of the emplaced container. Projections of microstructural changes from the time of emplacement form the basis of analysis for the different corrosion, oxidation, and embrittlement degradation modes that can occur after emplacement. These projections are pursued in Information Needs 1.4.2 (Section 8.3.5.9.2) and 1.4.3 (Section 8.3.5.9.3) under the topic of aging phenomena.

A major emphasis in this subactivity is development of the ability to predict what the microstructural features should be for the as-emplaced container. These predictive abilities derive from an understanding of physical and mechanical metallurgy of the container material and the effect of the thermochemical process history on the microstructure of the container material. This will be substantiated by (1) examination of laboratory-size specimens that are produced to simulate the fabrication and welding processes to be used on actual size waste package containers and (2) examination of prototype containers (of the dimensions and process history as the actual production container but not filled with waste).

The experimental work in this subactivity will establish what population of examined microstructures of laboratory- and prototype-size containers constitutes a representative sample population of production-size containers. Standard laboratory metallographic and microscopic techniques are available for characterizing microstructures. Advanced microscopic techniques may be needed to the extent of resolving subcritical size particles that would later grow into potentially detrimental microstructural features. The need for these will be indicated by the modeling activities (Information Need 1.4.3) with regard to the container material and degradation mode(s) requiring this amount of attention. After a material is selected for the final design and after fabrication of prototype containers is undertaken, a through characterization of a representative as-fabricated and as-assembled container will be needed.

Most of the techniques for thorough characterization of microstructures involve destructive examination of the metal cross-section; therefore, quality control of the container production stream is obtained by periodic examination of unfilled quality control containers that have the same process history as the filled containers. Work in this subactivity will provide the technical basis for establishing the frequency of this inspection for process control. Some nondestructive, semiquantitative techniques can be routinely used in the production facility for evaluating certain microstructural features (e.g., amount of ferrite in the weld determined by magnetic flux measurement techniques). Which features to pursue will depend on the outcome of the modeling activities in deciding which microstructural features are most important to affecting the container performance. Some destructive testing on prototype or witness specimens will be needed to confirm that the desired microstructure is obtained. For example, evidence of copper oxide inclusions in copper or significant sigma phase formation in candidate

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austenitic stainless steels during the fabrication and welding process would likely be considered detrimental and would be cause for rejection.

With respect to the copper-based materials, the microstructures are generally simpler than those for the austenitic materials; however, segregation effects in the alloys may be more important because of the electrochemical implications of the wide difference between the more noble copper and the active alloy additions. Because a high percentage of copper is produced by recycling, the accumulation of potentially harmful impurities and their effect on embrittlement is noted for evaluation. Specifications requiring virgin copper or high-purity remelt scrap might be necessary.

Detailed microstructural analyses will be conducted on base metal, weld metal, and weld heat-affected zones. These analyses will be conducted on material in the condition expected for the as-fabricated container and also in the condition expected after simulated long-term exposure in the repository. Advanced techniques, including transmission electron microscopy and scanning electron microscopy, can be used to resolve features 10 angstroms in size. Moreover, microchemical analysis techniques of the matrix, precipitate and dislocation and grain boundary structures will be conducted using scanning transmission electron microscopy and Auger electron spectroscopy. These techniques will allow full characterization of the morphology and chemistry of the microstructure. The overall program will allow assessment of the stability of these features over the long term in the repository.

Microstructural stability will be a criterion in the selection of the metal barrier. Further discussion is contained under degradation mode testing (Section 8.3.5.9.2.3.2) and degradation modeling (Section 8.3.5.9.3.2.1).

The microstructure of the as-fabricated and as-assembled (by welding or other process) prototype containers must be thoroughly characterized, because it is not always possible to perform successfully all the possible variations scaling up from specimen or coupon-size workpieces. This more extensive characterization will only be pursued on the material selected for the advanced design and the process selected for actually fabricating the waste package container.

### 8.3.5.9.1.1.3 Subactivity 1.4.1.1.3: Physical properties

Objectives

The objective of this subactivity is to compile those physical properties whose values are needed for design and for projections of changes in the container in the postemplacement environment (i.e., temperature field, radiation field, stress field).

Parameters

The physical properties of interest include

1. Thermal conductivity.

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2. Density.
3. Coefficient of thermal expansion.

Description

These physical properties are not site or environment dependent, and so compilation from existing literature sources should be sufficient. How these properties depend on such factors as alloy compositions (and permissible variations) and temperature, however, is needed as waste package design information. These properties are not expected to be significantly affected by the fabrication processes for forming or joining the container materials.

8.3.5.9.1.1.4 Subactivity 1.4.1.1.4: State of stress in the container

Objectives

The objective of this subactivity is to analyze the state of stress at a number of locations in the container and to project the changes in the state of stress with time and temperature during the containment and postcontainment periods. When possible or feasible, the analysis will be supplemented by actual stress measurements on prototype containers.

Parameters

The initial state of stress at emplacement will depend on many processing variables in forming, assembling, joining, and handling the container and the residual stresses that these different processes impart to the container. The steady-state service load on the container (mostly due to its own weight and that of the contents) also figures in the analysis. Projections of the stress to different postemplacement time periods will consider the effects of any expected additional static or dynamic loads. The state of stress is concerned with the magnitude, nature (tensile, compressive, or shear), and axiality of the stresses, and the corresponding strains in the container materials associated with the stresses.

Description

The stress in the container is expected to vary considerably from location to location. The container lid and bottom will likely be designed with a thicker section than the main body in an effort to contour the stress. Different fabrication processes may be used for the main body (e.g., rolled and welded plate) and the lid and bottom sections (e.g., forgings), resulting in different stress and microstructure patterns. Other container fabrication processes under consideration eliminate some or all of the assembly welds, and it is possible to anneal the container body before the waste form is placed inside. These considerations have the objective of reducing residual stress in the as-fabricated container shell. The residual stress is expected to be highest at the closure weld, since it will be impractical to relieve all this stress by a postweld heat treatment. (Some localized stress relieving may be possible, if this can be performed in a hot cell and without damaging the waste form.) Proper selection of the welding process and weld parameters can create less residual tensile stress at the surface. Another possibility is a postweld surface peening process to put the outside surface of the weld in compression. All the weld processes and processes to mitigate against high residual tensile stresses, such as surface peening, will have to be evaluated in terms of being practical in a hot cell and not creating an undesirable side-effect problem.

There may be significant differences in fabrication processes between the copper-based and austenitic groups, which may lead to separate stress analyses. The density of copper is higher than that of the austenitic materials, and a thicker wall section will probably be specified if high-purity copper (CDA 102) is selected. The yield stress of copper is considerably lower than that of the other candidate materials. The copper-based alloys have yield strengths comparable to those of the candidate austenitic materials.

Calculational analyses of the state of stress will be supported by actual strain gage measurements on prototype containers with simulated waste form contents and closure welds.

Because all the candidate materials are corrosion resistant, a criterion for selection will be that wall thinning due to corrosion processes during the containment period will not be of engineering significance. Part of the effort will be to consider the effect of pits and other localized corrosion phenomena as stress raisers and potential sites of crack nucleation. A fracture mechanics, crack-growth methodology will be used. Stress-corrosion crack growth rate data will be obtained for the base metal, weld metal, and weld heat-affected zone. The data will cover conditions expected in the as-fabricated container and also the condition expected after simulated long-term thermal exposure in the repository. Failure of the container by pitting and other forms of localized corrosion, as well as resultant effects such as nucleation of stress corrosion cracks, will be included in the selection criteria.

### 8.3.5.9.1.1.5 Subactivity 1.4.1.1.5: Characterization and inspection of weld integrity

Objectives

The objective of this subactivity is to determine the soundness of the weld joints, with primary emphasis on inspection of the final closure weld. Nondestructive evaluation techniques are available to make this inspection and to determine the nature, population, size, and distribution of flaws. Detection of flaws in the welded region is important in ensuring the initial integrity of the as-emplaced waste package; analyses of some of the possible degradation modes (in Information Needs 1.4.2 (Section 8.3.5.9.2) and 1.4.3 (Section 8.3.5.9.3)) depend on whether flaws are present above a critical size at which they might be expected to grow as cracks during later time periods.

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Parameters

The parameters of this subactivity include

1. Weld process selected.
2. Weld process parameters (particularly those related to heat input).
3. Composition of filler material (for some weld processes).
4. Composition of weld cover gas.
5. Microstructure of weld.
6. Inspection method selected.

These parameters are interactive in determining the integrity of the weld. They influence how the weld can be inspected and how the signal or pattern from the technique used for the inspection can be interpreted.

Description

The welding process and welding parameters (such as heat input and the rate of heat input, which are often determined by the current and voltage, number of passes, and time of each pass) have an important effect on the weld integrity. Similarly, the cooling rate after welding is important. Both filler and autogenous processes are under consideration. The composition and microstructure in the fusion zone are important. The nature and composition of the protective cover gas are important to prevent significant oxidation in the weld region. The weld geometry, weld thickness, metallurgical composition, grain size, grain orientation, and other metallurgical and microstructural considerations govern the kinds of nondestructive evaluation techniques and the sensitivity and precision with which flaws can be detected.

Autogenous weld processes are commonly used for the copper-based materials. The welded microstructure is expected to be simpler for the copper-based materials than other materials considered, but any tendency for the alloying elements to segregate will need to be evaluated. The high heat input required to weld pure copper may cause possible problems for the waste form inside the container. Also, high-purity copper tends to pick up oxygen readily, and so control of the cover gas composition becomes very important. Small amounts of oxygen in the copper can cause embrittlement in some environments, but small additions of deoxidizing elements (most commonly phosphorus) alleviate the problem.

Autogenous and filler metal processes can be used to weld the austenitic materials. For welding some of the austenitic materials, a filler material of somewhat different composition from that of the base material is used to produce the desired microstructure or to compensate for alloying elements that are lost by oxidation or evaporation. It will be important to demonstrate that any compositional differences between the filler and base materials do not result in undesirable galvanic interaction between the two materials. Also, the composition and control of the composition of the cover gas during the welding operation is important in ensuring high weld integrity and process consistency. The number of alloy components (including the titanium addition used to stabilize carbide formation) and the all-austenitic structure of alloy 825 sometimes presents concerns about weld cracking; these are overcome by control of the microconstituents (especially the carbon, phosphorus, and sulfur) and the cover gas composition.

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Because the weld inspection technique is so closely tied to the material and process variables, selection and development of the inspection technique will parallel the efforts made on selecting the candidate container material and on selecting and developing the container fabrication and closure weld processes. Because of nondestructive inspection techniques, all the containers destined for the repository can be inspected. However, many details of conducting the inspection will need to be addressed in the future, including the constraints of remotely performing the operation. Interpretation of the signals or images produced by the nondestructive test will also have to be worked out. This will likely involve inspection of prototype weldments and container sections with intentional flaws of different kinds, sizes, and distributions.

Since welds and weld heat-affected zones often limit the performance of structures, particular attention will be focused on the properties and performance of these regions of the container. The material properties and degradation models are discussed in Sections 8.3.5.9.2 and 8.3.5.9.3.

### 8.3.5.9.1.1.6 Subactivity 1.4.1.1.6: Characterization of the container surface

Objectives

The objective of this subactivity is the detection of potentially harmful surface conditions on the as-emplaced container, resulting from handling operations. These conditions (seemingly innocuous at emplacement and, therefore, not a cause of waste package rejection) may lead to conditions that will favor one of the degradation modes discussed in Information Needs 1.4.2 (Section 8.3.5.9.2) and 1.4.3 (Section 8.3.5.9.3). Two classes of conditions are of concern: (1) mechanical defects such as scratches and gouges that could develop into crevices or into stress raisers and (2) chemical contamination of the surface. Of particular concern are residues of chloride ion that could result in locally high concentration of chloride ion developing at a later period. Both mechanical defects and chemical contamination will be of more concern with types 304L and 316L stainless steel (SS) materials, which are more susceptible than the other materials being considered.

Parameters

The production of surface defects and contamination depends on many process and operational variables. These will become better defined as decisions on process selections are made and details of the operations are more focused. The critical sizes of surface defect or levels of chemical contamination will be determined in activities dealing with the various degradation modes and with the sensitivities of accelerating the degradation modes (Information Needs 1.4.2 and 1.4.3).

Description

Much of the work in this subactivity will be directed toward developing ways of detecting small mechanical defects and surface residue concentrations. Detailed specifications for container handling in the surface facil-

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ity and in the repository will be made in an effort to minimize potentially harmful surface effects. The extent to which the activity will be pursued also depends on the material ultimately selected. (Type 304L SS, for instance, will be much more susceptible to crevice effects and chloride residues than will alloy 825.) Characterization of the container surface may not be as critical for the copper-based materials because these materials are not nearly as susceptible to crevice-induced corrosion effects, or to chloride-induced corrosion problems, as other materials considered.

### 8.3.5.9.1.2 Activity 1.4.1.2: Integrate design and materials information (alternate barriers investigations)

This activity has been expanded into an alternate barriers investigations, to bring forward one or two viable material-system, design, and fabrication alternatives, in case a waste package fabricated from any of the six candidate materials cannot achieve the containment objectives allocated to the container. These investigations shall integrate design and materials information for waste package containers based on materials systems that fall in the following three classes:

1. Ceramic-metal systems.
2. Bimetallic/single metal systems.
3. Coatings and filler systems.

An obvious advantage of the alternate barriers investigations is the potential to clearly project repository performance that is superior to what is possible with the metal barrier candidates under consideration. Because of the large number of containers needed, cost-effective fabrication and material availability are also important considerations. To meet this challenge, the following concepts will be evaluated:

1. Ceramic-metal systems. Wherein the choice of a suitable ceramic, alumina ($Al_2O_3$) and titania ($TiO_2$), will provide the best possible long-term corrosion resistance because of higher chemical stability in the near field environment. Additionally, ceramic composites can increase the toughness and delayed fracture resistance of the ceramic. If required, a metal container for the ceramic monolith can serve as a fabrication aid and enhance the resistance to damage incurred during transportation and handling of the metal ceramic-metal container. A very important consideration is the ease and reliability of the closure in any container involving a ceramic. Both metallic closure concepts and ceramic closure concepts will be considered.

2. Bimetallic/single metal systems. Wherein a bimetallic system will use galvanic effects to provide long-term corrosion resistance. The inner liner, cathodic to the outer liner, will be designed for long-term stability and corrosion resistance at lower temperatures. The outer liner will provide corrosion resistance in the short term at higher temperatures and higher gamma dose rates. Alternate single metals such as titanium, its dilute alloys, and high-nickel

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corrosion-resistant alloys, with high potential corrosion resistance compared with metal barrier selections, will be considered.

3. Coatings and filler systems. Wherein coatings will be used primarily to enhance the performance of a single metal barrier. Coatings must demonstrate closed porosity and long-term stability in regard to substrate adherence, resistance to cracking, and corrosion resistance. Fillers (metal or nonmetallic) will provide added mechanical support between spent fuel elements and the container. Thermal properties must be compatible with waste package design regarding maximum waste-form temperature. Fillers will also provide additional long-term protection against corrosion and control the release of radionuclides.

Five subactivities support this activity.

### 8.3.5.9.1.2.1 Subactivity 1.4.1.2.1: Survey of alternative barrier designs, materials, and processes to determine feasibility of fabricating a satisfactory waste package

The purpose of this subactivity is to first survey those designs, materials, and fabrication and closure methods that fall into the three categories just discussed and that have high potential for meeting the containment objectives. Because of the large number of containers needed, the impact of cost on container fabrication, closure, and material selection will also be carefully considered.

Fabrication studies will be divided into at-repository and off-repository categories with the goal of prefabricating and inspecting as much as possible off the site. Only closure and postclosure inspection would be completed at the repository. Cost assessments, as well as fabrication feasibility, will be considered. Economics of scale will be taken into account.

Alternate single and bimetal concepts will focus on three basic metal fabrication methods: (1) rolled and welded, (2) centrifugal casting, and (3) extrusion. Gas pressure bonding will be considered for bimetal concepts. For ceramic-related concepts, it is expected that (1) extrusion/slip cast/sinter, (2) cold press/sinter, and (3) hot isostatic pressing will be applicable. Special fabrication considerations will be required for composite candidates and coating/liner concepts.

Later in the survey, the fabrication survey will be more focused and more detailed regarding the specific requirements. For ceramic containers, an early assessment will be made to determine if current technology and industrial experience can produce ceramics in the size required and with microstructural integrity that will meet repository requirements.

After the screening of alternative candidates, two alternates from each category will be chosen for detailed study. A set of criteria will be developed to guide the selection process. Some of these guidelines will include

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1. How will the container meet the performance objective of substantially complete containment?

2. Can long-term performance be predicted?

3. Is the material available and is fabrication practicable?

4. What are the estimated costs?

5. Can the fabricated and sealed container be adequately inspected?

Each alternative concept will be evaluated by a separate team of consultants, some with direct industrial experience fabricating the same materials in related product lines.

## 8.3.5.9.1.2.2 Subactivity 1.4.1.2.2: Mechanical properties

### Objectives

The objective of this subactivity is to compile relevant data on the mechanical properties of the candidate materials over the temperature range of interest (approximately room temperature to 300°C).

### Parameters

The principal mechanical properties of interest for both ceramic and metallic materials that may be used in the alternate barriers investigations are

1. Yield and ultimate strength in all stress modes, percent elongation, and reduction in area.

2. Strain rate effects on impact strength, fracture toughness, and ductility.

3. Elastic constraints.

4. Residual stress after fabrication and closure.

Knowledge of the effect of fabrication and closure on the microstructure and mechanical properties is needed.

## 8.3.5.9.1.2.3 Subactivity 1.4.1.2.3: Microstructural properties

### Objectives

The objective of this subactivity is the compilation of available information and the completion of required experimentation to characterize microstructure as it relates to mechanical properties, corrosion resistance, and ultimately to the overall performance of the as-emplaced waste package

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container. Because the microstructure is intimately related to fabrication process variables and, in some instances, to relatively small compositional variations, this dependence will be documented.

Predictions of microstructural properties are compared with examinations of microstructures in prototype containers. The characteristics of the as-emplaced container microstructure serve as a basis for predicting what microstructural changes will occur in the postemplacement time periods.

Parameters

The microstructural features that will impact the performance of ceramics, metallic alloys, and coatings and fillers that are candidates for inclusion in the alternative barriers investigations include

1. Primary and secondary phases, impurity phases, and segregation effects.

2. Grain size, its distribution, and any preferred orientations.

3. Discontinuities such as porosity and cracking (primarily in ceramics and coatings) and other flaws, their sizes, and size distributions.

Discussion

A major emphasis in this subactivity is development of the ability to predict what the microstructural features should be for the as-emplaced container. This will be accomplished by (1) examination of laboratory-size specimens that are produced to simulate the fabrication and closure processes to be used on actual size waste package containers and (2) examination of prototype containers.

The experimental work in this subactivity will establish what population of examined microstructures of laboratory- and prototype-size containers constitutes a representative sample population of production-size containers. Standard laboratory metallographic and microscopic techniques are available for characterizing microstructures. Advanced microscopic techniques may be needed to the extent of resolving subcritical size defects that would later grow into potentially detrimental microstructural features. The need for these will be indicated by the modeling activities with regard to the container material and degradation mode(s) requiring this amount of attention. After a material is selected for the final design and after fabrication of prototype containers is undertaken, a thorough characterization of a representative as-fabricated and as-assembled alternate container will be needed.

The microstructure of the as-fabricated and as-assembled prototype containers must be thoroughly characterized, because it is not always possible to perform successfully all the possible variations scaling up from specimen or coupon-size workpieces. This more extensive characterization will only be pursued on the material selected for the advanced design and the process selected for actually fabricating the waste package container.

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8.3.5.9.1.2.4 Subactivity 1.4.1.2.4: Thermophysical properties

Objectives

The objective of this subactivity is to compile those physical properties whose values are needed for design and for projections of changes in the container in the postemplacement environment (i.e., temperature field, radiation field, and stress field).

Parameters

The physical properties of interest include

1. Thermal conductivity and radiation effects thereon.
2. Density.
3. Heat capacity.
4. Crystal structure.
5. Coefficient of thermal expansion.

Discussion

These physical properties are not site or environment dependent, and so compilation from existing literature sources should be sufficient. How these properties depend on such factors as composition and temperature, however, is needed as waste package design information. These properties are not expected to be significantly affected by the fabrication or closure processes.

8.3.5.9.1.2.5 Subactivity 1.4.1.2.5: Nondestructive characterization of the alternate barrier investigations waste package container

Objectives

The objective of this subactivity is the development of nondestructive characterization methods and the application of these methods to the detection of potentially harmful internal and surface defects on the ceramic-metal, bimetallic/single metal, or coating and filler-based waste package container that evolve from the alternative barriers investigation. These potentially harmful interior and surface defects may result from container fabrication, post-fabrication closure, or handling operations.

Parameters

The nondestructive characterization methods of potential usefulness include

1. Visual inspection, with or without optical aids.
2. Liquid penetrant inspection.
3. Magnetic particle inspection.
4. Radiography.
5. Eddy current.
6. Ultrasonic.
7. Acoustic emission.

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Depending on the particular candidate container material system being inspected, several of these nondestructive characterization methods will be selected for evaluation of intentionally introduced flaws that simulate the defect types that are expected (discussed below).

The emphasis in this subactivity will be on adaptation of presently operational nondestructive characterization methods to the alternative concepts, with the development of new characterization methods being undertaken only if a pressing need for such work is clearly identified.

Discussion

The presence of undetected and unremoved interior and/or surface defects could accelerate one or more of the degradation modes discussed in Information Needs 1.4.2 (Section 8.3.5.9.2) and 1.4.3 (Section 8.3.5.9.3). Two classes of conditions are of concern to ceramic and metallic alternate barrier investigations designs:

1. Mechanical defects such as porosity, cracks, scratches, and inclusions, all of which could degrade container performance through delayed failure after emplacement.

2. Chemical contamination of the surface. Of particular concern here are impurity residues (such as chlorides) that could ultimately concentrate over time and contribute to container failure.

The production of surface defects and contamination depends on many process and operational variables. These will become better defined as decisions on process selections are made and details of the operations are more focused. The critical sizes of surface defect or levels of chemical contamination will be determined in activities dealing with the various degradation modes.

Much of the work in this subactivity will be directed toward developing ways of detecting small mechanical defects and surface residue concentrations. Detailed specifications for container handling in the surface facility and in the repository will be prepared to minimize potentially harmful surface effects. The extent to which the activity will be pursued also depends on the material ultimately selected.

## 8.3.5.9.2 Information Need 1.4.2: Material properties of the container

### Technical basis for addressing the information need

This information need addresses the material properties of the candidate metals that are needed to establish the prediction of the performance of the selected container material. Behavior of weld metal and weld heat-affected zones will be considered in addition to the base metal. Because the borehole liner will be made from the same alloy family as the container, information gathered here will provide a description of the performance of the borehole liner. Information from this testing program supplies the models discussed in the Information Need 1.4.3 (Section 8.3.5.9.3) for each possible degra-

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dation mode that the container might experience in the postemplacement repository environment.

This information need also covers the characterization of the material systems proposed for use in an alternative design for spent fuel packages. This option will be pursued since the feasibility of producing such a liner is favorable if the demonstration that a metal-only waste package can meet the containment objectives proves too difficult.

Link to the technical data chapters and applicable support documents

The six candidate metallic container materials for the waste package are introduced and discussed in Section 7.4.2 on metal barriers. Representative mechanical properties and the metallurgical industry standard composition ranges are given in Section 7.3 for each candidate material. The post-emplacement environmental conditions that will surround the containment barrier are discussed in Section 7.4.1; the geochemical modeling of the environment is described in Section 7.4.4. The waste package design and a brief discussion on fabrication and welding (or other closure) processes for producing the waste package are given in Section 7.3.

The material presented in Section 7.4.2 deals with experimental work performed by the Yucca Mountain Project from 1983 to 1986 and data available from other published sources. A large portion of this work is centered on austenitic stainless steels (including some work on austenitic alloy 825), with a smaller portion centered on copper and its alloys. Although the earlier emphasis was on the austenitic stainless steels, all candidate materials are being equally considered in the selection process for the material to be used in the license application design.

Parameters

Information needed from other information needs includes

1. The candidate container materials (Information Need 1.4.1, Sec- tion 8.3.5.9.1).

2. The design features that influence container material selection and performance of the container material (Information Need 1.4.1).

3. Characterization of the as-emplaced container with respect to its mechanical microstructural, and physical properties (Information Need 1.4.1).

4. Scenarios developed to describe the waste package near-field environment (Information Need 1.5.3, Section 8.3.5.10.3).

5. Results of geochemical modeling calculations to give the chemical composition and speciation of solutions that may contact the container (Information Need 1.5.3).

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6. Feasibility of using a waste package selected from one of the three concept classes covered in the alternative barriers investigations (Section 8.3.5.9.1.2) (Information Needs 1.4.1 and 1.10.2, Section 8.3.4.2.2).

The following data are to be obtained:

1. A selection of the metallic container materials to be used for advanced design analysis. The basis on which the selection is made and the methodology used in carrying out the process are parts of this information need.

2. Analyses of the different degradation modes that the candidate container materials can undergo in the thermal and environmental conditions expected in the repository after waste package emplacement.

3. A laboratory testing program centered around the selected material and the assessment of its likely degradation modes. The results from the testing program are used in modeling activities to predict the rates at which the different degradation modes will operate in the container material.

4. A laboratory testing program conducted to evaluate the metallurgical condition and properties of the full size container. This work will be conducted only on the selected metal barrier alloy. The work will consist of detailed microstructural and microchemical analyses as well as corrosion and mechanical properties tests on coupons cut from the container.

The work in this information need is divided into four activities. The first activity concerns the process for selecting the material for the license application design. The next three activities are specific to the container materials: (1) copper-based materials, (2) austenitic materials, and (3) material systems that evolve from work in the alternative barriers investigations. These three activities deal with the analyses of the different degradation modes and the testing program needed to provide data for the predictive performance models in Information Need 1.4.3 (Section 8.3.5.9.3). The subactivities described in the material-specific activities will not all be completed. Some of the analyses for each material category need to be performed to provide input into the selection process, but the full range of testing activities and modeling activities will be carried out only on the material selected for the final design.

### 8.3.5.9.2.1 Activity 1.4.2.1: Selection of the container material for the license application design

This activity is focused on selection of the container material for more detailed characterization of its properties relevant to attaining the performance objectives of the postemplaced container. This activity involves

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the metallic materials and ceramic-metal systems, bimetallic/single metal systems, and coatings and filler systems. Two subactivities support this activity.

### 8.3.5.9.2.1.1 Subactivity 1.4.2.1.1: Establishment of selection criteria and their weighting factors

Objectives

The objective of this subactivity is to develop a methodology to select the container material from the list of candidate materials. A peer review group will be formed to review this methodology and its use to arrive at the final material choice.

Parameters

The following is a preliminary list of the criteria for selecting a container material for the license application design:

1. Which material will meet the performance allocated to the container in achieving the containment objectives (substantially complete containment under anticipated processes and events occurring in the repository)?

    a. Resistance to oxidation.

    b. Resistance to general aqueous corrosion.

    c. Resistance to environmentally accelerated cracking (stress corrosion cracking and hydrogen embrittlement).

    d. Resistance to pitting, crevice, or other localized attack.

    e. Demonstration of adequate mechanical properties.

    f. Resistance to mechanical embrittlement.

2. Can the performance of the material under repository conditions be adequately predicted?

    a. Predictability of physical and chemical properties of as-emplaced container.

    b. Existence of models to explain and predict degradation phenomena, or ability to develop such models.

    c. Existence of models to extrapolate laboratory data on degradation phenomena to repository time scales and conditions, or ability to develop such models.

3. Will the container material interact favorably with other components?

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   a. Interactions with waste form.
   b. Interactions with borehole liner.
   c. Interactions with the package environment.

4. Can the container be made of this material?

   a. Fabricability of container body.
   b. Weldability of container (closeability if a nonwelded closure).
   c. Inspectability of closure.

5. Are the container material and process for fabricating it practicable?

   a. Availability of container material.
   b. As-fabricated container costs.
   c. Quality control requirements (and costs).
   d. Repository handling costs.

6. How can the confidence in the selection be gained?

   a. Previous engineering applications with the material.
   b. Available data base on material.
   c. Favorable (or unfavorable) experiences with material.

Weighting factors for each of the preceding criteria will need to be established. It is expected that criteria 1, 2, and 4 will have the heaviest weighting, but all the criteria have some importance. One approach is to assign a maximum number of points to each item in the criteria list and a minimum number for each item that the material must pass. As a rather extreme sample, it does no good to have a highly corrosion resistant material that cannot be fabricated and closed.

Where appropriate and available, examples of methods that have successfully been used to predict longer term behavior of materials from short-term laboratory or field tests will be used. Examples may derive from atmospheric corrosion testing, marine corrosion testing, underground testing, chemical process industry testing, and nuclear and fossil fuel power plant testing. These examples will provide information to some of the items listed in criteria 2 and 6.

Description

Development of the selection criteria and organization of the peer review group are the first items to be completed in this subactivity. The Yucca Mountain Project will use its own staff and consultants to develop the selection criteria and weighting factors. The selection criteria and weightings will then be reviewed by the peer review panel. Following revision, if necessary, the criteria will be used to assess the candidate materials and select a material or materials. The peer review panel will then review the selection assessments. The peer review panel will consist of approximately seven individuals with backgrounds in different areas of metallurgy and materials science and with different work experiences to achieve a balance of viewpoints and perceptions.

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8.3.5.9.2.1.2 Subactivity 1.4.2.1.2: Material selection

After the review panel is organized and selection criteria established, the next step is to perform the selection. Input into the selection process comes, in part, from (1) the Yucca Mountain Project analyses on the significance of different possible degradation modes (discussed in the next activity) and (2) available published literature concerning the performance of candidate materials in applications and environments that have analogies with expectations of conditions in the proposed Yucca Mountain repository. Depending on the outcome of the selection process, the Yucca Mountain Project may elect to carry more than one material forward for additional characterization for the license application design.

8.3.5.9.2.2 Activity 1.4.2.2: Degradation modes affecting candidate copper-based container materials

This analysis concerns the analysis of which degradation modes have any significant chance of occurring on the candidate copper-based materials in the postemplacement periods and laboratory testing activities to provide information for the modeling activities discussed in Information Need 1.4.3 (Section 8.3.5.9.3). The candidate copper-based materials are Copper Development Association (CDA) 102 (high-purity, oxygen-free copper), CDA 613 (aluminum bronze), and CDA 715 (70/30 copper-nickel).

Eight subactivities address the evaluation.

8.3.5.9.2.2.1 Subactivity 1.4.2.2.1: Assessment of degradation modes in copper-based materials

Objectives

The objective of this subactivity is to evaluate the likelihood of each potential degradation mode occurring under conditions anticipated at Yucca Mountain.

Parameters

The parameters for this subactivity are

1. Literature data documenting the causes for failure of this class of materials.

2. Interpretation of these causes of failure in the context of fabricating a container (Information Need 1.4.1, Section 8.3.5.9.1) and emplacing it in the Yucca Mountain repository (Information Need 1.5.3, Section 8.3.5.10.3).

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Description

The corrosion and oxidation resistance of the copper-based material relies first of all on the electrochemical nobility of copper and secondly on the formation of a protective surface layer. The protective layer is a thick oxide that forms on the copper-based materials and acts as diffusion barrier to mass transport. Thus, the rates of oxidation and general aqueous corrosion are initially high but become progressively lower with the growth of the protective layer. The rate of corrosion or oxidation is expected to be proportional to the oxidation-reduction potential of the environment, so that the oxidation or corrosion rate increases with an increase in the oxidizing nature of the environment. On the other hand, when the protective layer is broken, the underlying metal is not very active electrochemically. Hence, active-path corrosion phenomena (e.g., pitting and stress corrosion cracking) are usually not as severe as they are with active-passive materials such as the austenitic materials when the passive film is broken on these. A more complete discussion of these points is found in Section 7.4.2.

Copper and its alloys do have their vulnerabilities, and a substantial part of the laboratory testing program is focused on whether these vulnerabilities are substantive in the context of conditions at Yucca Mountain. There are three areas of particular concern:

1. The formation of strongly oxidizing species such as nitrogen dioxide or nitric acid in irradiated moist atmospheres is expected to increase the corrosion rates of copper.

2. The presence of ammonia, which can be formed by radiolysis of atmospheric gases in some circumstances, is a concern because it forms very soluble complexes with copper and destroys protective films. As a consequence, the general corrosion rate increases substantially, and ammonia provokes stress corrosion cracking (transgranular crack pattern) in copper and many copper-based alloys. Other chemical species have been implicated in causing stress corrosion cracking in copper-based materials; as with ammonia, the role of these species is probably one of destabilizing the protective film. Whether the presence of any of these or similar species would be significant in the postemplacement environment at a Yucca Mountain repository needs to be demonstrated.

3. The presence of segregation effects in the long term, particularly if there is segregation of the less noble constituent from the copper, creating a large galvanic cell within the alloy. The segregation effects may be of concern even though copper and the two candidate alloys appear to have simple metallurgical microstructures.

Classification of degradation modes generally follows from the morphology of the attack (uniform, localized, stress-assisted, embrittlement) as indicated earlier in the material under the issue-level discussion. For the purposes of organizing the work in this and the next information need, the degradation modes have been placed into seven groups. This grouping is based on the performance models discussed in Information Need 1.4.3 (Section 8.3.5.9.3). The analysis in this information need emphasizes the vulnerabilities of the materials, and much of the effort is directed toward

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establishing how much these vulnerabilities matter in demonstrating performance of the material.

Seven degradation modes of copper and copper-based alloys are being considered:

1. Metallurgical aging and phase stability.
2. Low temperature oxidation.
3. General aqueous corrosion.
4. Hydrogen entry and embrittlement.
5. Pitting, crevice, and other localized attack.
6. Stress corrosion cracking.
7. Other potential degradation modes.

The order these degradation modes were presented in does not imply a ranking according to importance, but rather was developed to streamline the discussion in this and the next Information Need (1.4.3). In summary, this activity reviews the pertinent literature on the different copper-based material degradation modes as well as the relevance of previous Yucca Mountain Project laboratory work (Section 7.4.2). This activity assesses the potential for occurrence of each mode and estimates the severity of attack. All this information provides input to the container material selection (Activity 1.4.2.1, Section 8.3.5.9.2.1).

8.3.5.9.2.2.2 Subactivities 1.4.2.2.2 through 1.4.2.2.8: Laboratory test plan for copper-based materials

The following subactivities cover the laboratory test plans and programs appropriate to each of the enumerated degradation modes:

| Subactivity | Degradation mode |
|---|---|
| 1.4.2.2.2 | Metallurgical aging and phase stability |
| 1.4.2.2.3 | Low temperature oxidation |
| 1.4.2.2.4 | General aqueous corrosion |
| 1.4.2.2.5 | Hydrogen entry and embrittlement |
| 1.4.2.2.6 | Pitting, crevice, and other localized attack |
| 1.4.2.2.7 | Stress corrosion cracking |
| 1.4.2.2.8 | Other potential degradation modes |

These subactivities will be discussed as a group and will be pursued (1) according to which material is selected for the advanced designs and (2) where literature review and analysis indicate the need to obtain data specific to Yucca Mountain conditions. The laboratory test plan will only be carried out in full for the material(s) selected for the advanced designs. The sequence of these major activities is given in the schedule and milestone section at the end of this information need.

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Objectives

For the selected material, the objective of this group of subactivities is to develop and implement a laboratory test plan to provide information to the modeling activities in Information Need 1.4.3 (Section 8.3.5.9.3). The test plan is oriented toward quantifying particular degradation modes or proving that the degradation mode(s) will not be operative under conditions anticipated at the Yucca Mountain repository.

Parameters

The expected important parameters for each of the degradation modes are listed under the respective activity for the modeling work in Information Need 1.4.3.

Description

The plan is to develop an experimental approach for each of the possible degradation modes. In many instances, the laboratory investigations are expected to be performed under environmental, metallurgical, or strain conditions that are intentionally made more severe than those expected to occur in the repository environment. This approach is used to accelerate the phenomenon under investigation so that measurement can be made in a reasonable amount of laboratory time (hours, days, weeks, months, and in some instances, up to a few years). Also, confidence in the modeling activities is gained by systematically extending the period of observation from shorter times with more aggressive conditions to making predictions for longer times with less aggressive conditions and then performing tests under these conditions for confirmation.

A technical review of the test plans and procedures will be conducted to assess the adequacy of the test conditions for the degradation modes. The review will be intended to ensure that the tests cover the range of conditions anticipated in the repository over the period of concern.

This approach requires sufficient understanding of the causative mechanisms for each of the degradation modes so that predictions for container failure can be made, as stated in the performance goals in Issue 1.4 and consistent with the required confidence level (Table 8.3.5.9-1). It is further recognized that several of the degradation modes are rather closely related, and it is possible that one or more can be operable under a given set of conditions. For example, aging and segregation reactions can lead to phases that create local electromechanical cells within the material.

Experiments will be performed to determine the nature of radiolytic products in the water or air that may be deleterious to copper-based alloys such as ammonia and other nitrogen-bearing compounds. Localized corrosion and stress corrosion tests in water and vapor containing these nitrogen-bearing compounds will be conducted. A criterion for selection of the metal barrier alloy will be broad resistance to these types of attack. Further details are provided in Sections 8.3.5.9.3.1.5 and 8.3.5.9.3.1.6.

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The long-term, low-temperature oxidation is expected to condition the surface of the container and will influence all the other subsequent degradation modes. These points are also taken into account in the modeling activities.

In the category of "other potential degradation modes" particular corrosion and mechanical degradation processes are possible, but unlikely, based on the current understanding of conditions of Yucca Mountain. With regard to high-purity copper (CDA 102), the possibility of low temperature creep has been discussed previously and largely discounted because of the expected use of a somewhat thicker section (2-3 cm) for a container fabricated from this material. The relatively low temperatures that will occur on the container surface (maximum peak temperatures in the range of 230 to 250°C for the spent fuel packages with highest thermal loading) suggest that high-temperature metallurgical deformation and fracture processes are not significant. The possibility of a major change in the waste package environment caused by the multiplication of thermophilic bacteria has been raised. Copper and its alloys are usually resistant (although not entirely immune) to microbiological attack, probably because of the toxicity of copper compounds to lower life forms. To some extent the chemical effects of microbiological propagation can be evaluated by laboratory testing in simulated environments (for example, formation of sulfide by sulfate-reducing bacteria could be important for copper-bearing materials) if later Project analysis indicates that such microbiological entities could be introduced during the operational period and could survive in the thermal environment in Yucca Mountain. Galvanic effects will also be evaluated.

The rationale for choosing candidate copper-based alloys is described in Section 7.4.2.9.

### 8.3.5.9.2.3 Activity 1.4.2.3: Degradation modes affecting candidate austenitic container materials

This activity concerns the analysis to determine which degradation modes have a significant chance of occurring for the candidate austenitic materials in the postemplacement periods and laboratory testing activities to provide information for the modeling activities discussed in Information Need 1.4.3 (Section 8.3.5.9.3). The candidate austenitic materials are AISI types 304L and 316L stainless steels and the nickel-base austenitic alloy 825.

This activity consists of nine subactivities.

#### 8.3.5.9.2.3.1 Subactivity 1.4.2.3.1: Assessment of degradation modes in austenitic materials

Objectives

The objective of this subactivity is to evaluate the likelihood of each potential degradation mode to occur under conditions expected at Yucca Mountain.

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Parameters

The parameters for this subactivity are as follows:

1. Literature data documenting the causes for failure of this class of materials.

2. Interpretation of these causes of failure in the context of fabricating a container (Information Need 1.4.1, Section 8.3.5.9.1) and emplacing it in the Yucca Mountain repository (Information Need 1.5.3, Section 8.3.5.10.3).

Description

The fundamental feature in analyzing the behavior of the candidate austenitic materials is understanding that their oxidation and corrosion resistance depends on the formation and maintenance of a thin but protective passive film that slows down the reaction rate between the alloy and the environment. Mechanical or chemical processes that break down the passive film are responsible for initiation of degradation modes. Metallurgical reactions in the alloy fortify or weaken the stability of the passive film. Material on the analysis of potential degradation problems in the austenitic materials is given in Section 7.4.2.

Classification of degradation modes generally follows from the morphology of the attack (uniform, localized, stress-assisted, embrittlement) as indicated earlier in the material under the issue-level discussion. For the purpose of organizing the work in this and the next information need, the degradation modes have been placed into eight groups. This grouping is based in the performance models that are discussed in Information Need 1.4.3 (Section 8.3.5.9.3).

The analysis of this information need emphasizes the vulnerabilities of the materials, and much of the effort is directed toward establishing how much these vulnerabilities matter in demonstrating performance of the container. The three prominent vulnerabilities of the austenitic materials that are important in understanding the degradation modes in a variety of natural and chemical environments are (1) sensitivity to chloride ion in the environment, (2) tendency toward developing sensitized (chromium-depleted) microstructure, and (3) metallurgical metastability of austenite in the two candidate stainless steels. These vulnerability features influence the eight degradation modes around which the laboratory testing and modeling activities are centered.

The austenitic material degradation modes are the following:

1. Metallurgical aging and phase transformations.
2. Low temperature oxidation.
3. General aqueous corrosion.
4. Intergranular attack and intergranular stress corrosion cracking.
5. Hydrogen entry and embrittlement.
6. Pitting, crevice, and other localized attack.
7. Transgranular stress corrosion cracking.
8. Other potential degradation modes.

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As noted previously in the discussion of the copper-based material, the presentation order is only to facilitate the discussion of the important parameters for causing the particular degradation modes and does not indicate the importance of the particular mode. In summary, this activity reviews the pertinent literature on the different austenitic material degradation modes, as well as the relevance of previous Yucca Mountain Project laboratory work (Section 7.4.2). The activity assesses the potential for occurrence of each mode and estimates the severity of attack. All this information provides input to the container material selection (Activity 1.4.2.1, Section 8.3.5.9.2.1).

8.3.5.9.2.3.2 Subactivities 1.4.2.3.2 through 1.4.2.3.9: Laboratory test plan for austenitic materials

The following subactivities cover the laboratory test plans and testing program appropriate to each of the potential degradation modes just presented. These subactivities will be discussed as a group.

| Subactivity | Degradation mode |
|---|---|
| 1.4.2.3.2 | Metallurgical aging and phase transformations |
| 1.4.2.3.3 | Low temperature oxidation |
| 1.4.2.3.4 | General aqueous corrosion |
| 1.4.2.3.5 | Intergranular attack and intergranular stress corrosion cracking |
| 1.4.2.3.6 | Hydrogen entry and embrittlement |
| 1.4.2.3.7 | Pitting, crevice, and other localized attack |
| 1.4.2.3.8 | Transgranular stress corrosion cracking |
| 1.4.2.3.9 | Other potential degradation modes |

Subactivities 1.4.2.3.2 through 1.4.2.3.9 will be pursued (1) according to which material is selected for the advanced designs and (2) literature reviews and analyses that indicate the need to obtain data specific to Yucca Mountain conditions. The laboratory test plan will only be carried out in full on the material(s) selected for the advanced designs. The sequence of the major activities is given in the schedule and milestone section at the end of this information need.

Objectives

For the selected material, the objectives of this group of subactivities is to develop and implement a laboratory test plan to provide information to the modeling activities in Information Need 1.4.3 (Section 8.3.5.9.3). The

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test plan is oriented toward quantifying a particular degradation mode(s) or proving that the degradation mode(s) will not be operative under conditions anticipated at the Yucca Mountain repository.

Parameters

The expected important parameters for each of the degradation modes are listed under the respective activity for the modeling work in Information Need 1.4.3 (Section 8.3.5.9.3).

Description

The plan is to develop an experimental approach for each of the possible degradation modes. In many instances, the laboratory investigations are expected to be performed under environmental, metallurgical, or strain conditions that are intentionally made more severe than those expected to occur in the repository environment. This approach is used to accelerate the phenomenon under investigation so that measurement can be made in a reasonable amount of laboratory time (hours, days, weeks, months, and in some instances, up to a few years). Also, confidence in the modeling activities is gained by systematically extending the period of observation from shorter times with more aggressive conditions to making predictions for longer times with less aggressive conditions and then performing tests under these conditions for confirmation.

A technical review of the test plans and procedures will be conducted to assess the adequacy of the test conditions for the degradation modes. The review will be intended to ensure that the tests cover the range of conditions anticipated in the repository over the period of concern.

This approach requires sufficient understanding of the causative mechanisms for each of the degradation modes so that predictions for container failure can be made, as stated in the performance goals in this issue (Table 8.3.5.9-1) and consistent with the required confidence level (highest in the containment period, lower in the postcontainment period). In addition, several of the degradation modes are rather closely related to one another, and it is possible that one or more can be operable under a given set of conditions. For example, aging and transformation reactions can lead to phases (e.g., martensite) that are more susceptible to one of the degradation modes (hydrogen embrittlement) than the parent phase. Some theories of transgranular stress corrosion cracking in stainless steels ascribe crack initiation from the bottom of a previously formed pit. The long-term, low temperature oxidation is expected to condition the surface of the container and will influence all the other subsequent degradation modes. These points are also taken into account in the modeling activities.

As discussed in the previous section on degradation modes for the copper-based materials, "other potential degradation modes" covers the corrosion and mechanical degradation modes that appear to be inconsistent with the present understanding of conditions in the Yucca Mountain repository. Creep and high-temperature deformation and fracture mechanisms on the austenitic materials appear unlikely because of the higher strength (compared with copper) of these alloys and the relatively low temperatures that will develop in the near-package environment. Galvanic effects will also be evaluated.

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Another example in this category is the propagation of microbiological entities that could exist in the thermal environment after waste package emplacement and that could cause significant changes in the chemical nature of the environment. With regard to the corrosion of stainless steels, some combination of circumstances could lead to aggressive environmental conditions that could result in the formation of more acidic environmental conditions that would intensify pitting, crevice, stress corrosion, and possibly hydrogen enbrittlement if sulfuric acid-forming bacteria could be introduced during the repository operational period, if a sulfur-containing food source were available, and if the bacteria could survive the long thermal period after container emplacement. The nickel-based alloy is more resistant to acid attack (and concentration of anionic species that would also occur). Bacteria that use nitrogen or iron as food sources may also attack iron-based materials. To some extent laboratory testing can simulate the chemical effects of the environment modification by microbiological entities. Further analysis of whether the correct conditions for microbiological life forms would ever occur in Yucca Mountain will be evaluated before initiation of this work.

One of the reasons for exploring the use of alternative container materials and designs is the potential occurrence of exceedingly aggressive conditions (such as those discussed previously) so that any of the candidate metals could not be successfully demonstrated to withstand these conditions. This is discussed in the next activity.

### 8.3.5.9.2.4 Activity 1.4.2.4: Degradation modes affecting ceramic-metal, bimetallic/single metal, or coatings and filler systems

This activity concerns potential degradation modes that can affect an alternative waste package container developed under the alternate barriers investigations, and the testing studies needed to quantify and model these degradation phenomena. These degradation modes will apply to the post-emplacement periods and laboratory testing activities to provide information for the modeling activities discussed in Information Need 1.4.3 (Sec- tion 8.3.5.9.3).

#### 8.3.5.9.2.4.1 Subactivity 1.4.2.4.1: Assessment of degradation modes affecting ceramic-metal systems

Objectives

The first objective of this subactivity is to evaluate the potential degradation modes that are likely to occur in waste package containers fabricated from ceramic-metal systems under environmental conditions at Yucca Mountain. The second objective of this subactivity is to consider the experimental test programs needed to model and quantify these degradation phenomena. The ceramic-metal container may be pursued as an option if the technological feasibility study (Information Need 1.4.1, Section 8.3.5.9.1) indicates such an alternate package is feasible. The two ceramics, alumina ($Al_2O_3$) and titania ($TiO_2$), are initial candidates because of their excellent

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chemical stability in many aqueous environments. Other ceramic materials may be evaluated if they, too, are chemically resistant and if they meet the containment objectives.

Parameters

The parameters for this subactivity are

1. Literature data documenting the modes for failure of candidate ceramic-metal systems.

2. Interpretation of these causes of failure in the context of fabricating a waste package container (Information Need 1.4.1, Section 8.3.5.9.1) and emplacing it in the Yucca Mountain repository (Information Need 1.5.3, Section 8.3.5.10.3).

Description

At the present time, the degradation modes believed to be significant are (1) chemical dissolution of alumina and titania under repository environmental conditions and (2) delayed crack propagation and fracture driven by preexisting residual stresses at the time of emplacement and by postemplacement stresses.

In regard to corrosion, the Yucca Mountain repository, located more than 100 m above the water table, presents a relatively dry environment. Furthermore, only after several hundred years will the temperatures of most of the containers have declined sufficiently to allow water condensation and the possibility of continuous liquid contact with the container surface. This scenario presents a very challenging problem to corrosion considerations. Liquid and gas-phase reactions, as well as radiolysis effects at the container surface, must be taken into account.

Alumina ($Al_2O_3$) and titania ($TiO_2$) are believed to exhibit excellent corrosion resistance in aqueous environments when compared with other materials. Both of these compounds have been studied for waste containment application. Bulk corrosion rates on the order of $1 \times 10^{-1}$ to $1 \times 10^{-4}$ µm/yr have been reported and are very encouraging. However, special consideration must be given to the more localized corrosion of ceramic closures. Closure composition (metallic versus ceramic) and the possibility of localized stress-corrosion effects will be taken into consideration.

Fracture via delayed crack propagation under stress is believed to be a more limiting property of these ceramic materials than is bulk corrosion. There are two potential fracture sources to consider: (1) preexisting defects at the time of emplacement and (2) defects formed or extended by expanding corrosion products after emplacement.

It is important to emphasize that slow crack growth cannot occur without stress. Of paramount importance is an understanding of the nature and magnitude of residual stresses in the container at the time of emplacement and the stresses imparted by the repository after emplacement. All fabrication and closure methods under consideration will have to be modeled to quantify residual stresses. Once stress levels have been determined, corrosion con-

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tributions can be added and determinations can be made of the maximum allowable initial defect size to prevent failure over a given life span. We expect that corrosion factors will only apply in the case of tensile or shear stresses, and not in the case of compression.

For low levels of stress, generation of crack growth data for the long lifetimes required in this application will not be possible. Extrapolations will be required from the data base at higher levels of crack velocity. Additional data may be required. New mechanisms that could cause accelerated growth rates at lower stress must be considered. Proof testing to minimize delayed crack growth uncertainties will need to be considered.

After consideration of these fracture-mechanic studies, safety factors and maximum-permissible flaw sizes will be obtained. The plausibility of detecting flaws on the order of the maximum-permissible flaw size using current nondestructive characterization methods will be investigated.

### 8.3.5.9.2.4.2 Subactivity 1.4.2.4.2: Laboratory test plan for ceramic-metal systems of the alternate barriers investigations

The objective of this group of subactivities is to develop and implement a laboratory test plan to provide information for modeling activities in Information Need 1.4.3 (Section 8.3.5.9.3). The test plan is oriented toward quantifying particular degradation modes or showing that the degradation modes will not be operative under conditions anticipated at the Yucca Mountain repository.

The plan is to develop an experimental approach for each of the possible degradation modes. In many instances the laboratory investigations are expected to be performed under simulated environmental conditions that are intentionally made more severe than those expected to occur in the repository environment. This approach is used to accelerate the phenomenon under investigation so that measurement can be made in a reasonable amount of laboratory time. Confidence in the modeling activities is gained by systematically extending the period of observation from shorter times with more aggressive conditions to making predictions for longer times with less aggressive conditions, and then performing tests under these conditions for confirmation.

This approach requires an understanding of the mechanisms for each of the degradation modes so that predictions for container failure can be made, as stated in the performance goals in Issue 1.4 and consistent with the required confidence level (Table 8.3.5.9-1). It is further recognized that corrosion and delayed fracture are closely related, and it is possible that one or more can be operable under a given set of conditions.

In the category of "other potential degradation modes" particular corrosion and mechanical degradation processes are possible, but unlikely, based on the current understanding of conditions of Yucca Mountain.

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8.3.5.9.2.4.3 Subactivity 1.4.2.4.3: Assessment of degradation modes affecting bimetallic/single metal systems

Objectives

The objectives of this subactivity are to evaluate the likelihood of each potential degradation mode occurring under conditions anticipated at Yucca Mountain.

Parameters

The parameters for this subactivity are

1. Literature data documenting the causes for failure of this class of materials.

2. Interpretation of these causes of failure in the context of fabricating a container (Information Need 1.4.1, Section 8.3.5.9.1) and emplacing it in the Yucca Mountain repository (Information Need 1.5.3, Section 8.3.5.10.3).

Description

The reason for choosing a waste package container design concept that involves the use of two different metallic alloys is to be able to match the anticipated performance of the inner container and outer container to environmental conditions, as in the following:

1. Inner container -- The material will be chosen for long-term microstructural and mechanical stability and corrosion resistance at lower temperatures where significant quantities of liquid water are possible.

2. Outer container -- The material will be chosen for shorter-term microstructural and mechanical stability and corrosion resistance at higher temperatures and gamma radiation fields.

The inner container material will be chosen to be cathodic to the outer container material for additional corrosion protection and additional assistance in meeting performance goals.

Since neither materials for the inner container nor the outer container have been chosen at this time, descriptions of detailed analyses for possible degradation modes of any potentially useful bimetallic combination are premature. However, it is appropriate to point out that, in addition to any degradation modes inherent to the candidate materials as "single metal barriers" (discussed in Subactivities 1.4.2.2.1 and 1.4.2.3.1), the following factors unique to operation of bimetallic (or ceramic-metal) systems will be carefully analyzed for any specific choice of material couple:

1. The "corrosion potentials" of the metals, $M_A$ and $M_B$, forming the couple under conditions anticipated in the Yucca Mountain repository.

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2. The nature and kinetics of the cathodic reaction at the surface of the more electropositive metal, and the nature and kinetics of the anodic reaction at the surface of the more electronegative metal.

3. Operation of the catchment-area principle, which involves the direct proportionality of the galvanic corrosion rate to the area of the cathodic metal, under conditions of the galvanic current being limited to the diffusion rate of dissolved oxygen to the cathode.

4. The nature and conductivity of the impurity-laden water in contact with the waste package container.

The reason for choosing a waste package container design concept that involves the use of an alternate single metallic alloy, such as a nickel-based corrosion-resistant alloy, or one of the newer duplex or "super ferritic" stainless steels, is to have available a fully characterized material system that will resist much higher levels of water, a more aggressive water chemistry, and higher mechanical loads than are presently anticipated.

The fundamental feature in analyzing the behavior of candidate alternate single materials is understanding that their oxidation and corrosion resistance depends on the formation and maintenance of a thin but protective passive film that slows down the reaction rate between the alloy and the environment. Mechanical or chemical processes that break down the passive film are responsible for initiation of degradation modes. Metallurgical reactions in the alloy fortify or weaken the stability of the passive film.

Classification of degradation modes generally follows from the morphology of the attack (uniform, localized, stress-assisted, embrittlement). For the purpose of organizing the work in this and the next information need, the degradation modes have been placed into groups. This grouping is based on the performance models that are discussed in Information Need 1.4.3 (Section 8.3.5.9.3).

The plan is to develop an experimental approach for each of the possible degradation modes. Laboratory investigations will be performed under environmental, metallurgical, or strain conditions that are intentionally made more severe than those expected to occur in the repository environment, to accelerate the phenomenon under investigation so that measurement can be made in a reasonable amount of laboratory time. Confidence in the modeling activities is gained by systematically extending the period of observation from shorter times with more aggressive conditions to making predictions for longer times with less aggressive conditions and then performing tests under these conditions for confirmation.

### 8.3.5.9.2.4.4 Subactivity 1.4.2.4.4: Laboratory test plan for bimetallic/single metal material systems

The following subactivities cover the laboratory test plans and testing program appropriate to each of the potential degradation modes. These subactivities are discussed as a group.

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| Subactivity | Degradation mode |
|---|---|
| 1.4.2.4.4.1 | Metallurgical aging and phase transformations in base metals, heat-affected zones, and welds |
| 1.4.2.4.4.2 | Low temperature oxidation |
| 1.4.2.4.4.3 | General aqueous corrosion |
| 1.4.2.4.4.4 | Intergranular attack and stress corrosion cracking |
| 1.4.2.4.4.5 | Hydrogen entry and embrittlement |
| 1.4.2.4.4.6 | Pitting, crevice, and other localized attack |
| 1.4.2.4.4.7 | Gamma flux effects |
| 1.4.2.4.4.8 | Galvanic effects at welds, oxide-inclusions, and surface oxides |

Each of these subactivities will be pursued according to (1) the specific choices of metallic materials selected for the advanced designs and (2) literature reviews and analyses that indicate the need to obtain data specific to Yucca Mountain conditions. The laboratory test plan will only be carried out in full on the material(s) selected for the advanced designs. The sequence of the major activities is being developed.

Objectives

For the selected material, the objectives of this group of subactivities are to develop and implement a laboratory test plan to provide information to the modeling activities in Information Need 1.4.3 (Section 8.3.5.9.3). The test plan is oriented toward quantifying a particular degradation mode(s) or proving that the degradation mode(s) will not be operative under conditions anticipated at the Yucca Mountain repository.

Parameters

The expected important parameters for each of the degradation modes will be developed once choices of specific metallic materials are made.

Description

The plan is to develop an experimental approach for each of the possible degradation modes. The laboratory investigations are expected to be performed under environmental, metallurgical, or strain conditions that are made more severe than those expected in the repository environment to accelerate the phenomenon under investigation so that measurement can be made in a reasonable amount of laboratory time. Confidence in the modeling activities is gained by systematically extending the period of observation from shorter times with more aggressive conditions to making predictions for longer times with less aggressive conditions and then performing tests under these conditions for confirmation.

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This approach requires sufficient understanding of the causative mechanisms for each of the degradation modes so that predictions for container failure can be made, consistent with the required confidence level (highest in the containment period, lower in the postcontainment period). In addition, several of the degradation modes are rather closely related to one another, and it is possible that one or more can be operable under a given set of conditions.

### 8.3.5.9.2.4.5 Subactivity 1.4.2.4.5: Assessment of degradation modes in coatings and filler systems

Coatings are expected to be either ceramic or metallic. Their primary purpose would be to substantially enhance the corrosion resistance of metal containers. We do not expect their application to the ceramic-metal systems. Coatings must demonstrate closed porosity and long-term stability in regard to substrate adherence, resistance to cracking, and corrosion resistance.

Fillers (metal or nonmetallic) will provide added mechanical support between spent fuel elements and the container. Thermal properties must be compatible with waste package design regarding maximum waste-form temperature. Fillers should also provide additional long-term protection (post 1,000-yr performance) against corrosion and control the release of radionuclides.

### 8.3.5.9.2.4.6 Subactivity 1.4.2.4.6: Laboratory test plan for coatings and filler systems of the alternate barriers investigations

The following subactivities cover the laboratory test plans and testing program appropriate to each of the potential degradation modes. These subactivities are discussed as a group.

| Subactivity | Degradation mode |
|---|---|
| 1.4.2.4.6.1 | Low temperature oxidation |
| 1.4.2.4.6.2 | Metallurgical stability and toughness under repository conditions |
| 1.4.2.4.6.3 | General aqueous corrosion |
| 1.4.2.4.6.4 | Hydrogen entry and embrittlement |
| 1.4.2.4.6.5 | Gamma flux effects |
| 1.4.2.4.6.6 | Mechanical degradation |
| 1.4.2.4.6.7 | Galvanic corrosion |
| 1.4.2.4.6.8 | Localized corrosion |

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These subactivities will be pursued according to (1) which coating or filler is selected for the advanced designs and (2) literature reviews and analyses that indicate the need to obtain data specific to Yucca Mountain conditions. The laboratory test plan will only be carried out in full on the material(s) selected for the advanced designs. The sequence of the major activities is given in the schedule and milestone section at the end of this information need.

Objectives

For the selected material, the objectives of this group of subactivities are to develop and implement a laboratory test plan to provide information to the modeling activities in Information Need 1.4.3 (Section 8.3.5.9.3). The test plan is oriented toward quantifying a particular degradation mode(s) or proving that the degradation mode(s) will not be operative under conditions anticipated at the Yucca Mountain repository.

Parameters

The expected important parameters for each of the degradation modes will be developed once choices of specific materials are made.

Description

The plan is to develop an experimental approach for each of the possible degradation modes. The laboratory investigations are expected to be performed under environmental, metallurgical, or strain conditions that are more severe than those expected to occur in the repository environment. This approach is used to accelerate the phenomenon under investigation so that measurement can be made in a reasonable amount of laboratory time. Confidence in the modeling activities is gained by systematically extending the period of observation from shorter times with more aggressive conditions to making predictions for longer times with less aggressive conditions and then performing tests under these conditions for confirmation.

This approach requires sufficient understanding of the causative mechanisms for each of the degradation modes so that predictions for container failure can be made, consistent with the required confidence level (highest in the containment period, lower in the postcontainment period). In addition, several of the degradation modes are rather closely related to one another, and it is possible that one or more can be operable under a given set of conditions.

## 8.3.5.9.3 Information Need 1.4.3: Scenarios and models needed to predict the rate of degradation of the container material

Technical basis for addressing the information need

This information need combines the scenarios and conditions for the near-field provided by Information Need 1.5.3 (Section 8.3.5.10.3) and the performance of metal materials under a range of conditions provided by Information Need 1.4.2 (Section 8.3.5.9.2). Behavior of weld metal and weld heat-affected zones will be considered in addition to the base metal. The models developed here, together with data developed in Information Needs 1.4.1 and 1.4.2 (Sections 8.3.5.9.1 and 8.3.5.9.2), will be used to predict the performance of the container during both the containment period and the postcontainment period.

Deterministic models linked to the relevant degradation modes will be developed for the selected specific container material for advanced design work. The modeling activities discussed in Section 7.4.5.4.6 and in the remainder of this section will be based on physical, chemical, metallurgical, and mechanical parameters covering the range of expected repository conditions.

Link to the technical data chapters and applicable support documents

The scenarios and conditions for container degradation are derived from the information onsite geology (Chapter 1), hydrology (Chapter 3), geochemistry (Chapter 4), emplacement environment (Section 7.1), waste package design (Section 7.3), and waste package postemplacement environment (Section 7.4.1). Some of the scenarios requiring analysis will arise from information needs of the total system performance assessment (Issue 1.1), which is discussed in Section 8.3.5.13.

Performance assessment models that will be used to predict metal barrier performance are discussed in Section 7.4.5. Design inputs to those analyses appear in Section 7.3. Details of activities that will develop waste package process models that will be implemented in performance assessment modeling appear in waste package environment (Section 7.4.1), metal barrier studies (Section 7.4.2), and geochemical modeling (Section 7.4.4). Further details are provided in information needs under Issues 1.5 (Section 8.3.5.10) and 1.10 (Section 8.3.4.2).

Parameters

The information needed from other information needs includes

1. Scenarios developed under Information Need 1.5.3 (Section 8.3.5.10.3) to describe the waste package near-field environment before container failure.

2. Results of geochemical modeling calculations from Information Need 1.5.3 to characterize the chemical composition and speciation of the solutions that might contact the container.

3. The container design characteristics from Information Need 1.4.1 (Section 8.3.5.9.1)

4. The container material properties from Information Need 1.4.2 (Section 8.3.5.9.2).

The output parameters for container performance models are tools that will allow the performance of the container to be predicted under repository postemplacement conditions.

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The scenarios developed under Information Need 1.5.3 (Section 8.3.5.10.3) will define the range of conditions that correspond to the anticipated processes and events for 10,000 and 100,000 yr. The models developed here will be used in combination with the waste package performance assessment code to provide the predictions of the conditions of the container for the first 1,000 yr after repository closure and for the postcontainment period. This will provide the information needed to calculate potential releases from waste packages during the containment period and thereafter. The parameters used in the performance assessment calculations will contain probabilistic information.

Logic

Prediction of the long-term performance of the metal barrier under repository conditions requires that all significant degradation mechanisms be identified and the probability of their occurrence be quantified. For all degradation modes that might be significant, a physical-chemical model must be developed that will allow extrapolation of data gathered in the laboratory to the times and conditions relevant to the repository. In many instances, the analysis to determine whether the degradation mode might occur requires the same model that will allow prediction of long-term behavior. Thus, in this information need, activities are included that both assess the relevance of particular degradation processes and develop models to describe their action under repository conditions. The tools developed under this information need will be used in Information Need 1.4.4 (Section 8.3.5.9.4) to predict the condition of the containers as a function of time for both anticipated processes and events and for other, low probability cases for which source term data is requested by the total system performance assessment task.

There are three activities in this information need. The first covers the investigation of copper-based materials, while the second covers the investigation of the austenitic materials. After alloy selection, only one of these activities will continue, and only one (or at most two) material will be the subject of intensive study. Other members of the alloy family may be included in testing activities if they provide insight into the behavior of the candidate materials. The third activity concerns models to predict the performance of an alternative material system.

The modeling activities discussed in this information need and the laboratory testing activities discussed in Information Need 1.4.2 (Section 8.3.5.9.2) are closely related. They are both described in fairly general terms in this document with much greater detail to be provided in the laboratory test plan that will be written for the material(s) selected for the advanced designs. Particularly for localized corrosion and stress corrosion cracking there is a considerable need to select test methods as well as materials, and this selection is best left until after the final material is selected. The sequence of activities is indicated in the schedule and milestone section at the end of this information need.

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### 8.3.5.9.3.1 Activity 1.4.3.1: Models for copper and copper alloy degradation

Seven subactivities support this evaluation.

#### 8.3.5.9.3.1.1 Subactivity 1.4.3.1.1: Metallurgical aging and phase stability

Objectives

This subactivity will examine the kinetics of segregation effects in the high-purity oxygen-free copper (CDA 102) and the segregation and possible precipitation kinetics in the candidate alloys CDA 613 and CDA 715. The objective is to determine whether any significant segregation or precipitation of secondary phases could occur under disposal conditions; if they occur, to what extent; and what the consequences of these reactions are on induced embrittlement or enhanced susceptibility of the metal to corrosion processes.

Parameters

Information needed from other information needs includes

1. Description of the near-field waste package environment (especially the projections of time-temperature profiles).

2. Laboratory data on the kinetics of phase segregation reactions.

3. Mechanical properties of the segregation products.

4. Electrochemical effect of segregation products on the base metal.

5. Strain in the container body material and in the heat-affected zone around the closure.

6. Residual stress.

The output parameters are the prediction of the phases that might be present in the metal container and the abundance of those phases as a function of time and repository conditions.

Description

In general, there are fewer considerations (compared with those for the other candidate materials) in the modeling of the long-term behavior of the copper systems because of the simple structure of the materials. Copper has no phase transformations, and high-purity copper has no intentional alloy constituents. The main concerns are (1) the possibility of segregation in the copper alloys over long periods of time and the effects of this on corrosion performance and (2) the precipitation of minor alloy constituents, such as iron in CDA 715 and tin in CDA 613, and their effects on corrosion and embrittlement. The aluminum content in CDA 613 approaches the solubility limit, and the effect of other alloy constituents may favor the precipitation of second phases in this alloy under some conditions.

This activity will first assess the possibility for alloy constituent precipitation and segregation in the alloys. If any of these separation effects are found to be likely, then an appropriate nucleation or diffusion-based model for the separation will be developed. Results from this model will then be used with the models for other degradation modes (such as pitting corrosion and stress corrosion) to assess the potential for container degradation. For high-purity copper it may be necessary to model the low temperature creep of the material because of the comparatively low strength of pure copper. Although the waste package will not be under large static loads in the environment expected at Yucca Mountain, the thicker walls considered for a pure copper container may create sufficient self loading to allow significant low temperature creep over very long times. Again, the first step will be to assess the need for the model and, if necessary, develop the model. It may be advantageous to add a small amount of deoxidizer (e.g., phosphorus, beryllium, aluminum, chromium, and rare earth elements) to the high-purity copper to prevent oxygen pickup during hot working or welding. In this instance, a model for the long-term effect of the deoxidizing element in the metallurgical microstructure may be needed.

### 8.3.5.9.3.1.2 Subactivity 1.4.3.1.2: Low-temperature oxidation

Objectives

The objectives of this subactivity are to (1) determine the amount of metal loss by oxidation and the rate law explaining the oxidation behavior of the copper-based material over the relevant times and temperatures for the repository and (2) characterize the oxide or other protective layer formed.

Parameters

The information needed from other information needs includes

1. Results of weight loss or gain tests under relevant time-temperature conditions.

2. Description of the container environment.

3. Description of oxidation product layers.

4. Effect of radiation on moist air.

The output parameters are rate laws for the degradation of the metal by oxidation and a model for predicting the behavior of oxide layers under repository conditions. Occasionally, depending on environmental species present, other anionic species are incorporated into the oxidation product, so that a basic copper nitrate, basic copper carbonate, basic copper chloride, or basic copper sulfate is found in the oxidation product layers.

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Description

Tests will be conducted under Information Need 1.4.2 (Section 8.3.5.9.2) to determine the rates of oxidation over the temperature range of interest. These data will be used to develop a model for the oxidation process under Yucca Mountain conditions. Of particular concern with copper and copper-based alloys is the rate of oxidation that will occur in the time period just after emplacement when both the temperature and the radiation dose rate is highest. Radiolysis of the expected moist atmosphere can produce oxides of nitrogen that could cause high oxidation rates and formation of nonprotective oxides. The limited amount of testing performed in a high gamma radiation field thus far (discussed in Section 7.4.2) does not indicate excessive oxidation rates.

Oxidation studies performed on copper and copper-based alloys at temperatures generally less than 300°C (low-temperature oxidation) indicate that the oxide growth kinetics follow a cubic (or higher order) rate law. The oxide layer is dominantly $Cu_2O$. No indications of spilling or exfoliation of the oxide are given. Very little information on oxidation in the presence of gamma radiation is available.

The main work in the oxidation studies will most probably involve characterizing the properties of the oxide that would develop on the container surface during the long period when the surface temperature is above the boiling point of water and the environment is relatively dry. This oxide film then establishes the surface characteristics of the metal when the temperature has cooled enough that liquid water can enter the near-package environment.

8.3.5.9.3.1.3 Subactivity 1.4.3.1.3: General aqueous corrosion

Objectives

The objective of this subactivity is to determine the amount of metal loss by general aqueous corrosion and to establish whether a uniform pattern of attack occurs. Aqueous corrosion can occur when a more or less continuous moisture film is present on the container surface or when some portion of the container surface is immersed in water.

Parameters

The information needed from other information needs includes

1. Results of weight loss tests.
2. Description of the environment near the waste package surface.
3. Description of corrosion product layers.
4. Chemical modeling of solution composition.
5. Radiolysis effects in aqueous media.

The output parameters are estimates of the wastage of the metal container that can occur during the containment and postcontainment periods. The Project would like to be able to characterize both oxidation and general

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aqueous corrosion well beyond the thousand-year postclosure period. Most of the container surface will still be present in this period, and could thus provide a catchment location for water. The controlled release rate models for radioactive nuclides will depend on the potential of the container to affect water movement to and from the waste form.

Description

Many of the same points of discussion made on the low-temperature oxidation of copper and copper-based alloys apply to the discussion on general aqueous corrosion. Indeed, it is difficult to draw a hard line between oxidation and corrosion, and from the point of view of model development, many of the same features will be found in both phenomena. An important link between corrosion and oxidation is development of a thin-film electrolyte model where the "dry" oxidation case is given by the limit of a zero thickness film. Because an electrolyte is present in the aqueous corrosion case, the model is amenable to experimental verification by measurements of corrosion potentials and corrosion currents.

The characterization of the corrosion product layers in general aqueous corrosion is also important to establish whether the patinas formed on a corroding copper surface are protective. In addition, the oxide (including whatever anionic species may be incorporated with it) characteristics (e.g., composition, thickness, and defect structure) govern its behavior with regard to models for localized corrosion and stress corrosion. Models for these nonuniform kinds of corrosion will include the treatment of the breakdown and repair of protective films or layers on the metal surface.

Tests will be conducted under Information Need 1.4.2 (Section 8.3.5.9.2) to determine the rates of general corrosion over the range of temperature and water composition that could be expected in the repository. These data will be used to develop a model for the corrosion process under Yucca Mountain conditions. The general features of the model will include

1. Prediction of the oxidation-reduction potential in the environments of interest.

2. Prediction of the corrosion potential for the metal in the environments of interest.

3. Prediction of the corrosion current (and hence the corrosion rate) as a function of potential.

The oxidation-reduction potential is a measure of the oxidizing or reducing nature of the environment, and the corrosion potential is a measure of the response of the metal to the environmental oxidation-reduction potential. This model will establish boundaries for the possible range of corrosion potentials as a function of temperature and the nature and concentration of chemical species in the water (including effects of pH and dissolved atmospheric gases). Also, the residual effect of radiolysis in the environment will be considered. By the time the temperature permits liquid water in the near-package environment, the radiation field is expected to have decayed to a level at which radiolysis effects are small. The model for corrosion potentials will also be related to models being developed for localized cor-

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rosion, hydrogen embrittlement, and stress corrosion. Prediction of corrosion susceptibility depends on the values of the critical potentials required to initiate and propagate these kinds of corrosion relative to the value of the corrosion potential.

8.3.5.9.3.1.4 Subactivity 1.4.3.1.4: Hydrogen entry and embrittlement

Objectives

The objective of this subactivity is to assess what level of hydrogen in copper-based materials is necessary to cause embrittlement of the material and to significantly affect other degradation rates and mechanisms. The subactivity will then examine the environmental conditions at Yucca Mountain to determine whether that amount of hydrogen could conceivably enter the metal structure. If the required hydrogen would be available, the necessary laboratory studies will be conducted under Information Need 1.4.2 and a model developed in this subactivity to determine the effects of hydrogen embrittlement.

Parameters

The information needed from other information needs includes

1. Hydrogen production rate by radiolysis and corrosion.
2. Hydrogen recombination rate by all processes.
3. Maximum rate of hydrogen entry into the alloy.
4. Maximum concentration of hydrogen in the alloy.
5. Phase structure of the alloy.
6. Effects of hydrogen in copper-based materials.

Description

The assessment of hydrogen effects centers around a bounding calculation for the maximum availability of atomic hydrogen at the metal surface. (Molecular hydrogen does not diffuse into the metal.) The analysis will consider both the external and internal container environments. The latter is necessary because some fuel rods that breached in reactor service may contain water that would be released to the container inner atmosphere under disposal conditions.

The model will consider the maximum rate of hydrogen permeation in the metal (i.e., the net result of hydrogen entry and loss by outward diffusion). The total trapped hydrogen will be compared with the level that produces significant effects on the container material performance under Yucca Mountain conditions. If the amount of trapped hydrogen is less than the critical level, no further work will be done. If the amount is greater, the effects of the hydrogen will be assessed. An early determination concerning the probability for embrittlement of copper should be possible and no further work will be needed.

One particular effect that occurs in high-purity copper is that of "hydrogen sickness." This is caused by the copper picking up oxygen during a

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hot forming or welding operation. The oxygen forms oxides in the copper that are unstable in the presence of a hydrogen-containing environment. The result is formation of water vapor blisters in the copper. Addition of a small amount of deoxidizing element (e.g., aluminum, phosphorus, beryllium, chromium, and rare earths) to the copper appears to prevent hydrogen sickness.

### 8.3.5.9.3.1.5 Subactivity 1.4.3.1.5: Pitting, crevice, and other localized attack

Objectives

The objective of this subactivity is to determine whether the necessary environmental conditions will exist to initiate pitting, crevice, or other localized corrosion attack under Yucca Mountain repository conditions. If pitting or crevice corrosion were predicted to occur, then the rate of propagation of the attack would be determined. Another kind of localized attack that is specific to some copper-based alloys is selective leaching of the less noble constituent (aluminum from aluminum bronze, nickel from copper-nickel). Therefore, this activity will assess whether selective leaching could occur in the repository environment.

Parameters

The information needed from other information needs includes

1. Near-field waste package environment conditions, especially the concentration of ions known to favor these modes of attack.
2. Quantities of electrolyte needed to set up localized corrosion cells.
3. Temperature.
4. Solution pH.
5. Metal microstructure.
6. Corrosion potential.
7. Pitting (and other critical potentials).

Description

Pitting attack occurs when the temperature and aggressive ion concentrations are sufficiently high and the pH sufficiently low to cause localized corrosion cells to initiate and propagate on the metal surface. The metal microstructure can also be important because it can lead to local breakdown of the passive corrosion films and to the establishment of galvanic cells. Precipitates and inclusions can be particularly important in favoring pitting corrosion. The ions of concern for copper and its alloys are sulfide and certain heavy metal ions (e.g., ferric and manganese). These ions are not

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present in the waters beneath Yucca Mountain in significant quantities, and they are not expected to be present in the vadose water at levels great enough to cause concern. (These species could possibly be introduced during the repository construction and operational periods.) Metallurgical effects on localized corrosion initiation will be assessed; these include inclusions in the metal, precipitation reactions in the metal, and segregation reactions.

The model for pitting corrosion will determine critical values for the electrochemical potential above which pitting occurs and will determine whether this potential could be reached in the system under anticipated Yucca Mountain conditions.

Crevice corrosion is not commonly observed in copper and copper-based alloys, but a full assessment of whether it can occur under repository conditions will be undertaken. Models for crevice corrosion will use critical potential analysis combined with an analysis for the potential for propagation of the crevice attack. The latter analysis will use the crevice geometry and the local chemical conditions as its basis. The data for this model development will be collected in activities described under Information Need 1.4.2 (Section 8.3.5.9.2).

Selective leaching effects are possibly tied to the segregation effects in alloys or to codissolution of both the copper and other alloy constituents with later redeposition of the copper as a sponge-like material. Selective leaching effects are most commonly associated with copper-zinc alloys; of the candidate materials, the aluminum bronze would appear to have the greatest susceptibility because of the large electrochemical potential between copper and aluminum. However, the expected oxidizing conditions in the repository would be expected to passivate the alloy and mitigate against selective leaching. This will need to be demonstrated. Severe metallurgical or environmental inhomogeneity could conceivably initiate and drive a selective leaching reaction. Selective leaching effects are also potential dependent, and so a model for this kind of localized attack will be based on analyses of critical potentials for initiating and propagating the phenomenon.

The probability of localized forms of corrosion appears to be of lesser concern than other corrosion and degradation mechanisms for copper-based materials. Any modeling activities undertaken for these materials will determine the critical potential over a wide range of environmental conditions and alloy compositions, and relate those potentials to the expected range of conditions for the repository and for the as-assembled container. Successful validation of the model in water with relatively high ionic contents will add confidence to the extrapolations needed to reach the expected repository conditions of low ionic contents.

### 8.3.5.9.3.1.6 Subactivity 1.4.3.1.6: Stress corrosion cracking

<u>Objectives</u>

The objective of this subactivity is to determine the potential for stress corrosion cracking to occur under the repository disposal conditions,

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and if it occurs, to provide a prediction for the rate of crack initiation and growth.

Parameters

The information needed from other information needs includes

1. Ammonia concentrations that could contact the container.
2. Temperature.
3. Stress (and stress intensity).
4. Alloy segregations.
5. Other ions in solutions.
6. Corrosion potential.
7. Critical potential for crack initiation.

Description

By far the most important documented failures and research investigations on stress corrosion cracking (SCC) of copper and its alloys are in ammonia and ammonia-containing environments. Ammonia (and ammonium ion, and in some instances, organic compounds that decompose to form ammonia) form highly soluble complexes with copper. These complexes destabilize the otherwise protective patinas on copper in most environments and create very active sites where the stressed protective layer is broken and rapid anodic dissolution occurs to initiate the crack. Ammonia is effective in initiating SCC in the most susceptible materials (brasses) at small concentrations. There are possible occurrences for ammonia formation in the waste package environment. For example, radiolysis of atmospheric gases ($N_2$ and $H_2O$) could produce $NH_3$. Although the dominant oxidizing conditions are thought to mitigate against significant ammonia formation, ammonia could form as a transient species and be present on the container surface in sufficient amounts and for sufficient times to initiate cracking. Experimental determination of these critical concentrations and times can be compared with calculations of the radiolysis reaction yield rates for ammonia.

Ammonia could also possibly form inside those waste package containers containing water-logged spent fuel. Even though the spent fuel water packages will be backfilled with argon, nitrogen will be present as an impurity, and irradiation of the internal atmosphere can produce ammonia, particularly since the absence of oxidizing conditions will favor a longer residence time or higher concentrations of ammonia.

The usual stress corrosion crack propagation mode is transgranular, but occasionally an intergranular path is observed. Oxygen (or other oxidizing species) in conjunction with ammonia also appears to be necessary for crack formation and likely influences the crack path. Segregation effects in the alloys (particularly those at grain boundaries) would influence the crack propagation path, as will the stress (or stress intensity) to maintain crack growth. Both high-purity copper and aluminum bronze are quite susceptible to ammonia-induced SCC; copper-nickel is more resistant but not immune to SCC caused by ammonia.

Besides ammonia, other chemical species have occasionally been implicated in causing SCC in copper and some of its alloys. Nitrite ion has been reported to cause SCC in pure copper; the vadose water associated with Yucca Mountain naturally contains nitrate ion and radiolysis of atmospheric nitrogen may produce various oxides of nitrogen. In the presence of a metal container (as a reducing agent), some amount of nitrite ion is likely to be produced.

Many of these important environmental and metallurgical parameters can be expressed in terms of critical electrochemical potentials that would correspond to SCC initiation and propagation, and a model for SCC in copper and the candidate alloys would logically begin with determination of these critical potentials in ammonia-containing environments and possibly in other environments free of ammonia.

The addition of tin (in the approximate 0.2 to 0.5 percent range) to commercial aluminum bronzes is important to prevent SCC in steam environments. The CDA 613 and 614 grades contain tin in this range.

There has been one reported occurrence of intergranular cracking of a laboratory heat of pure 70/30 copper-nickel in a high temperature steam environment (≈300°C). This might have been caused by the absence of alloy additions (especially iron) that are present in the commercial version of the alloy. This occurrence will be investigated and assessed. The role of small alloying additions may need to be investigated further if one of the copper alloys is selected for advance designs to ensure understanding of how these additions work.

8.3.5.9.3.1.7 Subactivity 1.4.3.1.7: Other potential degradation modes

This subactivity will screen other potential degradation modes not discussed previously to determine whether there is a cumulative probability of occurrence greater than 0.01 over the time interval of interest. If the probability exceeds that level, a model will be developed for the corrosion or degradation mode. Examples of models to be screened are mechanical fracture (e.g., low temperature creep) and the effect of microbiological activity on the previously discussed corrosion mechanisms. Another model for a possible degradation mode involves galvanic interaction of the metal container with other metallic components in the engineered barrier system and affiliated repository components. As mentioned in Section 8.3.4.2, the borehole liner and container material are proposed to be made from materials in the same alloy family to minimize galvanic effects.

8.3.5.9.3.2 Activity 1.4.3.2: Models for austenitic material degradation

The following eight subactivities support this evaluation.

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8.3.5.9.3.2.1 Subactivity 1.4.3.2.1: Metallurgical aging and phase transformations

Objectives

This subactivity will examine the kinetics of phase transformations in the austenitic materials AISI types 304L, 316L, and alloy 825. The objective is to determine (1) whether phase transformations occur under disposal conditions; (2) if they occur, to what extent; and (3) the consequences of these phase transformations on the susceptibility of the metal to degradation by other processes.

Parameters

The information needed from other information needs includes

1. Description of the near-field waste package environment (especially the projections of time-temperature profiles).

2. Laboratory data on the kinetics of phase transformation reactions.

3. Mechanical properties of the transformation products.

4. Alloy composition of the base metal and the weld metal.

5. Strain in the container body material and in the heat affected zone around the closure.

6. Residual stress.

The output parameters are the prediction of the phases that might be present in the metal container and the abundance of those phases as a function of time and repository conditions.

Description

This subactivity will address the concern that metastability in some of the austenitic materials, particularly in types 304L and 316L, might lead to the production of brittle phases that can significantly degrade the mechanical properties of the material during the containment period. Alloy 825 is considered a stable alloy; no phase transformations should occur. (However, some precipitation reactions will occur in this alloy; these are usually thought to be beneficial (i.e., formation of tic rather than chromium-rich $M_{23}C_6$). The long-term effect of possible intergranular reactions involving aluminum, titanium, molybdenum, and other alloying elements in this material will need to be investigated.) Changes in mechanical properties could affect preclosure considerations such as ability to retrieve the waste packages. In the postclosure period, changes in mechanical properties are only of concern if they result in changes in the degradation rate of the container material by other processes. This is true because the waste packages will not be subjected to large static or dynamic loads under anticipated conditions at Yucca Mountain.

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The model to be developed will address the issue of whether the long times at elevated temperature change the microstructure of the metal to the extent that the corrosion and oxidation behavior of the material is changed. Some examples of the consideration are the effect of martensite on hydrogen embrittlement (especially in type 304L), the effect of sigma phase on enhancing intergranular attack (especially in type 316L), and the effect of possible intergranular precipitates in alloy 825. The basic features of the model to be developed include the following:

1. The kinetics of the phase transformations with time due to combined effects of radiation, temperature, stress, alloy composition, and initial metallurgical structure.

2. The change in mechanical properties as a result of transformations.

Changes in corrosion performance resulting from phase transformations will be modeled under the applicable degradation mode.

The transformations to be considered are as follows:

1. Austenite to martensite (especially strain induced).
2. Austenite to ferrite.
3. Austenite to ferrite to sigma.
4. Austenite to sigma.
5. Austenite to other brittle phases (chi, Laves).
6. Austenite to intergranular precipitates (especially in alloy 825).

The transformations to sigma, ferrite, chi, an Laves are nucleation and growth reactions that will be modeled by diffusional processes. The transformation to martensite is diffusionless and will be modeled by critical temperature analysis for the start and end of the reaction.

### 8.3.5.9.3.2.2 Subactivity 1.4.3.2.2: Low-temperature oxidation

#### Objectives

The objectives of this subactivity are to determine the amount of metal loss by oxidation and the kinetics of metal oxidation and to characterize the properties of the protective films and the aging of the films with long times at the repository temperatures.

#### Parameters

The information needed from other information needs includes

1. Results of weight loss or gain tests under relevant conditions.
2. Description for the container environment.
3. Description of oxidation product layers.
4. Effect of radiation on the atmosphere surrounding the waste package.

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The output parameters are rate laws for the degradation of the metal by oxidation and a model for the behavior of passivating oxidation product layers under repository conditions.

Description

Tests will be conducted under Information Need 1.4.2 (Section 8.3.5.9.2) to determine the rates of oxidation under repository relevant temperature, environmental, and radiation dose rate conditions. Because of the low rates expected, the oxidation rate is not expected to be a degradation mode that will cause breach of the container in 1,000 yr. Characterization of the oxidation product layers is important in establishing the conditions that will prevail on the container surface at a time when water can intrude into the waste package environment, wet the surface, and allow various aqueous corrosion processes to occur.

As discussed in the parallel activity for the copper-based materials, a model for oxidation (and for general aqueous corrosion, described in the next section) will be developed.

8.3.5.9.3.2.3 Subactivity 1.4.3.2.3: General aqueous corrosion

Objectives

The objective of this subactivity is to determine the amount of metal loss by general aqueous corrosion and to establish whether a uniform pattern of attack occurs. Aqueous corrosion can occur when a more or less continuous moisture film is present on the container surface or when some portion of the container surface is immersed in water.

Parameters

The information needed from other information needs includes

1. Results of weight loss tests.
2. Description of the environment near the waste package surface.
3. Description of corrosion product layers.
4. Chemical modeling of solution composition.
5. Radiolysis effects in aqueous media.

The output parameters are estimates of the wastage of the metal container that can occur during the containment and postcontainment periods. The Project would like to be able to characterize both oxidation and general aqueous corrosion well beyond the thousand-year postclosure period. Most of the container surface will still be present in this period, and could thus provide a catchment location for water. The controlled release rate models for radioactive nuclides will depend on the potential for the container to affect water movement to and from the waste form.

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Description

Much of the discussion on general corrosion of copper-based materials applies to the discussion on austenitic materials with respect to data acquisition and model development. General aqueous corrosion is not expected to be a container failure mode during (and well beyond) the containment period, but characterization of the corrosion behavior and passive films formed on these materials are of interest in the models being developed for the different kinds of localized corrosion and stress corrosion discussed in the next several sections.

The general features of the model will include the following:

1. Prediction of the oxidation-reduction potential of the environment.

2. Prediction of the corrosion potential for the metal in the environment.

3. Prediction of the corrosion current (hence the corrosion rate) as a function of potential.

4. Prediction of the total loss of material for the containment and post-containment periods.

8.3.5.9.3.2.4 Subactivity 1.4.3.2.4: Intergranular attack and intergranular stress corrosion cracking

Objectives

The objective of this subactivity is to determine whether sensitization is a necessary precursor to intergranular attack and intergranular stress corrosion cracking under the conditions anticipated for the repository at Yucca Mountain. This subactivity will also determine the model to predict the time to sensitization for materials under those conditions. For conditions where cracking might be expected, a model will be developed to predict the rate of crack growth.

Parameters

The information needed from other information needs includes

1. Postemplacement environment conditions.

2. Diffusion rate of chromium in the metal as a function of temperature.

3. Diffusion mechanism for chromium in the metal.

4. Strain.

5. Alloy composition.

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6. Effects of transformation products on diffusion rates.

7. Composition of carbide precipitates formed.

8. Amounts of sigma and chi phases.

The output parameters are a model to predict time to sensitization, a model to predict the probability of intergranular stress corrosion cracking and intergranular attack, and a crack propagation model.

Description

For the conditions expected at Yucca Mountain, sensitization is thought to be a necessary precursor (a prerequisite) for the intergranular stress corrosion cracking and intergranular attack of the candidate alloys. This subactivity will examine the premise that sensitization is a necessary precursor and document the conclusions of that analysis. If sensitization is determined to be a necessary precursor, a model will be developed to determine the time to sensitization under the relevant time-temperature conditions for the repository. The model will be based on the diffusion of chromium out of the metal matrix and the precipitation of carbides on the grain boundaries of the metal structure. The model will determine the time at which a continuous layer of material with chromium content less than 12 percent exists. This value is the boundary for which the passive film formed by uniform corrosion becomes unstable and leads to localized attack, especially in oxidizing environments.

The most important parameters in the model development are temperature, strain, and alloy composition. Temperature is important because the process is controlled by an activation energy. Strain is important because this can result in defects in the metal that lower the activation energy for diffusion, and alloy composition is important because of its effects on the diffusion rate of chromium and the availability of carbon to form the grain boundary chromium carbides.

The model development activities will begin with types 304 and 304L stainless steel (SS) and then extend to the molybdenum-bearing type 316 and 316L SS. The higher alloying content of the 316 types is expected to increase the activation energy for the diffusion process and thereby increase the time to develop a sensitized microstructure. The molybdenum additions also modify the chromium activity in the matrix and the carbide phases. Next, the model will be extended to the high-nickel alloy 825. In this alloy, other kinds of carbides and more complex carbides can form. The higher alloy content and more complex carbides will require a more complex model than that for types 304 and 316 SS. While alloy 825 is generally very resistant to sensitization, it is possible to sensitize this alloy.

Phases formed by transformation processes, such as discussed in Subactivity 1.4.3.2.1 (Section 8.3.5.9.3.2.1), can affect the susceptibility to intergranular attack in two ways. First, they can have different diffusion rates for chromium and can alter the time to sensitization of the metal microstructure. Second, some of the phases form at grain boundaries and are themselves subject to preferential attack under some environmental conditions. Examples of the latter are the sigma and chi phases.

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Crack initiation does not necessarily imply a defect through the container wall. To determine the rate of failure of the container by cracking, it is necessary to model the crack growth process. This model will consider the role of stress and oxidation-reduction potential on the rate of crack growth.

While sensitization appears to be the most important cause for intergranular attack modes, the possibility exists that other grain boundary precipitates could favor localized attack paths in these candidate materials. Sigma phase formation could be a possible intergranular precipitate in type 316L SS, as well as some of the aluminum, titanium, or molybdenum-rich phases in alloy 825. The possibilities of these will be investigated here and in conjunction with the activities discussed under metallurgical aging and transformation.

## 8.3.5.9.3.2.5 Subactivity 1.4.3.2.5: Hydrogen entry and embrittlement

### Objectives

The objective of this subactivity is to assess what level of hydrogen in austenitic materials is necessary to cause embrittlement of the material and to have a significant effect on other degradation rates and mechanisms. The subactivity will then examine the environmental conditions at Yucca Mountain to determine whether that amount of hydrogen would be available. If the level of hydrogen is available, then the necessary laboratory studies will be conducted under Information Need 1.4.2, and a model developed to determine the effects of hydrogen embrittlement.

### Parameters

The information needed from other information needs includes

1. Hydrogen production rate by radiolysis and corrosion.
2. Hydrogen recombination rate by all processes.
3. Maximum rate of hydrogen entry into the alloy.
4. Maximum concentration of hydrogen in the alloy.
5. Phase structure of the alloy.
6. Effects of hydrogen in austenitic materials.

### Description

The assessment of hydrogen effects centers around a bounding calculation for the maximum availability of atomic hydrogen at the metal surface (molecular hydrogen does not diffuse into the metal). The analysis will consider both the external and internal container environments. The latter is necessary because some fuel rods that breached in reactor service may contain water that would be released to the container inner atmosphere under disposal conditions.

The model will consider the maximum rate of hydrogen permeation in the metal (i.e., the net result of hydrogen entry and loss by outward diffusion). The total trapped hydrogen will be compared with the level that produces

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significant effects on the container material performance under Yucca Mountain conditions. High nickel materials (e.g., alloy 825) are sometimes more susceptible to hydrogen embrittlement than the types 304L and 316L stainless steels. If the amount of trapped hydrogen is less than the critical level, no further work will be done; if the amount is greater, the effects of the hydrogen will be assessed.

### .3.5.9.3.2.6 Subactivity 1.4.3.2.6: Pitting, crevice, and other localized attack

Objectives

The objective of the subactivity is to determine whether the necessary environmental conditions will exist to initiate pitting and crevice corrosion under Yucca Mountain repository conditions.

Parameters

The information needed from other information needs includes

1. Near-field waste package environment conditions, especially the concentration of ions known to favor these modes of attack.

2. Temperature.

3. Solution pH.

4. Chloride and fluoride ion concentration.

5. Metal microstructure.

6. Corrosion potential.

7. Pitting potential.

Output parameters are quantities of electrolyte needed to set up localized corrosion cells and a model to predict the likelihood of pitting or crevice corrosion.

Description

Pitting attack occurs when the temperature and chloride concentrations are sufficiently high and the pH sufficiently low to cause localized corrosion cells to be set up on the metal surface. The metal microstructure can also be important because it can lead to local breakdown of the passive corrosion films and to the establishment of galvanic cells. Sulfide inclusions can be particularly important in favoring pitting corrosion.

The model for pitting corrosion will determine critical values for the electrochemical potential above which pitting occurs and will determine whether this potential could be reached in the system under anticipated Yucca Mountain conditions.

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The model for crevice corrosion will use critical potential analysis combined with an analysis for the potential for propagation of the crevice attack. The latter analysis will use the crevice geometry and the local chemical conditions as its basis. The data for this model development will be collected in activities described under Information Need 1.4.2 (Section 8.3.5.9.2).

The model will determine the critical potential over a wide range of environmental conditions and alloy compositions and relate those potentials to the expected range of conditions for the repository and for the as-assembled container. Successful validation of the model at relatively high ionic strengths of relevant ions will add confidence to the extrapolations needed to reach the expected repository conditions of low chloride and other ionic contents.

#### 8.3.5.9.3.2.7 Subactivity 1.4.3.2.7: Transgranular stress corrosion cracking

Objectives

The objective of this subactivity is to determine the potential for transgranular stress corrosion cracking to occur under the repository disposal conditions, and if it occurs, to predict the rate of initiation and growth of transgranular cracks.

Parameters

The information needed from other information needs includes

1. Chloride concentrations of water that could contact the container.
2. Temperature.
3. Stress.
4. Alloy constituents.
5. Other ions in solutions.
6. Corrosion potential.

The output parameters will be critical potentials for crack initiation and propagation.

Description

The most significant parameters for this mode of degradation are the chloride ion concentrations in solutions in contact with the metal and the stress. At very high chloride concentrations, the critical stress is below the yield stress, while in dilute solutions it is above the yield stress. There is considerable uncertainty about the level of chloride that would cause the critical stress to be at the yield stress. This is important because the welded zone of a container would be at or near the yield stress. Therefore, this model development activity will attempt to determine the critical chloride level for Yucca Mountain disposal conditions.

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The model will consider the initiation of transgranular cracking to occur when the critical chloride concentration is reached. The concentration of oxygen, nitrate, and other oxidizing species is expected to influence the critical chloride level for crack initiation. The model will then provide the means to extrapolate to more dilute solutions, similar to those expected in the repository, and to provide a probability for the occurrence of transgranular cracking under those conditions. Crack growth, following crack initiation, will be modeled as a function of stress (or stress intensity), chloride content, pH, temperature, applied electrochemical potential, and content of other ionic species in solution.

The slip dissolution model for understanding the basic mechanism of stress corrosion cracking accounts for the environmental, metallurgical, and mechanical contributions in promoting stress corrosion cracking, and it treats both the properties of the metal and the oxide film on the metal surface. This model has been most extensively used in explaining stress corrosion cracking in Fe-Cr-Ni alloys, but it has been extended to other alloy systems (Staehle, 1971).

The propagation of stress corrosion cracks is determined by using sensitive techniques to measure the growth. Various geometries of test specimens are used; some types of specimens are self-loaded and others are stressed by application of a load external to the test cell environment. Some test techniques follow the growing crack directly by optical means (i.e., traveling microscope); other methods measure the crack growth by indirect means. Measurement of changes in the electrical resistance is probably the most sensitive of these techniques (Shreir, 1976).

### 8.3.5.9.3.2.8 Subactivity 1.4.3.2.8: Other potential degradation modes

This subactivity will screen other potential degradation modes not discussed previously to determine whether there is a cumulative probability of occurrence greater than 0.01 over the time interval of interest. If the probability exceeds that level, a model will be developed for the corrosion or degradation mode. Examples of models to be screened are mechanical fracture and the effect of microbiological activity on the previously discussed corrosion mechanisms. Another model for a possible degradation mode involves galvanic interaction of the metal container with other metallic components in the engineered barrier system and affiliated repository components. As mentioned in Section 8.3.4.2, the borehole liner and container material are proposed to be made from materials in the same alloy family to minimize galvanic effects.

### 8.3.5.9.3.3 Activity 1.4.3.3: Models for degradation of ceramic-metal, bimetallic/single metal, and coatings and filler alternative systems

With the introduction of alternative waste package container concepts based on ceramic-metal, bimetallic/single metal, and coatings and fillers, the spectrum of potential degradation modes for these three classes of

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materials systems will strongly depend on the materials chosen for inclusion as a result of the feasibility study described in Information Need 1.4.1 (Section 8.3.5.9.1). This activity, which is divided into three subactivities (one for each class of material system), will be developed in more detail in progress reports.

### 8.3.5.9.3.3.1 Subactivity 1.4.3.3.1: Models for degradation of ceramic-metal systems

At least two plausible degradation modes can be identified for the ceramic element of ceramic-metal systems: (1) aqueous corrosion and (2) delayed fracture in the presence of residual and applied stresses. Brief discussions of these degradation modes follow.

1. Aqueous corrosion. The objective is to determine for very long lifetime material the amount of loss by general aqueous corrosion and to establish whether a uniform pattern of attack occurs. Aqueous corrosion can occur when a more or less continuous moisture film is present on the container surface or when some portion of the container surface is immersed in water.

The information needed from other information needs includes

1. Results of weight loss tests.
2. Description of the environment near the waste package surface.
3. Description of corrosion product layers.
4. Chemical modeling of solution composition.
5. Radiolysis effects in aqueous media.

The output parameters are estimates of the material loss from containers that can occur during the containment and postcontainment periods. The Project would like to be able to characterize aqueous corrosion well beyond the thousand-year postclosure period. The controlled release rate models for radionuclides will depend on the potential for the container to affect water movement to and from the waste form.

2. Delayed fracture in the presence of residual and applied stresses. The objective is to model and determine the long range failure potential of the ceramic element in ceramic-metal systems due to effects such as pores, weakened or stressed grain boundaries, inclusions, and cracks. These defects cause stress concentrations when the material is subjected to load. Locally the theoretical strength is exceeded and the defect grows until failure occurs.

### 8.3.5.9.3.3.2 Subactivity 1.4.3.3.2: Models for degradation of bimetallic/single metal systems

Depending on the particular materials chosen, the identification of critical degradation modes, and the results of laboratory programs for each of these critical degradation modes, models will be developed and validated.

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The potential critical degradation modes that may require detailed modeling are enumerated below.

| Bimetal container | Single metal container | Potential critical degradation mode |
|---|---|---|
| X | X | Metallurgical aging and phase transformations in base metal, heat-affected zones, and welds |
| X | X | Low temperature oxidation |
| X | X | General aqueous corrosion |
| X | X | Intergranular attack and stress corrosion cracking |
| X | X | Hydrogen entry and embrittlement |
| X | X | Pitting, crevicing, and other localized attack |
| X | X | Gamma flux effects |
| X | | Galvanic effects at welds, oxide inclusions, and surface oxides |

8.3.5.9.3.3.3 Subactivity 1.4.3.3.3: Models for degradation of coatings and filler systems

Depending on the particular materials chosen, the identification of critical degradation modes, and the results of laboratory programs for each of these critical deformation modes, models will be developed and validated. The potential critical degradation modes that may require detailed modeling are

1. Low temperature oxidation.
2. Metallurgical stability and toughness under repository conditions.
3. General aqueous corrosion.
4. Hydrogen entry and embrittlement.
5. Gamma flux effects.
6. Mechanical degradation.
7. Galvanic corrosion.
8. Localized corrosion.

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8.3.5.9.4 Information Need 1.4.4: Estimates of the rates and mechanisms of container degradation in the repository environment for anticipated and unanticipated processes and events, and calculation of the failure rate of the container as a function of time

Technical basis for addressing the information need

Link to the technical data chapters and the applicable support documents

The bases for the models required to obtain these estimates have been discussed in Section 7.4.5. The activities that develop data, parameters, and models to obtain these estimates are described in Sections 8.3.3., 8.3.4, 8.3.5.9, and 8.3.5.10.

Parameters

Parameters needed for estimating rates and mechanisms of container degradation include the following:

1. Waste package design (Information Need 1.10.2, Section 8.3.4.2.2).

2. Waste package design features affecting the performance of the container (Information Need 1.4.1, Section 8.3.5.9.1).

3. Material properties of the container (Information Need 1.4.2, Section 8.3.5.9.2).

4. Scenarios for anticipated and unanticipated processes and events, and models for extrapolation of container performance (Information Need 1.4.3, Section 8.3.5.9.3, and Activity 1.5.3.1.1, Section 8.3.5.10.3.1.1).

5. Characteristics of the shaft and borehole seals that may affect waste package container performance (Information Needs 1.12.1, 1.12.2, and 1.12.4 in Section 8.3.3.2.1, 8.3.3.2.2, and 8.3.3.2.4).

6. Waste package system model and uncertainty methodology (Information Need 1.5.3, Section 8.3.5.10.3).

7. Waste package environment description (Information Need 1.10.4; Section 8.3.4.2.4).

The output parameters are the rates of container degradation and container failure rate.

Logic

Once the environmental scenarios for calculating time to failure of containers and the models for predicting failure of containers have been developed and tested, the rates of container degradation may be estimated. These estimates will then be used to calculate the failure rate of the container as a function of time. The models and methodologies used for this calculation are developed in Section 8.3.5.10.3 under Information Need 1.5.3 and applied here for the container failure rate calculation.

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One activity will be performed under this information need. It will exercise both the deterministic system model and its associated uncertainty methodology developed in Information Need 1.5.3.

8.3.5.9.4.1 Activity 1.4.4.1: Estimate of the rates and mechanisms of container degradation in the repository environment for anticipated and unanticipated processes and events, and calculation of container failure rate as a function of time

The following two subactivities support this analysis.

8.3.5.9.4.1.1 Subactivity 1.4.4.1.1: Deterministic calculation of rates of container degradation in the repository environment for anticipated and unanticipated processes and events, and calculation of container failure rate as a function of time

Objectives

The objective of this subactivity is to use the deterministic waste package system model developed in Activity 1.5.3.5 (Section 8.3.5.10.3.5) and the scenarios developed in Activity 1.5.3.1 (Section 8.3.5.10.3.1) to estimate (1) the container degradation rates and (2) the time to initiation of release of radionuclides from the waste package. This system model incorporates models for container performance developed in Information Need 1.4.3 (Section 8.3.5.9.3).

Parameters

The parameters required for this activity are given in the preceding combined list in the technical basis section for the information need. The output parameters are the times at which the corrosion modes can be initiated (due to aging, sensitization, and environmental conditions), the rates of container degradation, and time to initiation of release of radionuclides from the waste package under specified conditions for the scenarios representing anticipated and unanticipated processes and events.

Description

The system model is discussed in Section 8.3.5.10.3. The estimates of container performance will be made in three phases: (1) for the design concepts discussed in Section 7.3, (2) for the advanced conceptual design, and (3) for the license application design. The later phases will use modeling concepts developed in the previous phases, and therefore are difficult to discuss at this point. However, it is likely that analyses in all phases will incorporate many of the same elements.

The analysis of waste package designs will proceed by assembling sets of system model input parameters developed in Section 8.3.5.9.3 (Information Need 1.4.3) and executing the system model code to obtain estimates of rates and mechanisms of container degradation. These estimates will be calculated

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for the range of values of those parameters determined to be important to container performance. These estimates will be calculated for scenarios that represent both anticipated and unanticipated processes and events. In addition, in the earlier phases of waste package design, information developed in the system model calculations will be available as input to later design phases.

8.3.5.9.4.1.2 Subactivity 1.4.4.1.2: Probabilistic calculation of rates of container degradation and distribution of time to initiation of release of radionuclides from the waste packages

Objectives

Because of heterogeneities in both the environment and components of the waste package design, deterministic calculation of performance alone will not be sufficient to provide the performance measures for the set of waste packages for this issue and to support the reasonable assurance standard required by the NRC. The objective of this subactivity is to provide a probabilistic analysis of waste package container performance addressing these uncertainties, using the uncertainty modeling methodologies developed in Activity 1.5.3.5 (Section 8.3.5.10.3.5).

Parameters

The input parameters for the activity are given in the preceding combined list for the information need. The output parameter is the cumulative distribution function for time to initiation of release of radionuclides from the waste package.

Description

The uncertainty methodologies developed in Activity 1.5.3.5 (Section 8.3.5.10.3.5) will be employed using the waste package system model to assess the reliability of the waste package with respect to failure of the container. This task will be accomplished in concert with the phases of system model development and application. Development of the waste package system model is discussed in Section 8.3.5.10.3, in the context of a model for release calculations. However, the waste package system model will also provide the time to failure of the container. The most likely approach for determining the distribution for time to failure of the container and initiation of the release of radionuclides from the waste package will be to exercise the system model for a range of model inputs selected by a procedure for sampling from distributions of input variables. The input variables may be random variables having probability distributions, or they may be variables that range over known actual distributions. The latter case might apply for example, to the distribution of package heat generation rates after all the packages have been loaded and documented. For less important input variables, bounding distributions may be used.

The uncertainty calculations will be performed for each of the design phases, although they are only required for the license application design analysis. This procedure will allow testing on the early design phases, and

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modifications of other methodology during later phases. At least two types of uncertainty will be addressed. First, the uncertainty in the predicted times to failure of the containers resulting from uncertainties in the fabrication and environment of the waste packages will be calculated. Then the secondary uncertainty (that is the confidence in the best estimate of cumulative distribution function for time to failure of the containers) will be assessed. Together with the deterministic simulations for bounding cases for time to container failure, these results will provide the time of initiation of the radionuclide release from the waste package. Thus, these results will address container failures, whose limitations during the containment period is one of the design objectives for resolution of this issue.

### 8.3.5.9.5 Information Need 1.4.5: Determination of whether the set of waste packages meets the performance objective for substantially complete containment for anticipated processes and events

Technical basis for addressing the information need

Link to the technical data chapters and the applicable support documents

The basis for the models required to perform these calculations has been discussed in Section 7.4.5. The activities that perform these calculations are described in Sections 8.3.5.9.4 and 8.3.5.10.4. The activities that develop data, parameters, and models to support the calculations in Sections 8.3.5.9.4 and 8.3.5.10.4 are described in earlier sections of 8.3.5.9 and 8.3.5.10, respectively.

Parameters

The parameters needed for the determination of whether the substantially complete containment performance objective for anticipated processes and events is met are as follows:

1. Quantitative interpretation of substantially complete containment.

2. Calculation of times to initiation of release of radionuclides from the waste package from Section 8.3.5.9.4 (Information Need 1.4.4).

3. Release rate of radionuclides from failed waste packages from Section 8.3.5.10.4 (Information Need 1.5.4).

The output parameter is a determination of whether substantially complete containment has been satisfied during the containment period. If not satisfied, a second output parameter is the earliest time at which the requirement is not satisfied.

Logic

The design requirements set to fulfill substantially complete containment (discussed in Section 8.3.5.9, under Regulatory basis for the issue) will impose quantitative requirements on performance measures to be

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maintained during the containment period up to 1,000 yr after closure. To evaluate these design requirements, the following quantities are required:

1. The fraction of radioactivity retained within the set of waste packages for the duration of the containment period.

2. The annual release rate of radioactivity from the engineered barrier system.

The results of calculations to determine the percentage of containers that will not provide total containment of radionuclides for the duration of the containment period and times to container failure are taken from Information Need 1.4.4, Section 8.3.5.9.4. The release rate of radionuclides summed over the subset of failed containers and the total quantity of radioactivity inside the waste packages are taken from Information Need 1.5.4, Section 8.3.5.10.4. These calculational results are compared with the design objectives of the interpretation of substantially complete containment to determine whether this issue (1.4) has been resolved.

One activity will be performed under this information need. It will compare the calculation of performance of the repository ensemble of waste packages with the interpretation of substantially complete containment.

### 8.3.5.9.5.1 Activity 1.4.5.1: Determination of whether the substantially complete containment requirement is satisfied

Objectives

Waste package system modeling results developed in Activity 1.4.4.1 (Section 8.3.5.9.4.1) and Information Need 1.5.4 (Section 8.3.5.10.4) will be used to predict waste package containment performance using the scenarios and models developed in Section 8.3.5.10.3. The results of these calculations will then be compared with the interpretation of substantially complete containment to determine whether the performance objective has been met for all times during the containment period.

Parameters

The parameters required for this investigation are given in the technical basis section for this information need. The output parameter is the determination of substantially complete containment under specified conditions represented by the scenarios.

Description

The calculation of waste package container performance was made in Information Need 1.4.4 (Section 8.3.5.9.4), and the calculations for release of radionuclides from failed waste packages are performed in Information Need 1.5.4 (Section 8.3.5.10.4). Comparison of these results with the interpretation of substantially complete containment (Section 8.3.5.9) will complete this investigation.

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## 8.3.5.10 Issue resolution strategy for Issue 1.5: Will the waste package and repository engineered barrier systems meet the performance objective for radionuclide release rates as required by 10 CFR 60.113?

### Regulatory basis for the issue

The NRC regulations will set a performance objective for control of the release rate of radionuclides from the engineered barrier system for the time period following the end of the containment period. The Environmental Protection Agency (EPA) has determined that the duration of the period of regulatory concern extends for 10,000 years following permanent closure of the repository. The Nuclear Waste Policy Act requires that the NRC regulations not be inconsistent with the EPA standards (NWPAA, 1983). Therefore, the DOE infers that the performance objective for controlled release extends from the end of the containment period to 10,000 years following permanent closure of the repository.

The portion of 10 CFR Part 60 that sets the performance objective for control of radionuclide release rate is Section 60.113(a)(1)(ii), and it states, in part, the following:

> the engineered barrier system shall be designed, assuming anticipated processes and events, so that...(B) The release rate of any radionuclide from the engineered barrier system following the containment period shall not exceed one part in 100,000 per year of the inventory of that radionuclide calculated to be present at 1,000 years following permanent closure, or such other fraction of the inventory as may be approved or specified by the Commission; provided that this requirement does not apply to any radionuclide which is released at a rate less than 0.1 percent of the calculated total release rate limit. The calculated total release rate limit shall be taken to be one part in 100,000 per year of the inventory of radioactive waste, originally emplaced in the underground facility, that remains after 1,000 years of radioactive decay.

This issue is specifically restricted to showing that the engineered barrier system is designed in conformance with the statement quoted in the preceding paragraph; however, there are other needs for release rate information to support resolution of other issues. To simplify the presentation of the plans and to minimize redundancy in the discussion, the information needs under this issue will include plans to gather data to support resolution of the following issues:

1. Issue 1.1: This issue requires source term data for use in the system analysis calculations. Data on the release rate of radionuclides from the engineered barrier system for a period of 10,000 yr under anticipated processes and events to support these calculations will be provided. Data on release rates of radionuclides under lower probability scenarios (unanticipated processes and events) for 10,000 yr will also be provided. Plans for collection of the data will be given in Information Needs 1.5.4 and 1.5.5 (Sections 8.3.5.10.4 and 8.3.5.10.5).

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2. Issue 1.4: This issue requires information on the rate of cladding failure and radionuclide release rate data from failed containers during the containment period. Plans for collection of the data will be given in Information Needs 1.5.1 through 1.5.4 (Section 8.3.5.10.1 through 8.3.5.10.4).

3. Issue 1.9: This issue deals with the higher level findings required under 10 CFR Part 960. In particular, 10 CFR 960.3-1-5 requires calculation of the cumulative releases to the accessible environment during 100,000 yr. The data gathered in Issue 1.5 will be used to support resolution of Issue 1.9.

These issues are addressed in Sections 8.3.5.13 (Issue 1.1), 8.3.5.9 (Issue 1.4), and 8.3.5.18 (Issue 1.9).

Approach to resolving the issue

The overall waste package compliance strategy was shown in Figure 8.3.4-1 with further details in Section 8.3.4. The essence of the waste package strategy lies in an iterative process of performance allocation, performance assessment, and testing to determine if the goals are met. If not, changes are made in design, materials, etc., and the process is repeated until the design objectives are met. Within this overall waste package compliance strategy, the strategy for resolution of Issue 1.5 is based on present knowledge of the repository emplacement environment, the data gathered on waste form performance in environments that can be related to the projected repository environment, and the use of models to assess the performance of various system elements. The testing and design activities described in this section are tentative and are subject to change. Any such change will be reported in semiannual progress reports.

Figure 8.3.5.10-1 shows the hierarchy of models. The highlighted portion is used in the resolution of Issue 1.5 and to provide input to Issues 1.1, 1.4, and 1.9. The lower levels of detailed models support the higher levels of aggregated models. The system model and the flow and transport model are used to assess the net performance with respect to regulatory issues. The experimental studies and activities (not shown in the figure) support the detailed models by explaining mechanisms and processes, guiding model development, examining processes to make sure that no important phenomena are being overlooked, providing data for models, and validating models.

Principal input parameters for the highlighted models are presented in Table 8.3.5.10-1. Other models that support the resolution of Issue 1.5 are found in Sections 8.3.4.2 and 8.3.5.9.

Under the current conceptual model, the repository horizon is located in the unsaturated zone in an area in which the downward vertical water flux is believed to be less than 0.5 mm/yr. Thus, negligible water is expected to contact the containers throughout the post-containment period. However, the potential release of radionuclides has been analyzed for the case in which some water may contact the waste packages as the repository cools. During the containment period, a maximum of 5 L/yr was allowed to contact up to 10 percent of the packages. In the post-containment period, the allowable

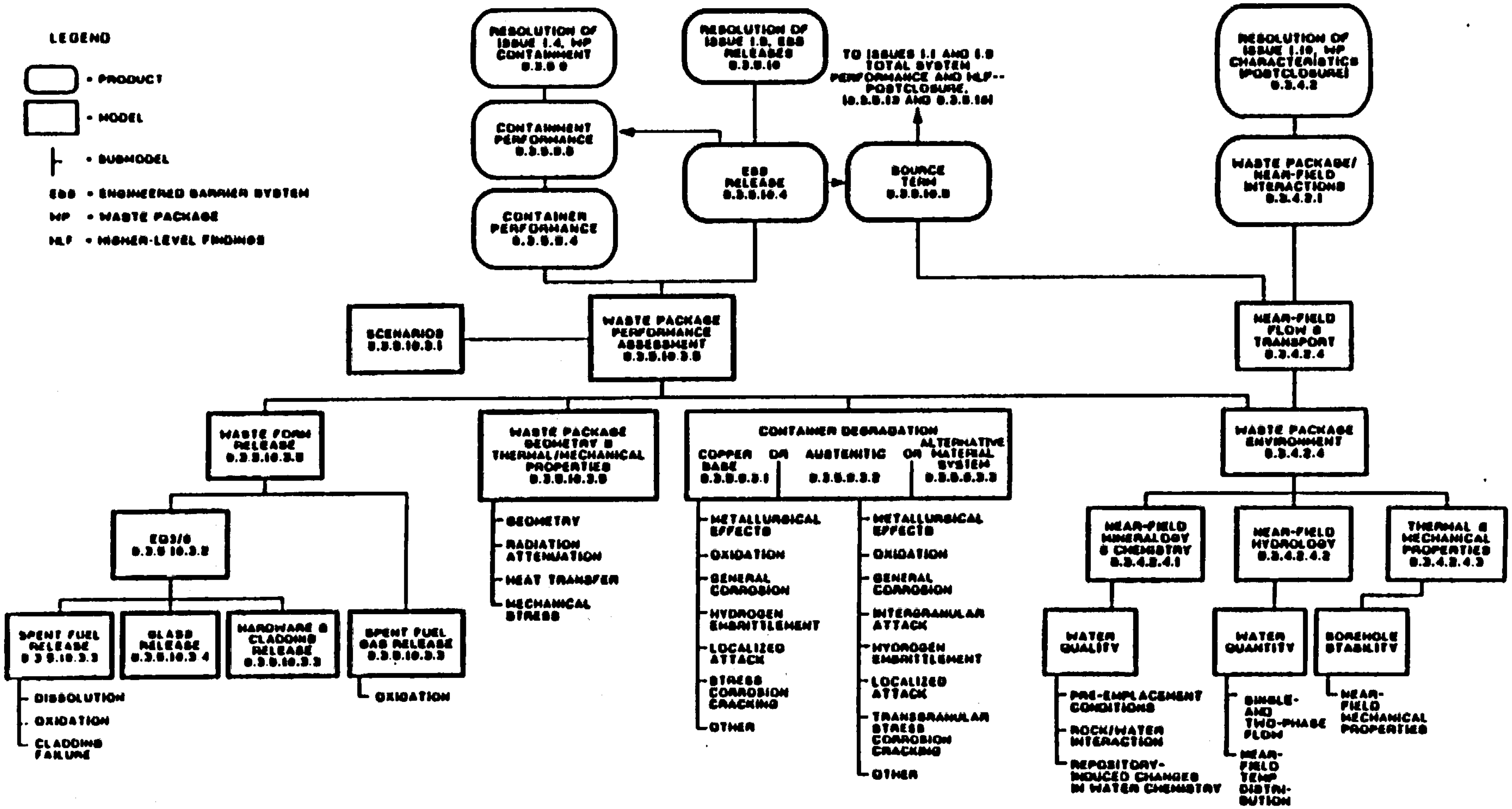


Figure 8.3.5.10-1. Model hierarchy for Issue 1.5 (engineered barrier system release rates).

Table 8.3.5.10-1. Input to predictive models for Issue 1.5, engineered barrier system release rates (page 1 of 7)

| Model | Model input | Needed confidence | SCP section |
|---|---|---|---|
| Scenarios | Parameters for nominal case and for potentially significant disturbed scenarios | High | 8.3.5.13 |
| | Parameters for modeling changes in geologic, hydrologic, and geomechanical conditions | Medium to high | 8.3.1.4, 8.3.1.5, 8.3.1.6, 8.3.1.7, 8.3.1.8 |
| | Parameters for modeling changes in geohydrologic geochemical conditions | Medium to high | 8.3.1.2, 8.3.1.3 |
| | Characteristics of shaft and borehole seals | Medium to high | 8.3.3.2 |
| | Characteristics of repository and engineered barriers | Medium to high | 8.3.2.2 |
| | Characteristics of waste package designs | High | 8.3.4.2.2, 8.3.4.2.3, 8.3.4.2.4 |
| | Waste package container failure modes and times | High | 8.3.5.9.4 |
| Waste package performance assessment | Scenarios | High | 8.3.5.10.3.1 |
| | Waste package geometry model | High | 8.3.5.10.3.5 |
| | Radiation attenuation model | Medium | 8.3.5.10.3.5 |
| | Heat transfer model | High | 8.3.5.10.3.5 |
| | Mechanical stress model | High | 8.3.5.10.3.5 |
| | Waste package environment (water movement and chemistry) model | High | 8.3.4.2.4 |
| | Container corrosion and degradation model | High | 8.3.5.9.3 |
| | Waste form release model | High | 8.3.5.10.3.5, 8.3.5.10.3.2, 8.3.5.10.3.3 |



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Table 8.3.5.10-1. Input to predictive models for Issue 1.5, engineered barrier system release rates (page 2 of 7)

| Model | Model input | Needed confidence | SCP section |
|---|---|---|---|
| Waste form release | EQ3/6 model for glass and spent fuel | High | 8.3.5.10.3.5 |
| | Gas release model | High | 8.3.5.10.3.3 |
| | Container failure rate | High | 8.3.5.9.4 |
| | Container configurations after failure | High | 8.3.5.9.4 |
| | Temperature from heat transfer model | High | 8.3.5.10.3.5 |
| | Water flow quantity | High | 8.3.4.2.4.2 |
| | Mechanism of water contact with waste package | High | 8.3.5.9.4, 8.3.4.2.4.2 |
| | Water quality | High | 8.3.4.2.4.1 |
| EQ3/6 waste model | Waste form degradation models | | 8.3.5.10.3 |
| | Spent fuel | High | 8.3.5.10.3.3 |
| | Hardware and cladding | High | 8.3.5.10.3.3 |
| | Glass | High | 8.3.5.10.3.4 |
| | Temperature | High | 8.3.5.10.3.1 |
| | Water flux contacting waste | High | 8.3.4.2.4.1 |
| | Water chemistry contacting waste | High | 8.3.4.2.4.1 |
| | Thermodynamic data for solids, gases, and aqueous species resulting from waste release | High | 8.3.5.10.3.2.1 |
| | Waste degradation scenarios | High | 8.3.5.10.3.1 |
| Spent fuel release | Water flux contacting waste | | 8.3.4.2.4.1 |
| | Near-field flux | High | 8.3.4.2.4.1 |
| | Water entering container | High | 8.3.5.10.3.5.3 |
| | Water contact scenario | High | 8.3.5.10.3.1 |
| | Water chemistry contacting waste | | 8.3.4.2.4.1 |
| | Initial chemistry | High | 8.3.4.2.4.1.3 |



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Table 8.3.5.10-1. Input to predictive models for Issue 1.5, engineered barrier system release rates (page 3 of 7)

| Model | Model input | Needed confidence | SCP section |
|---|---|---|---|
| Spent fuel release (continued) | Radiation-induced changes | Medium | 8.3.4.2.4.1.5 |
| | Repository material-induced changes | High | 8.3.4.2.4.1.2 |
| | Temperature-induced changes | High | 8.3.4.2.4.1.1 |
| | Corrosion-induced changes | High | 8.3.4.2.4.1.6 |
| | Temperature | High | 8.3.5.10.3.1 |
| | Fuel composition | High | 8.3.5.10.1.1.1 |
| | Fission gas release | High | 8.3.5.10.1.1.1 |
| | Oxidation state | High | 8.3.5.10.2.1.2 |
| | Cladding condition | Medium | 8.3.5.10.2.1.3 |
| | Fuel degradation rate constants | High | 8.3.5.10.2.1 |
| | Fuel dissolution rates | High | 8.3.5.10.2.1.1 |
| | Effect of | | |
| | Burnup | High | 8.3.5.10.2.1.1 |
| | Oxidation state | High | 8.3.5.10.2.1.1 |
| | Reactor type | Medium | 8.3.5.10.2.1.1 |
| | Grain size | High | 8.3.5.10.2.1.1 |
| | Radiation field | Medium | 8.3.5.10.2.1.1 |
| | Radionuclide content (at time of water contact) | High | 8.3.5.10.1.1.1 |
| | Container material | Medium | 8.3.5.10.1.1.1 |
| | Other waste characteristics | Medium | 8.3.5.10.1.1.1 |
| | Other repository characteristics | Medium | 8.3.5.10.1.1.3 |
| Glass release | Water flux contacting waste | | 8.3.4.2.4.1 |
| | Near field flux | High | 8.3.4.2.4.1 |
| | Water entering container | High | 8.3.5.10.3.5.3 |
| | Water contact scenario | Medium | 8.3.5.10.3.1 |



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Table 8.3.5.10-1. Input to predictive models for Issue 1.5, engineered barrier system release rates (page 4 of 7)

| Model | Model input | Needed confidence | SCP section |
|---|---|---|---|
| Glass release (continued) | Water chemistry contacting waste | | 8.3.4.2.4.1 |
| | Initial chemistry | High | 8.3.4.2.4.1.3 |
| | Radiation-induced changes | Medium | 8.3.4.2.4.1.5 |
| | Repository material-induced changes | High | 8.3.4.2.4.1.2 |
| | Temperature-induced changes | High | 8.3.4.2.4.1.1 |
| | Corrosion-induced changes | Medium | 8.3.4.2.4.1.6 |
| | Temperature | High | 8.3.5.10.3.1 |
| | Glass composition | High | 8.3.5.10.1.1.2 |
| | Glass degradation rate constants | High | 8.3.5.10.2.2 |
| | Glass dissolution rates | High | 8.3.5.10.2.2.1 |
| | Effect of interactions on rates | High | 8.3.5.10.2.2.2 |
| | Radionuclide content (at time of water contact) | High | 8.3.5.10.1.1.2 |
| | Ratio of glass surface area to water volume | High | 8.3.5.10.3.1 |
| | Container material | Medium | 8.3.5.10.1.1.2 |
| | Pour canister material | High | 8.3.5.10.1.1.2 |
| | Glass handling history | High | 8.3.5.10.1.1.2 |
| | Conformation with waste acceptance specifications | High | 8.3.5.10.1.1.2 |
| | Other waste characteristics | Medium | 8.3.5.10.1.1.2 |
| | Other repository characteristics | Medium | 8.3.5.10.1.1.3 |
| Hardware and cladding release | Water flux contacting waste | | 8.3.4.2.4.1 |
| | Near field flux | High | 8.3.4.2.4.1 |
| | Water entering container | High | 8.3.5.10.3.5.3 |
| | Water contact scenario | High | 8.3.5.10.3.1 |
| | Water chemistry contacting waste | | 8.3.4.2.4.1 |
| | Initial chemistry | High | 8.3.4.2.4.1.3 |
| | Radiation-induced changes | Medium | 8.3.4.2.4.1.5 |



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Table 8.3.5.10-1. Input to predictive models for Issue 1.5, engineered barrier system release rates (page 5 of 7)

| Model | Model input | Needed confidence | SCP section |
|---|---|---|---|
| Hardware and cladding release (continued) | Repository material-induced changes | High | 8.3.4.2.4.1.2 |
| | Temperature-induced changes | High | 8.3.4.2.4.1.1 |
| | Corrosion-induced changes | Medium | 8.3.4.2.4.1.6 |
| | Temperature | High | 8.3.5.10.3.1 |
| | Hardware and cladding composition | High | 8.3.5.10.1.1.1 |
| | Degradation rate constants | High | 8.3.5.10.2.1.3, 8.3.5.10.2.1.4 |
| | Humidity | Medium | 8.3.5.10.2.1.3, 8.3.5.10.2.1.4 |
| | Metal compatibilities | High | 8.3.5.10.2.1.3, 8.3.5.10.2.1.4 |
| | Radiation field | Medium | 8.3.5.10.2.1.3, 8.3.5.10.2.1.4 |
| | Irradiation history | High | 8.3.5.10.2.1.3, 8.3.5.10.2.1.4 |
| | Oxide thickness on cladding | High | 8.3.5.10.2.1.3 |
| | Hydride content of cladding | Medium | 8.3.5.10.2.1.3 |
| | Radionuclide content (at time of water content) | High | 8.3.5.10.1.1.1 |
| | Container material | Medium | 8.3.5.10.1.1.1 |
| | Other waste characteristics | Medium | 8.3.5.10.1.1.1 |
| | Other repository characteristics | Medium | 8.3.5.10.1.1.3 |
| Spent fuel gas release | Gas release scenario | High | 8.3.5.10.3.1 |
| | Temperature | High | 8.3.5.10.3.1 |
| | Fuel composition | High | 8.3.5.10.1.1.1 |
| | Cladding composition | High | 8.3.5.10.1.1.1 |
| | Hardware composition | High | 8.3.5.10.1.1.1 |





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Table 8.3.5.10-1. Input to predictive models for Issue 1.5, engineered barrier system release rates (page 6 of 7)

| Model | Model input | Needed confidence | SCP section |
|---|---|---|---|
| Spent fuel gas release (continued) | Fuel oxidation state | High | 8.3.5.10.2.1.2 |
| | Cladding condition | Medium | 8.3.5.10.2.1.3 |
| | Humidity | Medium | 8.3.5.10.2.1.3, 8.3.5.10.2.1.4 |
| | Radiation field | Medium | 8.3.5.10.2.1.3, 8.3.5.10.2.1.4 |
| | Irradiation history | High | 8.3.5.10.2.1.3, 8.3.5.10.2.1.4 |
| | Oxide thickness on cladding | High | 8.3.5.10.2.1.3 |
| | Radionuclide content | High | 8.3.5.10.1.1.1 |
| | Container material | Medium | 8.3.5.10.1.1.1 |
| | Other waste characteristics | Medium | 8.3.5.10.1.1.1 |
| | Other repository characteristics | Medium | 8.3.5.10.1.1.3 |
| Waste package geometry and thermal/mechanical properties | Geometry model | High | 8.3.5.10.3.5 |
| | Radiation attenuation model | Medium | 8.3.5.10.3.5 |
| | Heat transfer model | High | 8.3.5.10.3.5 |
| | Mechanical stress model | High | 8.3.5.10.3.5 |
| Geometry | Borehole and waste package configuration, dimensions | High | 8.3.4.2.2, 8.3.4.2.3 |
| | Waste package content | | |
| | Materials | High | 8.3.4.2.2 |
| | Mass | High | 8.3.4.2.2 |
| | Elemental composition | Medium | 8.3.5.10.1.1, 8.3.4.2.2 |



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Table 8.3.5.10-1. Input to predictive models for Issue 1.5, engineered barrier system release rates (page 7 of 7)

| Model | Model input | Needed confidence | SCP section |
|---|---|---|---|
| Geometry (continued) | Isotopic composition | | |
| | Important constituents | High | 8.3.5.10.1.1 |
| | Minor constituents | Medium | 8.3.5.10.1.1 |
| Radiation attenuation | Radiation source strength | High | 8.3.5.10.1.1 |
| | Gamma ray attenuation coefficients of materials | Medium | 8.3.4.2.2 |
| | Dose rate at waste form surface | Medium | 8.3.5.10.1.1 |
| | Dose rate at package surface | Medium | 8.3.4.2.2 |
| | Decay heat generation rates | High | 8.3.4.2.2 |
| Heat transfer (thermal) model | Thermal properties (heat capacity, conductivity) of single materials | High | 8.3.4.2.2 |
| | Effective thermal properties of composite materials | High | 8.3.4.2.2 |
| | Surface properties for convective and radiative heat transfer | Medium | 8.3.4.2.2 |
| | Interaction with host rock heat transfer | High | 8.3.4.2.4.3 |
| | Decay heat generation rate | High | 8.3.4.2.4.3 |
| Mechanical model | Mechanical properties of single materials | High | 8.3.4.2.2 |
| | Mechanical properties of composite materials | Medium | 8.3.4.2.2 |
| | Mechanical loads | High | 8.3.4.2.2, 8.3.4.2.4.3 |
| | Temperature field within package | Medium | 8.3.4.2.4.4 |





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quantity of water has been increased to reflect the possible increased capacity of existing water flow paths due to contraction cooling of the rock mass that causes increased fracture aperture, or the anticipated range of changes in climate during the post-containment period.

Figures 8.3.5.10-2 and 8.3.5.10-3 show the overall outline for the reference and alternative approaches to be used in the resolution of this issue. The reference approach includes branches for both the expected case, in which the amount of water contacting the waste form is negligible, and a bounding case (bounding for anticipated processes and events), in which 20 L/yr is allowed to contact the waste forms in up to 10 percent of the waste packages. The approach taken to resolve Issue 1.5 takes into account both a transition period, when the fraction of failed containers may increase and the fraction of wetted containers may increase, and an upper-limit period considering the environment and container performance limits of Table 8.3.5.10-2 as a steady state. Possible pulse releases from spent fuel of fractions of carbon-14 and of gap and grain boundary elements must be considered during the transition period. No assumption is made about the container performance during the containment period for the assessment of Issue 1.5; all possibilities between 0 and 1 fraction of containers failed and between 0 and 0.10 of waste packages in contact with liquid water at the start of the controlled-release period will be considered.

The data presented in Section 7.4.3 indicate that it is very likely that the performance objective for control of release rate from the engineered barrier system can be met by the waste forms in an unprotected condition, provided that the analysis is done using the conditions of the expected case. For the bounding case, the performance objective can be met provided credit can be taken for the fraction of waste packages where the waste form is not contacted by water, and for the mass-transfer resistance of breached containers and cladding. This resistance to release of radionuclides can be provided by breached containers and cladding, even in their degraded condition.

The limitation of wetted waste forms to 10 percent of the total depends on environmental and engineered elements. The existing information is not sufficient to allow a final selection of the components and performance measures. Several components and processes may provide barriers to water contact. These include

1. Hydrological--alteration of flow paths by the dehydration-rehydration cycle, and limited water flux available to reestablish prerepository partial saturation levels.

2. Water flux retained in porous rock component; not enough water flux for fracture flow or dripping.

3. Container and waste form are hotter than surroundings, can evaporate water.

4. Air gap will separate partially saturated rock from waste form over most of the perimeter of the waste package.

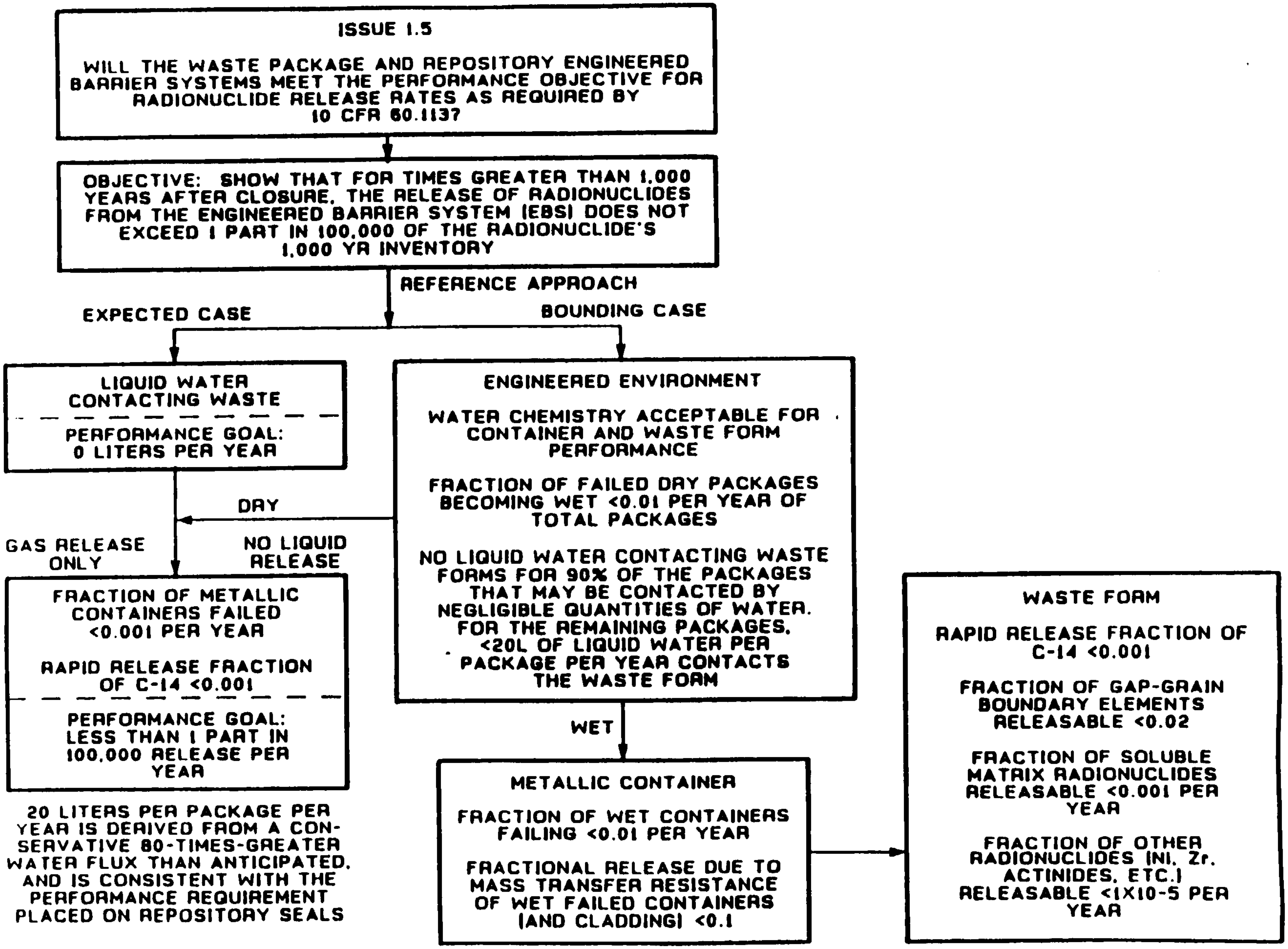


**Figure 8.3.5.10-2.** Reference approach to resolving Issue 1.5 (engineered barrier system performance).



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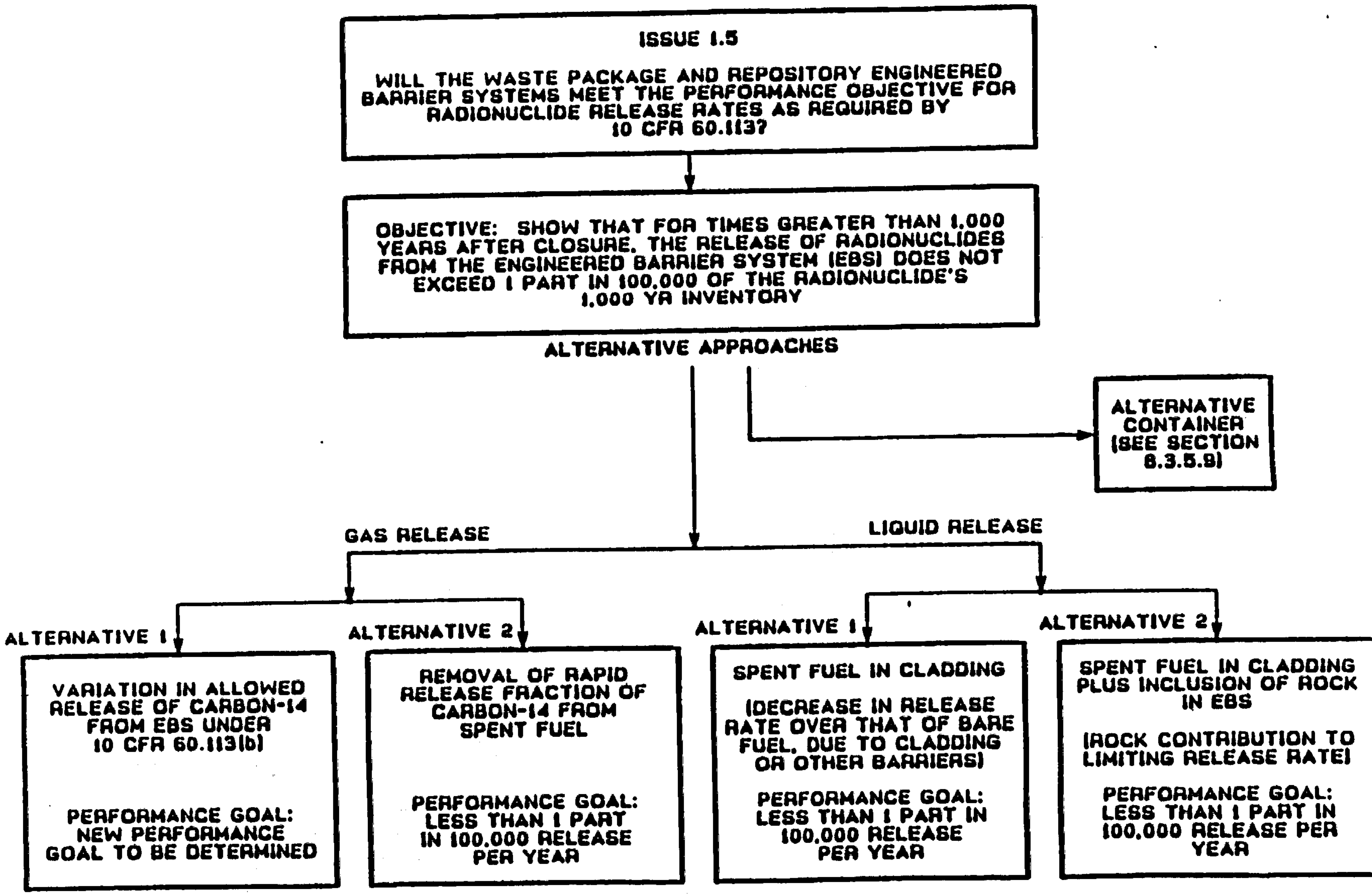


Figure 8.3.5.10-3. Alternative approaches to resolving Issue 1.5 (engineered barrier system performance), assuming case of maximum water flux per package (20 liters per year).

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5. Liner and container, even with breaches, will provide a function of separation of partially saturated rock from waste form.

6. Limited surface area of waste form contacted by water.

Because the repository horizon is in the unsaturated zone, release into the gas phase must be considered. For the expected case, negligible flux of liquid water, this may be the dominant release mechanism. Because of its long half-life, carbon-14 is the only significant radionuclide available for gas-phase release during the controlled release period. Because of the lower gamma fluxes and temperatures in the controlled release period, present data suggest that the release rate will be sufficiently low to meet the requirements of 10 CFR 60.113, based on the low release fraction and the low annual container failure rate.

Figure 8.3.5.10-3 outlines the various alternatives to be used if the reference approach proves inadequate to resolve the issue. The principal concerns are for carbon-14 on the exterior of cladding and assembly components and the readily soluble radionuclides present in the fuel.

Currently, there is considerable uncertainty in the release rates (dry and aqueous), mechanisms, and locations of carbon-14; because of this, two alternative approaches to carbon-14 control are given in Figure 8.3.5.10-3. These will be used if the carbon-14 release rates from failed containers are found to exceed the 10 CFR 60.113(a) limits and re-allocation of performance does not result in compliance. The first alternative would be to request a new allowed release rate for carbon-14 under the provisions of 10 CFR 60.113(b), provided that it can be shown that such a release rate does not compromise the overall system performance. The second alternative would be to remove carbon-14 from the exterior of the cladding and assembly components by heating (and oxidizing the carbon to carbon dioxide) before emplacement. The carbon-14 could then be dealt with separately from the spent fuel or solidified as calcium carbonate and disposed of in standard containers.

Two alternatives are shown in Figure 8.3.5.10-3 in the event that liquid release from the reference design is not low enough to meet the requirements of 10 CFR 60.113. The first is to take into account other components and processes to limit access of water to the spent fuel, while the second takes into account the possible contribution of the rock in the EBS in limiting release (contingent on an interpretation through a mechanism such as rulemaking that the EBS can include a portion of the host rack).

There are limited quantities of highly soluble radionuclides that are present in the spent fuel waste form at 1,000 yr after closure. These isotopes, primarily Tc-99, I-129, Cs-135, and Mo-93, account for about 0.8 percent of the total 1,000 yr inventory. Under the expected case of the reference approach (no liquid water), these nuclides would not be released from the engineered barrier system because there is no aqueous medium for dissolution and transport. Under the bounding case, in which bare spent fuel is contacted by as much as 20 L of liquid water per container per year for up to 10 percent of the packages, the solubility and availability of the nuclides may result in a short term release from the gap-grain fraction plus an annual release from the matrix fraction at $1 \times 10^{-3}$ per year. The short-term release is reduced to the performance objective ($1 \times 10^{-5}$ per year) by

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scenario-dependent factors (e.g., timing of breach or wetting). The matrix release component is reduced by a factor of ten due to the absence of water and a further factor of ten due to these other barriers (i.e., cladding and container breaches and their mass transfer resistance).

The existing information is not sufficient to allow a final selection of the components. Several components and processes are available to provide barriers to the release of gap and grain boundary radionuclides. These include

1. The rate of breach of the containers that are intact at the end of the containment period.

2. The rate of breach of fuel cladding during the entire period following closure.

3. The fraction of water that contacts and enters breached containers and claddings and contacts the waste form.

4. Dilution of concentration of radionuclides in solution within breached containers before release, in a scenario with standing water in a container.

5. The limited surface area of fuel contacted by water in trickle-through and unsaturated-contact scenario.

Several components and processes are available to reduce the engineered barrier system release rate due to soluble elements released from the waste form matrix. The existing information is not sufficient to allow a final selection of the components and performance measures, but as an overall performance measure, a factor of 10 reduction is assigned. Possible contributing components and processes include

1. Mass transfer rate through breached cladding.

2. Mass transfer rate through breaches in containers, or mass transfer rate along available diffusion pathways of partially saturated rock in contact with partially saturated waste form.

3. Limited surface area of fuel contacted by water.

4. Limited time periods of contact of water with fuel surface.

Tests and analyses to support the basis for allocating performance to these potential barriers are described under Information Needs 1.4.2 through 1.4.4 (Sections 8.3.5.9.2 through 8.3.5.9.4) and 1.5.2 through 1.5.4 in this section.

Another alternative (which applies to both waste forms) would be to include a portion of the host rock as part of the engineered barrier system. This rock is expected to significantly limit the release rate.

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In the event of failure to demonstrate all the previous approaches, an alternative container with considerably greater expected lifetime before breach might be used. A longer-life container could be developed as one alternative that might be considered in the alternate materials and containers program discussed in Section 8.3.5.9.

The strategy for meeting the controlled release requirements of 10 CFR 60.113 is based on the bounding assumption that less than 10 percent of the packages will be contacted by less than 20 L of water per package per year. This value was obtained by multiplying the cross sectional area of a vertical borehole by 80 times the maximum anticipated flux (9 times the flux for a horizontal borehole) passing downward through the cross-sectional area. The strategy applies only to an unsaturated repository.

For the reference approach, a performance allocation has been made to system elements on the basis of both the expected and the bounding case (see Figure 8.3.5.10-2). The performance measures and goals associated with the reference approach are listed in Table 8.3.5.10-2. Releases for various release scenarios are based on values given in the table, which are equal to or less than the regulatory requirement. Control of the quantity of water contacting the container to less than 20 L/yr requires that the engineered environment have several performance measures and related goals. These are further discussed in Issue 1.10 (Section 8.3.4.2) and are not addressed in detail here. There are seven performance goals and measures in the reference case. These apply to the quantity and quality of water contacting waste, the containers, and the waste forms. The performance parameters needed to evaluate some of these performance measures are listed in Table 8.3.5.10-3a. As discussed above, for some other performance measures, existing information is not sufficient to allow selection of components and allocation of the parameters' performance. The table also lists parameters that are consistent with meeting the performance goals of Table 8.3.5.10-2.

The first set of parameters (Table 8.3.5.10-3a) refers to the quality of the water contacting the waste. The goals for the constituents of the water are set so that they are consistent with the composition of well J-13 water as possibly modified by the thermal loading history of the repository (Chapter 4). These goals are used as a basis for setting goals for the other parameters. The information needs to establish both the quantity and quality of the water are discussed in Section 8.3.4.2.

The second and third sets of parameters are given in Table 8.3.5.10-3b. The second set of parameters is a list of the maximum concentrations of radionuclides permissible in effluent solutions exiting the engineered barrier system. The values given for the goals are the concentrations necessary to meet the design objectives of the controlled-release period. The third set of parameters is the analogous information for the glass waste form. These concentrations are for the upper-limit water flux of 20 L/yr contacting waste in 10 percent of the waste packages. For lower water fluxes through different waste packages, correspondingly higher concentrations are allowed. For diffusional contact scenarios, a corresponding limit in curies per package per year is allowed.

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Table 8.3.5.10-2. Performance measures and goals for Issue 1.5 (engineered barrier system release rates)

| System element | Performance measure | Tentative goal[a] | Needed confidence |
|---|---|---|---|
| Engineered environment[b] | Quantity of liquid water that can contact the container | No liquid water contacting waste forms for 90% of packages that may be contacted by negligible quantities of water. For the remaining 10% of the packages, <20 L of liquid water per package per year contacts the waste form | High |
| | | Rate of breached dry packages becoming wet <0.01/yr | High |
| | Water quality | Constrain water chemistry to acceptable levels for waste form performance | High |
| | Rock-induced load on waste package | Load less than design basis (see Table 8.3.4.2-3) | High |

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Table 8.3.5.10-2. Performance measures and goals for Issue 1.5 (engineered barrier system release rates) (continued)

| System element | Performance measure | Tentative goal[a] | Needed confidence |
|---|---|---|---|
| Container | Fraction of containers that have breached[c] | For t >1,000 for containers with no liquid water contact: <0.001/yr | High |
| | | For t >1,000 for containers with liquid water contact: <0.01/yr | High |
| | Fractional release due to mass transfer resistance of breached containers (and cladding) | For t >1,000 for containers with liquid water contact: <0.1 (see Section 8.3.5.10.3) | High |
| Waste Form | Release fractions or rates from waste form components | Release rates from breached packages via all mechanisms, together with the mass transfer resistance of packages, of <1 part in 10,000 (of 1,000 yr inventory) per year for each radionuclide | High |

[a]t = years after repository closure.

[b]Envelope for anticipated processes and events.

[c]Breach is defined as allowing air flow of $1 \times 10^{4}$ atm-$cm^3$/s. The maximum fraction of total failures will be determined as part of the container material studies and will be consistent with regulatory intent.

Table 8.3.5.10-3a. Performance parameters and goals for water composition for Issue 1.5 (engineered barrier system release rates)

| Performance measure | Performance parameters | Tentative performance parameter goal | Needed confidence | Current estimated range | Current confidence |
|---|---|---|---|---|---|
| Water quality[a] | pH | 5.5-9 | High | 6.1-7.7 | Medium |
| | $Cl^-$ | <20 ppm | High | <10 ppm | Medium |
| | $F^-$ | <6 ppm | High | <5.4 ppm | Medium |
| | $NO_3^-$ | <15 ppm | High | 0-11 ppm | Medium |
| | $SO_4^{=}$ | <50 ppm | High | 15-35 ppm | Medium |
| | $CO_3^{=}$, $HCO_3^-$ | <200 ppm | Medium | 90-160 ppm | Medium |
| | Total anions | <220 ppm | Medium | 110-160 ppm | Medium |
| | Organics | TBD[b] | TBD | NA[c] | NA |
| | Colloids | TBD | TBD | NA | NA |
| | $O_2$ | 0.1-8 ppm | High | <6.5 ppm | Medium |
| | $NH_3$ | <1 ppm | High | <1 ppm | Low |
| | Si | >20 ppm | High | 20-550 ppm | Medium |
| | Na | <100 ppm | High | 30-80 ppm | Medium |
| | K | <50 ppm | High | 1-30 ppm | Medium |
| | Na/Ca | >1 | High | >2 | Medium |
| | Total heavy metals (>Fe) | <2 ppm | High | TBD | Low |
| | Total other cations | <50 ppm | Medium | <30 ppm | Low |

[a]Not all combinations of the limits on the goals given in the above table will result in acceptable water chemistries (See Section 8.3.4.2).
[b]TBD = to be determined.
[c]NA = not applicable.



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Table 8.3.5.10-3b. Performance parameters and goals for spent fuel and glass waste forms for Issue 1.5 (engineered barrier system release rates) (page 1 of 8)

| Performance measure | Performance parameter | Performance goal (Concentration of radionuclide in effluent water) (mg/L) | Performance goal (Concentration of radionuclide in effluent water) (Ci/L) (see note a) | Needed confidence | Current estimated range (mg/L) | Current confidence |
|---|---|---|---|---|---|---|
| SPENT FUEL WASTE FORM (see notes b and c) | | | | | | |
| | (see note d) | | | | (see note e) | |
| Release rate from bare waste form inside failed container | C-14[f] | 2.06 | 1.80E-05[g] | High | <40 | Low |
| | Cl-36 | 5.61E-01 | 1.80E-05 | High | To be determined | Low |
| | Ca-41 | 1.65E-01 | 1.80E-05 | High | <20 | Medium |
| | Ni-59 | | 5.28E-05 | High | | |
| | Ni-63 | 127 | 1.80E-05 | High | <4 | Medium |
| | Se-79 | 8.55E-01 | 1.80E-05 | High | <6.0E-3 | Low |
| | Zr-93 | 1900 | 1.99E-05 | High | <1 | High |
| | Nb-93m | | 1.89E-05 | High | | |
| | Nb-94 | 7.32 | 1.80E-05 | High | <1 | High |
| | Mo-93 | 2890 | 1.80E-05 | High | To be determined | Low |
| | Tc-99 | 7.98 | 1.34E-04 | High | <0.8 | Low |
| | Pd-107 | 220 | 1.80E-05 | High | <10 | Low |
| | Sn-126 | 88.1 | 1.80E-05 | High | <.010 | High |
| | I-129 | 110 | 1.80E-05 | High | <0.2 | Low |
| | Cs-135 | 79.3 | 1.80E-05 | High | <1.6 | Low |
| | Sm-151 | 94.9 | 1.80E-05 | High | <1 | High |
| | Ho-166m | 1.00E-02 | 1.80E-05 | High | <1 | Low |
| | Pb-210 | 2.22E-04 | 1.80E-05 | High | <4.0E-8 | Medium |
| | Ra-226 | 1.82E-02 | 1.80E-05 | High | <8.0E-2 | Low |
| | Ac-227 | 2.47E-04 | 1.80E-05 | High | To be determined | Low |



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Table 8.3.5.10-3b. Performance parameters and goals for spent fuel and glass waste forms for Issue 1.5 (engineered barrier system release rates) (page 2 of 8)

| Performance measure | Performance parameter | Performance goal (Concentration of radionuclide in effluent water) (mg/L) | (Ci/L) (see note a) | Needed confidence | Current estimated range (mg/L) | Current confidence |
|---|---|---|---|---|---|---|
| | | SPENT FUEL WASTE FORM (continued) | | | | |
| Release rate from bare waste form inside failed container (continued) | Th-230 | 8.91E-01 | 1.80E-05 | High | <1.0E-08 | |
| | U-233 | | 1.80E-05 | | | |
| | U-234 | | 2.10E-05 | | | |
| | U-235 | 9970 | 1.80E-05 | High | <10 | High |
| | U-236 | | 1.80E-05 | | | |
| | U-238 | | 1.80E-05 | | | |
| | Np-237 | 25.5 | 1.80E-05 | High | To be determined | Medium |
| | Pu-238 | | 1.80E-05 | | | |
| | Pu-239 | | 3.15E-03 | | | |
| | Pu-240 | 77.8 | 4.94E-03 | High | <5.0E-3 | High |
| | Pu-241 | | 1.80E-05 | | | |
| | Pu-242 | | 1.80E-05 | | | |
| | Am-242m | | 1.80E-05 | | | |
| | Am-243 | 8.08E-01 | 1.61E-04 | High | <1.0E-2 | High |
| | Cm-245 | | 1.80E-05 | | | |
| | Cm-246 | 1.28E-01 | 1.80E-05 | High | <1.0E-3 | Medium |



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Table 8.3.5.10-3b. Performance parameters and goals for spent fuel and glass waste forms for Issue 1.5 (engineered barrier system release rates) (page 3 of 8)

| Performance measure | Performance parameter | Performance goal (Concentration of radionuclide in effluent water) (mg/L) | Performance goal (Concentration of radionuclide in effluent water) (Ci/L) (see note b) | Needed confidence | Current estimated range (mg/L) | Current confidence |
|---|---|---|---|---|---|---|
| | | **SPENT FUEL WASTE FORM (see notes b and c) (continued)** | | | | |
| Release rate from bare waste form inside failed container (continued) | Activity of $^{14}CO_2$ released as a gas (gaseous release) | (see note f) | | High | To be determined | Low |
| | | **GLASS WASTE FORM (see notes h and i)** | | | | |
| | Ni-59<br>Ni-63 | <br>1.4E-01 | 1.09E-05<br>8.78E-07 | High | 3.3E-04 to 4.9E-03[j] | Medium |
| | Se-79 | 8.1E-03 | 5.62E-07 | High | 1.9E-05 to 2.9E-04[j] | Medium |
| | Rb-87 | 2.1E+03 | 1.76E-07 | Low | 1.1E-03 to 1.6E-01[k] | Medium |
| | Zr-93 | 2.4E-00 | 6.22E-06 | High | 5.8E-03 to 8.8E-01[j] | High |
| | Nb-93m<br>Nb-94 | 9.5E-04 | 6.22E-06<br>1.76E-07 | High | 2.3E-06 to 3.4E-05[k] | Medium |



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Table 8.3.5.10-3b. Performance parameters and goals for spent fuel and glass waste forms for Issue 1.5 (engineered barrier system release rates) (page 4 of 8)

| Performance measure | Performance parameter | Performance goal (Concentration of radionuclide in effluent water) (mg/L) | (Ci/L) (see note a) | Needed confidence | Current estimated range (mg/L) | Current confidence |
|---|---|---|---|---|---|---|
| | | GLASS WASTE FORM (continued) | | | | |
| Release rate from bare waste form inside failed container (continued) | Tc-99 | 6.1E-01 | 1.03E-05 | High | 1.5E-03 to 2.2E-01[k] | Low |
| | Pd-107 | 3.4E-01 | 1.76E-07 | Medium | 2.4E-04 to 3.6E-03[k] | Medium |
| | Sn-126 | 2.9E-02 | 8.22E-07 | High | 7.0E-05 to 1.0E-03[j] | Medium |
| | Cs-135 | 5.3E-01 | 4.66E-07 | High | 1.3E-03 to 1.9E-01[j] | Medium |
| | Sm-151 | 1.5E-05 | 3.83E-07 | High | 3.5E-08 to 5.3E-07[j] | Medium |
| | Pb-210 | 2.3E-06 | 1.76E-07 | Medium | 1.9E-10 to 2.8E-09[k] | Medium |
| | Ra-226 | 1.8E-04 | 1.76E-07 | Medium | 1.9E-08 to 2.9E-07[k] | Medium |
| | Ac-227 | 2.4E-06 | 1.76E-07 | Low | 4.7E-12 to 7.1E-11[k] | Low |





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Table 8.3.5.10-3b. Performance parameters and goals for spent fuel and glass waste forms for Issue 1.5 (engineered barrier system release rates) (page 5 of 8)

| Performance measure | Performance parameter | Performance goal (Concentration of radionuclide in effluent water) (mg/L) | Performance goal (Concentration of radionuclide in effluent water) (Ci/L) (see note a) | Needed confidence | Current estimated range (mg/L) | Current confidence |
|---|---|---|---|---|---|---|
| GLASS WASTE FORM (continued) | | | | | | |
| Releaase rate from bare waste form inside failed container (continued) | Th-230 | 8.6E-03 | 1.76E-07 | Medium | 8.5E-10 to 8.5E-09[l] | High |
| | Pa-231 | 3.7E-03 | 1.76E-07 | Low | 9.5E-09 to 1.4E-07[k] | Low |
| | U-232 | | 1.76E-07 | | | |
| | U-233 | | 1.76E-07 | | | |
| | U-234 | 8.5E+01 | 4.54E-06 | High | 2.6E-01 to 2.6E-01[l] | Medium |
| | U-235 | | 1.76E-07 | | | |
| | U-236 | | 1.85E-07 | | | |
| | U-238 | | 1.76E-07 | | | |
| | Np-237 | 2.5E-01 | 1.76E-07 | Medium | 3.2E-04 to 4.8E-03[k] | Medium |
| | Pu-238 | | 1.20E-06 | | | |
| | Pu-239 | | 5.14E-05 | | | |
| | Pu-240 | 1.0E-00 | 3.00E-05 | High | 1.2E-06 to 1.2E-05[l] | Low[m] |
| | Pu-241 | | 1.76E-07 | | | |
| | Pu-242 | | 1.76E-07 | | | |





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Table 8.3.5.10-3b. Performance parameters and goals for spent fuel and glass waste forms for Issue 1.5 (engineered barrier system release rates) (page 6 of 8)

| Performance measure | Performance parameter | Performance goal (Concentration of radionuclide in effluent water) (mg/L) | (Ci/L) (see note a) | Needed confidence | Current estimated range (mg/L) | Current confidence |
|---|---|---|---|---|---|---|
| | | | GLASS WASTE FORM (continued) | | | |
| Release rate from bare waste form inside failed container (continued) | Am-241 | | 5.01E-05 | | | |
| | Am-242m | 1.5E-02 | 1.76E-07 | High | 3.7E-05 to 5.6E-04[j] | Low[m] |
| | Am-243 | | 1.76E-07 | | | |
| | Cm-245 | 1.0E-03 | 1.76E-07 | Low | 4.7E-10 to 7.1E-09[k] | Low |

[a]The concentrations are derived from the 1 x $10^{-5}$ per year or 0.1% calculated release rate limit (CRRL) (1.8 x $10^{-6}$) requirement for each individual radioisotope based on 20 liters per package per year flux for up to 10 percent of the packages together with the package loading assumptions in notes c and h.

[b]Limiting concentrations include stable isotopes of an element and were calculated assuming that all isotopes of an element are released congruently at a level determined by the limiting concentration of the radioisotope of that element requiring the most stringent control.

[c]All calculations based on 33,000 MWd/MTU fuel at 1,000 yr out-of-reactor. Inventory includes cladding and hardware. Calculations assume 62,000 MT of unoxidized spent fuel in 30,000 containers of which 10 percent are contacted by 20 liters of liquid water per year. Issue 1.4 allocates performance to the cladding in order to limit the quantity of oxidized fuel to less than 1 percent of the repository inventory, thereby controlling the release of those radionuclides in the fuel that are made more available for aqueous release by oxidation (e.g., Tc-99).

[d]Table includes all radionuclides that have half-lives greater than 10 yr and have total inventories per package such that, at the allowed release rate, it would take more than 10 yr to release the entire inventory.



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Table 8.3.5.10-3b. Performance parameters and goals for spent fuel and glass waste forms for Issue 1.5 (engineered barrier system release rates) (page 7 of 8)

Footnotes (continued)

[e]Current estimated ranges are based on experimental results discussed in Chapter 7, Section 7.4.3.1.1, and theoretical modeling of phase solubility. Ranges for Se, Tc, Pd, I, Cs, and Pb were estimated assuming that 1 g of U reacts per liter of water entering a container. The inventory of these elements associated with 1 g of U was then assumed to remain in solution. Note that 1 g of U per liter is far in excess of the expected U solubility.

[f]The allowed aqueous concentration of C-14 assumes that no C-14 is released as a gas. Similarly, the allowed gaseous release was calculated assuming no C-14 is released in solution. To meet the actual release requirements, the sum of the aqueous plus gaseous release must total <3.6E-05 Ci/yr per package.

[g]1.80E-05 is notation for $1.8 \times 10^{-5}$.

[h]Allowed maximum concentration in mg/L for all the radioisotopes of each element. Note that nonradioactive isotopes are not included.

[i]Inventory data for Defense Waste Processing Facility (DWPF) glass taken from 1,000-year inventory, Table 7-21. Allowed effluent per container in the maximum 20 liters water per package per year for 10 percent of the packages is calculated from the allowed release; 1 part in 100,000 or 0.1% of the calculated release rate limit (CRRL). Radionuclides whose total inventory could be released from the waste package at the allowed rate, in less than 10 yr, have been excluded from this table (e.g., Sr-90). Radionuclides whose half-lives are less than 10 yr (short-lived daughter products) have also been excluded; they are controlled by controlling the parent nuclide.

[j]Estimated range in concentration based on the congruent breakdown of glass. The allowed total for each radionuclide (note i) is equivalent to a silica concentration in solution of 4,150 mg/L (approximately 50 percent of the 1,660 kg glass in each DWPF container is silica; one part in 100,000, times the 10 percent of the containers that are contacted by water, partitioned into 20 liters, is 4,150 mg/L). Silica releases of this magnitude are not anticipated. Estimated ranges were obtained by considering that total silica released from glass (including that recrystallized) would not exceed that equivalent to 150 mg/L. This is the upper limit in the estimated range. The lower limit assumes that glass dissolves slowly at long times, and an equivalent silica release of 10 mg/L was used.

[k]For these radionuclides, the allowed release is 0.1 percent of the CRRL. However, the estimated release reflects the actual inventory (see Table 7-21, 1,000-yr inventory), which may be much smaller.

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Table 8.3.5.10-3b. Performance parameters and goals for spent fuel and glass waste forms for Issue 1.5 (engineered barrier system release rates) (page 8 of 8)

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Footnotes (continued)

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[l]For these radionuclides, the estimated concentrations based on either (j) or (k) exceeded the expected solubility of these elements in well J-13 water, as calculated using EQ3/6. The estimated range given is the calculated solubility at 25C, pH 7.6, with 1 order of magnitude uncertainty.

[m]Although these radionuclides are at or near the current predicted solubility limit, the current confidence is given as low because of the possibility that additional solution species (other ligands) may be found that raise the solubility, and the possibility that colloid transport may contribute significantly to release.



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The maximum radionuclide concentrations per liter of effluent (or per package per year) are based on release requirements that yield limits for the sums from all contributing modes of waste release. For spent fuel these sums include rapid fractional releases due to container breaches or to breached dry containers becoming wet, plus the gradual releases per year from the accumulated number of breached and wet containers. Performance goals for the rates of container failures and wetting were listed in Table 8.3.5.10-2. The performance measures and goals for the waste form components are listed in Table 8.3.5.10-3c.

Releases for various failure scenarios are based on rates given in the performance allocation tables.

The performance analyses linking the allocated values of the performance parameters to the higher-level measure of performance will consider several possible scenarios of component conditions and water contact modes. The container will be assumed to have failed at one or more locations, but the bulk of the container body has a wall thickness greater than one-tenth of the initial thickness, and remains structurally intact. The failure locations may allow the following water contact modes:

1. Water accumulates inside the package up to the level of the lowest breach and exits the package.

2. Water trickles or seeps through the container and drains out of a low-elevation breach.

3. A water contact without moving water exists between the partially saturated rock, container and corrosion product surfaces, and waste form, due to capillarity in these components.

No air-tight sealed condition is assumed for any spent fuel cladding or glass waste pour canisters.

The data in hand are insufficient to choose with high confidence of success a final licensing approach for rapid gaseous release of carbon-14. Several alternatives are under study (Figure 8.3.5.10-3). The development of a better understanding of the distribution and release characteristics of carbon-14 in Zircaloy cladding and assembly hardware is needed, since the relatively large release observed in the fuel temperature test may be due to the high temperature or the high radiation field or both. Another area of investigation is the breach rate of containers, since a time-distributed failure rate for containers would minimize the pulsed carbon-14 release. The interaction of carbon-14 releases from waste packages with natural carbon in the repository air and rock-water system will also be studied (Information Need 1.5.5, Section 8.3.5.10.5). Although these items are not part of the reference case, they are discussed along with the reference case items with which they are associated.

To satisfy the needs for information to be used in resolution of Issues 1.1 (Section 8.3.5.13) and 1.9 (Section 8.3.5.18), a more realistic estimate of release rates and total releases is needed. Some of the system components that could provide additional control on the release rate have not been included in the reference case calculations. Inclusion of an analysis

Table 8.3.5.10-3c. Performance parameters and goals for components of spent fuel waste for Issue 1.5 (engineered barrier system release rates)

| Performance measure | Performance parameter | Tentative goals | Needed confidence | Current estimated range | Current confidence |
|---|---|---|---|---|---|
| Release fractions or rates from components waste form | Fraction of total inventory of gap and grain boundary elements available for rapid release from unoxidized fuel | < 0.02 | High | 0.005 to 0.04 | Medium |
| | Fraction of C-14 inventory available for rapid release as a gas under temperatures prevailing after 1,000 yr | < 0.001 | High | <0.002 | Low |
| | Fraction of soluble matrix radionuclides releasable to water within waste package | < 0.001/yr | High | 0.0001 to 0.002/yr | Medium |
| | Fraction of other radionuclides releasable to water within waste package | $< 1 \times 10^{-5}$/yr | High | $<1 \times 10^{-5}$/yr | Medium |





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of the condition of the containers would allow credit to be taken for intact containers. Zircaloy or stainless steel cladding, either intact or with minor defects, will provide an additional control on the rate of dissolution of the pellets contained within the cladding. All these factors would result in lower estimates of the amount of waste elements leaving the container and the engineered barrier system. The Yucca Mountain Project adopted the current DOE interpretation of the EBS system boundary to coincide with the surfaces of the excavations within the underground facility. The DOE, however, requires the Project to reevaluate the interpretation before the completion of repository and waste package advanced conceptual design. If, in the future, portions of the host rock are to be included in the EBS, the near-field radionuclide transport studies will be needed to resolve this issue (1.5) and to provide the realistic source term to Issues 1.1 and 1.9.

Radionuclide source term calculations will examine the transport processes active in the first few meters of host rock surrounding an emplaced waste package. These calculations are required to provide detailed information on the anticipated response of the hydrogeologic and geochemical systems to the maximum design thermal loading, and to provide a basis for the assessment of the effectiveness of natural and engineered barriers against the release of radioactive materials to the environment (10 CFR 60.21). The release to the accessible environment will be calculated in activities described under Issue 1.1 (Section 8.3.5.13). A realistic source term will serve as a basis for establishing bounding conditions, and for demonstrating that predicted performance under those conditions is bounding. For radionuclide transport, many species exhibit a strong affinity for sorption onto the host rock. Under anticipated conditions, it is expected that these species will interact with rock that is within meters of the waste package, as opposed to hundreds or thousands of meters from the repository. Therefore, a radionuclide source term calculated across a boundary relatively near the waste package will serve as a realistic, although not necessarily bounding, source term for transport calculations to the accessible environment.

Table 8.3.5.10-4 presents the performance measure for this activity. The measure is the relative concentrations of radionuclide species as a function of time and distance from an emplacement hole that are adsorbed to host rock, dissolved in pore and fracture water, and in the pore and fracture gases. This activity will provide characterization of the effectiveness of the host rock against radionuclide transport. The parameters required for these assessments are host rock hydrologic properties, thermal properties, transport properties, and radionuclide sorption and exchange properties. Further, release of radionuclides from the engineered barrier system and a set of conditions representing anticipated and unanticipated processes and events are required. Table 8.3.5.10-5 provides the linkages to those parameters developed in other sections of this document. The parameters of Table 8.3.5.10-5 are developed in greater detail in Sections 8.3.4 and 8.3.5.13.

To ensure that the testing program and analyses would provide the information needed to resolve this issue and to support the resolution of Issues 1.1 (Section 8.3.5.13) and 1.9 (Section 8.3.5.18), characterization goals were set for the description of the waste form in its as-received condition,

Table 8.3.5.10-4. Performance allocation for radionuclide migration in near-field host rock

| SCP section requesting parameter | System element | Function | Process or condition | Performance measure | Goal | Needed confidence |
|---|---|---|---|---|---|---|
| 8.3.5.13 | Topopah Spring tuff | Limit migration of radionuclides through the near-field host rock | Radionuclide transport | Concentrations of radionuclide species in gas phase, liquid water, and adsorbed to solid phases within the near-field host rock | Adequate to determine effectiveness of natural barriers | High |



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Table 8.3.5.10-5. Performance measures, parameters, and parameter goals for calculating radionuclide source term for near-field host rock (page 1 of 2)

| System element | Performance measure | Parameter | Parameter goal | Current confidence | Needed confidence | Sections where parameters are developed |
|---|---|---|---|---|---|---|
| Topopah Spring tuff | Concentrations of radionuclide species in gas phase, liquid water, and adsorbed to solid phases within the near-field host rock | Host rock hydrologic properties | Properties known with accuracy sufficient to calculate differences in flow through the near-field rock resulting from anticipated and unanticipated events | Low | High | 8.3.1.2.2 |
| | | Radionuclide sorption properties | Properties known with accuracy sufficient to calculate radionuclide sorption to the near-field rock resulting from anticipated and unanticipated events | Low | High | 8.3.1.3.4 |
| | | Radionuclide transport properties | Properties known with accuracy sufficient to calculate transport through the near-field rock resulting from anticipated and unanticipated events | Low | High | 8.3.1.3.1, 8.3.1.3.4, 8.3.1.3.5, 8.3.1.3.6, 8.3.1.3.7 |
| | | Host rock thermal properties | Properties known with accuracy sufficient to calculate heat flow and temperature in the near-field rock resulting from anticipated and unanticipated events | Medium | High | 8.3.1.15.1, 8.3.1.15.2, 8.3.4.2.4 |

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Table 8.3.5.10-5. Performance measures, parameters, and parameter goals for calculating radionuclide source term for near-field host rock (page 2 of 2)

| System element | Performance measure | Parameter | Parameter goal | Current confidence | Needed confidence | Sections where parameters are developed |
|---|---|---|---|---|---|---|
| Topopah Spring tuff (continued) | Concentrations of radionuclide species (continued) | Releases from engineered barrier system | Knowledge of the engineered barrier system release rate | Medium | High | 8.3.5.10.4 |
| | | Anticipated and unanticipated processes and events | Events described in detail sufficient for resolution of Issues 1.1 and 1.5 (Section 8.3.5.13 and this section) | Medium | High | 8.3.5.10.3 |



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the handling and storage of the waste form before sealing it in a container, and the characterization of the physical and chemical processes that could affect radionuclide release rates. These topics are discussed briefly in the following paragraphs.

A. Waste form definition

The characteristics of the waste forms when they are received at the repository must be known to ensure proper handling, interim storage, packaging, and disposal conditions. A number of the characteristics are interrelated, such as radionuclide inventory, burnup, and waste age for spent fuel. In this instance, a characterization goal was set for one of these parameters: the radionuclide inventory. This parameter was selected because it is likely to be the least well known, and its value is used in the greatest number of cases in the performance analyses.

Data will need to be collected to allow description of a number of the characteristics of spent fuel. At present, there is insufficient information about the variability of the waste form, and the sensitivity of waste form performance to that variability, to allow a sensible goal for characterization to be set. Where this is the case, the most complete characterization consistent with the resources for information will be provided. As information on the relative importance of the various parameters becomes available, more precise characterization goals will be set.

There are two parameters known to be important with respect to spent fuel performance under disposal conditions: elements that migrate as gases during use in the reactor and the population of cladding that contains defects. The former is needed to allow prediction of the rapid release fraction of a small number of radionuclides (isotopes of cesium, iodine, and technetium); the latter is needed to allow estimation of the number of fuel rods for which water can immediately gain access to the spent fuel. For each of these parameters, a characterization goal is assigned. This information is required for resolution of both Issue 1.4 and this issue.

For glass waste forms, the characterization goals will be given in the waste acceptance specifications. The specifications relevant to resolution of Issue 1.5 given in the waste acceptance preliminary specifications (WAPS) (Stein, 1988) are as follows:

Specification 1.1 -- Chemical composition, requiring the producer to provide sufficient chemical and microstructural information necessary to characterize the elemental composition and crystalline phases for the product glass and expected variations in these characteristics during the life of the production facility.

Specification 1.2 -- Radionuclide inventory, requiring the producer to provide estimates of the total radionuclide inventory to be sent to the repository, estimates of the radionuclide concentration in each canister, and expected variations in these quantities during the life of the production facility.

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Specification 1.3 -- Leaching properties, requiring the producer to control the leaching characteristics of the glass waste form during production such that the normalized release rates for sodium, silicon, boron, cesium-137, and uranium-238 in a 28 day MCC-1 leach test in deionized water do not exceed one gram per square meter per day averaged over the duration of the test.

Specification 1.4 -- Chemical and phase stability, requiring the producer to provide glass transition temperatures and time-temperature-transformation data necessary to define the duration at any specific temperature which causes significant changes in the microstructure or phase compositions of the glass waste forms within the anticipated range of compositions.

WAPS specifications 1.1 and 1.2 allow the selection of input data (e.g., thermodynamic and kinetic properties) for glass waste form degradation models. Such data are largely a function of glass composition. The range of expected glass compositions must be known i order to guide the development and application of a glass properties degradation data base (Activity 1.5.2.2.2).

Specifications 1.3 and 1.4 are necessary to limit the classes of models that must be developed to resolute Issue 1.5. These specifications ensure that the glass waste form sent to the repository by the producer is, on the basis of durability and microstructure, similar to those glasses used in developing glass degradation models and a glass properties degradation data base for repository specific release rate predictions. The leach rates referenced in specification 1.3 are not intended to be a measure of the glass waste form performance in the repository or to act as a source term for the performance of the engineered barrier system. This specification in intended to discriminate between well-made glasses and nonvitreous products that may result from variations in process feed composition, process upsets during vitrification, and/or post-vitrification handling.

The goals for spent fuel description are as follows:

1. The inventory of radionuclides at emplacement will be established to within <20 percent for each radionuclide that will constitute more than 5 percent of the activity at any time during the first 10,000 yr after disposal.

2. The condition of the cladding will be described so that the number of rods containing defects in the cladding at the time the waste package is assembled can be estimated to within a factor of 2, or be shown to be less than 1 percent of the population.

3. The fission gas release to the pellet-cladding gap will be determined so that the gap inventory of cesium can be estimated to within a factor of 5.

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B. Postacceptance, pre-emplacement storage, and handling of waste forms

Certain storage and handling conditions can cause changes in the waste forms that would be detrimental to long-term performance under disposal conditions. To prevent the occurrence of those conditions, goals have been set on the handling and storage of the waste forms after receipt at the repository. These goals have been assigned on the basis of the current knowledge of waste properties and taking into consideration the present understanding of the relative importance of factors affecting performance. The goals on the number of preemplacement cladding failures are motivated by requirements on the amount of fuel that can be allowed to oxidize. Goals for cladding failures are also set under Issue 1.4.

The goals for handling and storage conditions are as follows:

1. The temperature of the spent fuel waste form and the access of air to the waste form will be controlled during transport, handling, and storage before emplacement such that oxidation of spent fuel through existing cladding defects is less than the amount that would result in 5 percent cladding strain.

2. The processes used to transport and handle spent fuel at the surface handling facility will be designed so that cladding failure from mechanical abrasion or deformation considering thermally induced effects will result in less than 5 percent cladding strain.

3. For glass waste forms, the storage conditions will be such that the transition temperature of the glass is not exceeded.

Analyses to define the temperature and air access limits required under goals 1, 2, and 3 are conducted under Information Need 1.5.2. Analyses conducted to resolve Issue 4.4 (Section 8.3.2.5) will show that the surface handling facility will comply with the temperature and air access goals determined for the spent fuel and glass waste forms.

C. Chemistry of water that enters the failed containers

Waste form dissolution rates and the solubility of mobilized radionuclides can be sensitive functions of the chemistry of the water that contacts the waste forms. The chemistry of water that contacts the containers and alterations to the water chemistry due to container corrosion will be determined in the resolution of Issue 1.10 under Information Need 1.10.4. (Section 8.3.4.2.4). The chemistry of water that could enter failed containers at a rate greater than 0.5 L/yr will be characterized to within the following limits.

| | |
|---|---|
| pH | ±1 pH unit |
| Anions: | ±1 mg/L for fluoride, chloride, and phosphate |
| | ±10 mg/L for nitrate and sulfate |
| | ±30 mg/L for carbonate and bicarbonate |

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Cations: ±1 mg/L for species originally present at less than 6 mg/L. (Nickel and chromium are excluded from this requirement.)
±5 mg/L for species originally present at between 6 and 40 mg/L
±20 mg/L for species originally present at greater than 40 mg/L.

D. Dissolution rate of the components of the waste forms and solubility of mobilized radionuclides

The long-term dissolution rate of the glass waste form is expected to be controlled by saturation-limited kinetics; as the solution in contact with the glass waste form approaches saturation with amorphous silica, the rate of dissolution is expected to drop to very low values. A model for glass dissolution by this mechanism has been developed and appears to yield a reasonable fit to the laboratory data (Grambow, 1984; Grambow et al., 1985, 1987). The long-term dissolution rate of spent fuel ($UO_2$) is also expected to follow a kinetic rate law; however, the available experimental evidence suggests that the forward dissolution rate of $UO_2$ under oxidizing conditions does not approach zero when the solution in contact with the waste reaches saturation with respect to secondary uranium-bearing phases.

For both waste forms, for any radionuclide shown to have a dissolution or mobilization rate greater than 1 part in 100,000 per year under the conditions given in the bounding case in Figure 8.3.5.10-2, the solubility and speciation of that radionuclide under anticipated conditions will be determined. in this case, the solubility of the radionuclide combined with the low water flow rate will act to limit the release rate of the radionuclide.

The spent fuel waste form is more complicated than the glass waste form because it has a number of components, each with different release or dissolution rates, for which account must be given. Note that complexity of description implies neither inferiority nor superiority of the waste form in terms of the ultimate performance that will be demonstrated.

The performance goal for the spent fuel waste form is to show that the sum of the radioactivity for the solutions and gases exiting the waste packages will contain no more than one part in 100,000 per year of the inventory of each radionuclide present in the total repository 1,000 yr after closure.

For glass waste forms, the performance goal is to show that the dissolution rate of the matrix and the mobilization of elements from the matrix will be low enough to ensure that water exiting a failed container will carry with it no more than 1 part in 100,000 per year of the container inventory of total radionuclides.

Tests and analyses to show that these goals are achieved will be conducted under Information Needs 1.5.1, 1.5.2, and 1.5.3 (Sections 8.3.5.10.1 through 8.3.5.10.3)

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E. Additional barriers available to be used to resolve this issue

The following additional barriers are available to resolve the issue, if needed, under the reference approach for liquid releases.

1. The container failure rate under anticipated conditions and under unanticipated conditions will be described. Tests and analyses to provide estimates of the container failure rate will be done under Information Needs 1.4.2 through 1.4.4 (Sections 8.3.5.9.2 through 8.3.5.9.4).

2. Cladding failure rate will be determined from the results of tests and analyses done under Information Needs 1.5.2 and 1.5.3 (Sections 8.3.5.10.2 and 8.3.5.10.3).

3. For water that encounters a breach in the container (and cladding), the fraction of water that enters the container (and cladding) and fraction that passes by the breach site without entering the container (and cladding) will be characterized. Tests to provide data for this analysis will be done under Information Need 1.5.3 (Section 8.3.5.10.3).

4. Water that accumulates within a failed container will provide variable dilution factors for different radionuclides. The concentration of readily soluble radionuclides due to the rapid release fraction will be diluted in proportion to the quantity of water that accumulates. This will have the effect of reducing the release rate from the container for these nuclides. Tests and analysis to support estimation of the dilution factors will be done under Information Need 1.5.3 (Section 8.3.5.10.3).

F. Potential barriers that will not be characterized

The following potential barriers will not be characterized:

1. The flow of air into a container for which a breach sufficiently large to sustain a flow of air at $1 \times 10^{-4}$ atm-$cm^3$/s will be assumed to proceed without impediment.

2. The pour canister on the glass waste form will be assumed to provide no barrier to fluid flow.

G. Transport of radionuclide-bearing solutions through the near-field environment

The system model for performance assessment will require a source term to represent the radionuclides released across some boundary in the repository and to help provide an assessment of the effectiveness of natural and engineered barriers against release of radionuclides to the environment. To accommodate the needs of the system model for a source term, tests and analyses will be conducted to show the effects of transport of solutions that leave the waste package and migrate through the near-field environment. These tests and analyses will be done under Information Need 1.5.5 (Section 8.3.5.10.5).

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No specific goals will be set for the results of these analyses; however, emphasis will be placed on actinides for which the EPA release limits to the accessible environment are a small fraction of the amount that could be released from the engineered barrier system under the performance objective for radionuclide release rate of 10 CFR 60.113. Data will be gathered predominantly for plutonium and americium (Oversby, 1986).

Interrelationships of information needs

Information Needs 1.5.1 and 1.5.2, and parts of 1.5.3 and 1.5.4 will be used for resolving Issue 1.4 and this issue (1.5). The data from Information Needs 1.5.3 and 1.5.4 not used for this issue (1.5) and all of Information Need 1.5.5 will be used in the resolution of Issues 1.1, 1.4, and 1.9, which are addressed in Sections 8.3.5.13, 8.3.5.9 and 8.3.5.18, respectively.

## 8.3.5.10.1 Information Need 1.5.1: Waste package design features that affect the rate of radionuclide release

Technical basis for addressing the information need

This information need addresses the condition of the waste (spent fuel or glass) as it arrives at the repository, and the Yucca Mountain Project waste package design features important to determining radionuclide release. To model the performance of the waste forms under repository conditions, reliable data are required on the population statistics for the parameters listed in the following parameters section.

Link to the technical data chapters and applicable support documents

The characteristics of the waste form are discussed in Chapter 7, Section 7.4.3. Glass waste forms will be further described in the waste qualification report from the waste producer. The waste container design is described in Section 7.3, and the behavior of the metal barrier components of the waste package are discussed in Section 7.4.2.

Parameters

For the spent fuel waste form, parameters are required for the fuel itself, the fuel cladding, and other assembly parts.

The fuel parameters are as follows:

1. As-fabricated fuel characteristics (composition, density, etc.).

2. Peak and average burnup.

3. Radionuclide inventory.

4. Peak linear heat generation rate (LHGR).

5. Reactor type: pressurized water reactor (PWR), boiling water reactor (BWR), or other.

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6. Fission gas release.

7. Microstructural changes in the fuel due to irradiation.

8. Discharge date.

9. Storage medium and access of air or water to the fuel.

10. Mean and peak storage temperature.

11. Pre-emplacement releases of radionuclides, if any.

The cladding parameters are as follows:

1. Chemical composition.

2. Rod pressurization.

3. Percentage of rods with defected cladding, types of defects, and circumstances under which failure occurred.

4. Degree of oxidation.

5. Amount and type of crud deposits.

6. Radionuclide inventory of cladding.

7. Degree of hydrogen embrittlement or hydride formation, if known.

8. Peak and average storage temperature.

9. Discharge date.

10. Degree of mechanical damage to cladding that does not result in immediate cladding failure.

11. Preemplacement releases of radionuclides from cladding or cladding deposits, if any.

The parameters required for other assembly parts are

1. Chemical composition.

2. Location in assembly.

3. Discharge date.

4. Chemical or physical changes in assembly components due to irradiation or storage.

5. Preemplacement releases of radionuclides from assembly components, if any.

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The parameters for glass waste forms are

1. Chemical composition.
2. Radionuclide inventory.
3. Chemical and phase stability.
4. Pour canister design.
5. Pour canister material.
6. Pour canister material properties.
7. Pour canister closure data.
8. Content of free liquids.
9. Gas content in canister voids.
10. Explosive, pyrophoric, or combustible material content.
11. Organic material content.
12. Free volume.
13. Decay heat generation rate.
14. Radiation dose rates.
15. Chemical compatibility of waste form with pour canister.
16. Weight of glass.
17. Cracking and fine particle production.
18. Chemical compatibility of pour canister and container.
19. Shipment, storage, and repository handling thermal history.

The parameters for both waste forms are

1. Timing of delivery of the various waste types to the repository.
2. Container design.
3. Container materials.
4. Chemical compatibility of waste forms with container.
5. Container orientation.
6. Borehole liner design.
7. Borehole liner materials.
8. Compatibility of waste form with borehole liners.
9. Borehole liner corrosion rate.
10. Borehole liner corrosion products.
11. Borehole shield plug design.
12. Borehole shield plug materials.
13. Compatibility of waste forms with borehole shield plugs.
14. Alteration or corrosion products of borehole shield plugs.
15. Repository thermal loading.
16. Package thermal cycle in repository.

Logic

The parameters just listed will provide a complete description of the waste as emplaced in the repository and provide the data to determine how the waste characteristics will change during the lifetime of the repository.

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8.3.5.10.1.1 Activity 1.5.1.1: Integrate waste form data and waste package design data

This activity accumulates the information in the parameters listed previously from waste producers, fuel manufacturers, and other repository studies. No tests or analyses are performed in this activity.

8.3.5.10.1.1.1 Subactivity 1.5.1.1.1: Integrate spent fuel information

This subactivity will involve participation in the Spent Fuel Working Group, liaison activities with the DOE Office of Storage and Transportation Systems and other groups that may provide data on spent fuel, and review and accumulation of spent fuel data and results to determine whether information specified in the parameters listed previously is adequately provided by producers and other repository studies.

8.3.5.10.1.1.2 Subactivity 1.5.1.1.2: Integrate glass waste form information

This subactivity will involve participation in the waste acceptance process; liaison activities with West Valley Demonstration Project (WVDP), Savannah River Laboratory (SRL), and the Defense Waste Processing Facility (DWPF); and review and accumulation of glass waste form data and results to determine whether information specified in the parameters listed previously is adequately provided by producers and other repository studies. Most of the information specified in the glass waste form parameters is expected to be provided in the Waste Qualification Report as part of the waste acceptance process. The major goal of this activity is to ensure that the needed data are provided.

8.3.5.10.1.1.3 Subactivity 1.5.1.1.3: Integrate waste package and repository design information

This subactivity will involve review and accumulation of data other than that provided by waste producers and the Yucca Mountain Project waste form studies, including the parameters common to spent fuel and glass waste forms in the list given earlier.

8.3.5.10.2 Information Need 1.5.2: Material properties of the waste form

Technical basis for addressing the information need

This information need covers the experimental work carried out to determine the material properties of the spent fuel and glass waste forms and to assess how these properties would affect the behavior of the waste forms under the Yucca Mountain Project repository conditions. The data generated

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by these activities will be used under Information Need 1.5.3 (Section 8.3.5.10.3) to develop models for the long-term performance of the waste forms.

Link to the technical data chapters and applicable support documents

The available data on spent fuel dissolution are discussed in Section 7.4.3.1.1, and data on the degradation and leaching of the glass waste form in Section 7.4.3.2. The available data on the oxidation of irradiated uranium dioxide ($UO_2$) fuel are discussed in Section 7.4.3.1.2. The available data on the corrosion of Zircaloy are discussed in Section 7.4.3.1.3.

Parameters

The information needed from other information needs includes the following:

1. Waste form characteristics and waste package design features (Information Need 1.5.1, Section 8.3.5.10.1).

2. Chemistry of the water contacting the waste form (Information Need 1.10.4, Section 8.3.4.2.4).

3. Temperature as a function of time (Information Need 1.10.4, Section 8.3.4.2.4).

4. Release scenarios (Information Need 1.5.3, Section 8.3.5.10.3).

The information that will be obtained in this information need includes the following:

1. Release rate of radionuclides from the spent fuel waste form (includes both fuel and nonfuel components).

2. Mechanisms of release from spent fuel.

3. Oxidation rate of spent fuel as a function of temperature.

4. Primary mechanisms and rates of Zircaloy cladding failure.

5. Release rate of radionuclides from the glass waste forms.

6. Mechanisms of release from the glass waste forms.

Logic

The parameters given in the preceding list define the material properties of the waste forms that will determine their performance in the repository.

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The primary mechanism for the transport of radioactivity from a failed waste package is dissolution of the waste form into ground water followed by migration due to the natural flow of ground water. It is thus important to determine both the release rate of the radionuclides of interest from the waste form as a function of time as well as the equilibrium solubilities of these elements in ground water of appropriate composition.

Because the spent fuel waste form in a failed container may be exposed to the oxygen in air for a period of time before its initial contact with ground water, it is necessary to determine the oxidation rate of uranium dioxide and the effect of oxidation on dissolution. In addition, the volume change attendant upon the conversion of $UO_2$ to $U_3O_8$ may cause gross failure of the Zircaloy cladding on a fuel rod with preexisting minor cladding defects. This would expose a much greater area of fuel to both oxygen and ground water than would be the case for an essentially intact fuel rod and would affect both the oxidation rate and the dissolution rate. This affects the resolution of both this issue and Issue 1.4 (Section 8.3.5.9).

The Zircaloy cladding in the spent fuel waste form may provide a barrier for the release of radionuclides, especially those elements present in the rapidly released, gap and grain boundary inventory such as cesium and iodine. The corrosion rate of Zircaloy will be studied to determine the effectiveness of the cladding in retarding the release of radionuclides. The nonfuel components of the spent fuel waste form (including cladding) that contain activation products will also contribute to the radionuclide inventory of the repository. Corrosion of these assembly parts is likely to be a major source of several radionuclides (nickel-59, niobium-94, carbon-14).

Radionuclide release from glass waste forms can only occur after breach of both the container and the pour canister, and subsequent entry of water.

### 8.3.5.10.2.1 Activity 1.5.2.1: Characterization of the spent fuel waste form

The purpose of this activity is to conduct tests that will provide data on the release rate of radionuclides from the spent fuel waste form. In all, this activity consists of six subactivities; however, the bulk of the experimental effort is covered by the first three subactivities discussed.

#### 8.3.5.10.2.1.1 Subactivity 1.5.2.1.1: Dissolution and leaching of spent fuel

Objectives

The objective of this subactivity is to determine the release rate of radionuclides from spent $UO_2$ fuel. Tests will be conducted to determine the effect on the release rate of the parameters in the following list. The results of these tests will be used to develop models of spent fuel dissolution and radionuclide release under Information Need 1.5.3 (Section 8.3.5.10.3).

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Parameters

Information is required for the following parameters:

1. Fuel burnup.

2. Fission gas release of the fuel during reactor operation.

3. Temperature.

4. Oxidation state of the uranium in the spent fuel.

5. Water chemistry.

6. Reactor type: pressurized water reactor (PWR), boiling water reactor (BWR), or other.

7. Grain size of the fuel.

8. Radiation field.

Description

The basic methodology of the tests will be to subject specimens of spent fuel rods to contact with the Yucca Mountain Project reference ground water (well J-13 water). Tests will be conducted using a variety of spent fuel types that are typical of the population of fuel expected to be emplaced in the repository. Periodic samples of the solution will be analyzed for water chemistry and radionuclide content. At the end of the tests, samples of the leached fuel will be examined with the scanning electron microscope and optical microscope to determine, if possible, the location of any preferential dissolution. An effort will be made to identify any phases that precipitate during the test. The results of two series of tests using these methods are summarized in Section 7.4.3.1.1.

Additional tests will be conducted to determine the rate of reaction of the uranium oxide matrix during oxidative dissolution. These tests will comprise both "static" dissolution tests using an isotope dilution technique (Bruton and Shaw, 1987) and flow-through tests. Experiments will be performed on both unirradiated uranium oxides and spent fuel. The effects on the reaction rate of temperature, solution chemistry, and the oxidation state of the uranium in the solid reactant will be determined. Combined electrochemical/spectroscopic techniques (Russo et al., 1987) will be used in other experiments to determine the chemical species present in solution and on the uranium oxide surface during the oxidative dissolution process. These data will be used in constructing a mechanistic model for the dissolution of the spent fuel matrix.

8.3.5.10.2.1.2 Subactivity 1.5.2.1.2: Oxidation of spent fuel

Objectives

The objective of this subactivity is to determine the oxidation rate of irradiated $UO_2$ fuel as a function of the parameters in the following list. The results of these tests will be used to support the development of release models under Information Need 1.5.3 (Section 8.3.5.10.3). Some of the oxidized fuel produced by this activity will be used in spent fuel dissolution tests.

Parameters

Information is required for the following parameters:

1. Temperature.
2. Grain size of the fuel.
3. Particle size of the fuel (fracture density).
4. Atmospheric humidity.
5. Radiation field.
6. Burnup of the fuel.
7. Fission gas release of the fuel.

Description

Two types of tests are planned: (1) thermogravimetric analyses (TGA) and (2) long-term dry-bath oxidation tests. Both techniques maintain the spent fuel specimen at a constant temperature and humidity in a 20% $O_2$ + 80% $N_2$ atmosphere. The primary means of determining the degree of oxidation is by monitoring the weight gain of the sample over the course of the test. The two methods are complementary. The TGA tests provide continuous monitoring of weight changes of small samples for periods up to approximately three months. The dry-bath oxidation tests use larger samples and can be run for longer periods of time (two or more years) and can, therefore, provide information on oxidation rates at lower temperatures than the TGA system; however, the record of weight gain by the sample is not continuous. After oxidation, fuel specimens from both types of tests will be examined using x-ray diffraction, ceramography, SEM, TEM, and the ion-microprobe. Einziger (1985) presents a more complete technical description of the tests to be conducted under this subactivity.

8.3.5.10.2.1.3 Subactivity 1.5.2.1.3: Corrosion of Zircaloy

Objectives

The objectives of this subactivity are to determine the principal modes of Zircaloy cladding degradation and to determine the failure rate of cladding due to these modes. The results of these tests will be used to support the development of release models under Information Need 1.5.3 (Section 8.3.5.10.3). Those release models are needed to resolve both this issue and Issue 1.4 (Section 8.3.5.9).

Parameters

Information is required for the following parameters:

1. Presence or absence of liquid water.

2. Water chemistry, especially iodine and fluorine content.

3. Stress levels in the cladding (includes pressurization and pressure due to fission gas release).

4. Temperature.

5. Compatibility of Zircaloy with other metals in the waste package.

6. Hydrogen (hydride) content of the cladding.

7. Thickness of the external oxide layer on the cladding.

8. Radiation field.

9. Irradiation, storage, and handling history of the cladding.

10. Presence and composition of residues or deposits (crud) on the cladding.

Description

Smith (1985) has summarized the conditions in a tuff repository as they pertain to Zircaloy corrosion and has identified the corrosion processes expected to operate under these conditions. As discussed in Section 7.4.3.1.3, the likely modes of cladding failure are (1) stress corrosion cracking, (2) other forms of electrochemical corrosion, and (3) hydride reorientation.

Stress corrosion cracking from the fuel side of the cladding is not considered a likely mode of failure in the repository. Existing models (Tasooji et al., 1984) suggest that the fuel rod temperature and stress histories are below threshold limits for initiation of failure by stress corrosion cracking. Additional experiments and analyses are planned, however, to support these indications. Uniform corrosion is thought to be too slow to be an important mode of cladding degradation. Nevertheless, the rate at which the Zircaloy corrodes will be studied as part of a series of electrochemical corrosion tests. These tests will also examine the potential for pitting corrosion. Within the range of expected water and vapor chemistry in the candidate repository, fluoride, and to a lesser extent chloride, iodine, cesium, and cadmium (the last three from the waste form) ions are the agents most likely to have an adverse effect on cladding integrity. Stress rupture of the cladding may occur if a small defect exists and the fuel oxidizes or if undefected rods are subjected to high temperatures. The hydrogen content of the cladding, particularly if the hydrogen is present as reoriented hydride platelets, may alter the susceptibility of the cladding to this mode of failure. Tests are planned to study each of the preceding modes of failure and to quantify the rate at which they occur. The effect of each of the relevant parameters given above will be examined. To obtain results on a

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laboratory time scale, it is likely that testing will need to be carried out under conditions more extreme than those anticipated for the tuff repository. Extrapolation of the results to repository conditions will require mechanistic models for the various failure modes and will be carried out under the activities for Information Need 1.5.3 (Section 8.3.5.10.3).

The report by Smith (1985) discusses the planned test matrix. Three types of tests are currently planned:

1. Electrochemical corrosion tests will involve the exposure of Zircaloy cladding to ground water in the presence of air, tuff, and candidate container materials at a variety of temperatures and pressures. As discussed above, the effect of particular ions thought to be important will be evaluated by modifying the chemistry of the solutions used in these tests. These tests will examine the rate of generalized corrosion and the susceptibility of the cladding to pitting corrosion under repository-relevant conditions. Post-test examination of the specimens will focus on any changes in the structure of the passivating oxide film and/or the growth of such a film for cases in which the pre-existing film was purposely removed.

2. Stress corrosion cracking testing will be carried out using C-rings of Zircaloy, which will be stressed to near their yield point in the presence of ground water, and by the use of an apparatus allowing cladding segments to be overpressurized in the presence of liquid water. The effect of initiating agents such as fluorine and iodine will be examined by altering the chemistry of the water. Other test methods, including standard methods where applicable, may be used to supplement these tests.

3. The role of stress rupture will be evaluated by overpressurizing Zircaloy-clad fuel rod segments. Tests will be conducted using cladding with a range of hydrogen content, hydride density, and hydride orientation to determine the effect of hydride reorientation on the mechanical strength of the cladding.

In all these tests, the Zircaloy will be examined after testing by a variety of techniques, including, but not limited to, metallography, scanning electron microscopy, transmission electron microscopy, and a scanning Auger technique. Additional tests may be undertaken as a better understanding of the behavior of Zircaloy under tuff repository conditions develops.

### 8.3.5.10.2.1.4 Subactivity 1.5.2.1.4: Corrosion of and radionuclide release from other materials in the spent fuel waste form

Objectives

The objectives of this subactivity are to quantify the corrosion rate and consequent release of radionuclides from components of the spent fuel waste form not included in the studies on the uranium dioxide fuel itself and its Zircaloy cladding. The primary components to be studied are stainless

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steel, Inconel, and naval brass parts used as spacers, fittings, and other structural elements of reactor fuel assemblies. The results of these tests will be used to support the development of radionuclide release models under Information Need 1.5.3 (Section 8.3.5.10.3).

Parameters

Information is required for the following parameters:

1. Composition of material.
2. Water chemistry.
3. Temperature.
4. Radiation field.
5. Irradiation history of the material.
6. Atmospheric humidity.

Description

At present, the tests to evaluate the release of radionuclides from assembly materials are in the planning stage. Some form of semistatic leach testing under conditions similar to those anticipated in the tuff repository will probably be performed. The tests will need to identify both the corrosion rate of the various assembly materials and the rate of radionuclide release from the materials and their corrosion products.

8.3.5.10.2.1.5 Subactivity 1.5.2.1.5: Evaluation of the inventory and release of carbon-14 from Zircaloy cladding

Objectives

The objectives of this subactivity are to determine the source, inventory, and location of carbon-14 in Zircaloy cladding. In addition, the potential for release of carbon-14 in the form of carbon dioxide from cladding will be studied. The parameters in the following list are presently thought to be of importance in determining both the inventory and release characteristics of carbon-14 in Zircaloy cladding. The results of these tests will be used to support the development of radionuclide release models under Information Need 1.5.3 (Section 8.3.5.10.3).

Parameters

Information is required for the following parameters:

1. Reactor type: pressurized water reactor (PWR), boiling water reactor (BWR), or other.

2. Irradiation history.

3. Extent and nature of crud deposits on the cladding.

4. Thickness of oxide film on the cladding.

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5. Temperature.

6. Radiation field.

7. Nature of atmosphere surrounding the cladding (oxidizing or inert).

Description

Experiments are planned to determine the release characteristics of carbon-14 from Zircaloy cladding. These involve heating whole assemblies, individual rods, or rod segments in an oxidizing atmosphere and measuring the release of carbon-14 as a function of time and temperature. Other studies are aimed at establishing how much of the carbon-14 is located within the Zircaloy and how much is carried by the external coatings of crud and zirconium oxide. These studies involve controlled etching of cladding segments before heating to release the carbon as carbon dioxide. In both types of test, the cladding will be examined to document the nature and extent of any surface deposits as well as any microstructure within the body of the cladding. Additional tests may be conducted as more information on carbon-14 in Zircaloy is gathered.

The source of the carbon-14 has a large role in determining whether the radioactive carbon is within the Zircaloy or is in surface deposits. If the carbon-14 is produced primarily by (n,p) reactions on nitrogen-14 impurities within the cladding, then most of the carbon-14 would be expected to be located there. If, on the other hand, (n,alpha) reactions on oxygen-17 in the reactor cooling water are the dominant source, then the carbon-14 will probably be mainly located in surface deposits. The relative importance of these sources may depend on the type of reactor involved. The release characteristics of the carbon-14 will depend strongly on the relative importance of these two sources; the carbon-14 should be released from surface deposits much more quickly than if the carbon must diffuse through a significant thickness of cladding. Isotopic analyses of the stable carbon-12 and carbon-13 associated with the released carbon-14 may aid in identifying the source of the latter.

### 8.3.5.10.2.1.6 Subactivity 1.5.2.1.6: Other experiments on the spent fuel waste form

Objectives

As testing continues on the properties and behavior of the spent fuel waste form, it is possible that additional tests not covered by the other five subactivities in this activity will be required. Those tests will be conducted under this subactivity. Currently, only one area of investigation falls into this category: the behavior of stainless steel-clad fuel under tuff repository conditions.

Description

Test descriptions will be issued as the need arises.

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8.3.5.10.2.2 Activity 1.5.2.2: Characterization of the glass waste form

The purpose of the subactivities in this analysis is to provide the data required to calculate release rates from glass waste forms.

8.3.5.10.2.2.1 Subactivity 1.5.2.2.1: Leach testing of glass

Objectives

The objectives of this subactivity are to (1) use static leach testing to provide high-quality, high-precision data on the rates and amounts of radionuclide release from waste glass in contact with standing water, and (2) use unsaturated leach testing to provide data on the rates and amounts of radionuclide release from waste glass that is contacted by water, which then flows off the glass without remaining for long periods of time.

Parameters

The information needed from other information needs includes

1. Waste glass composition.

2. Leaching water composition.

3. Temperature.

4. Ratio of water-to-glass surface area.

5. Container material.

6. Pour canister material.

7. Other waste form characteristics from Information Need 1.5.1 (Section 8.3.5.10.1).

The output parameters for this activity are the rates of release of radionuclides from waste glasses in contact with water and in the presence of important materials such as the container material.

Description

Leaching of glass in contact with standing water may occur when water fills a breached container and pour canister. Leach testing under static conditions will provide constraints on the release rate under these conditions. In addition, leach testing under static conditions is the simplest form of leach testing, and the results may be generally applied to provide constraints upon other leaching scenarios. The simplicity of these experiments makes them the most reproducible form of leach testing. The testing done in this activity is intended to test the most important scenarios for release (e.g., temperature, water chemistry, and interaction with repository materials). Accordingly, long-term test matrices will be set up drawing upon the information obtained from the materials interactions tests (Subactivity 1.5.2.2.2, Section 8.3.5.10.2.2.2) that will examine a broad range of possible leaching conditions.

A possible scenario for release from glass may be that in which water enters a breached pour canister and reacts with the glass (but is not held in contact with the glass) and then flows away. Water dripping onto and off glass is one example. Because of the extremely high glass-to-water ratios that may occur under these conditions, a special test called the Project Unsaturated Test Method has been developed to examine the effects of release under these conditions (Section 7.4.3.2). As part of this activity, unsaturated testing will be performed to provide the complementary data to that described for static leach testing.

Glasses to be tested in this subactivity will include both radioactive and simulated-waste glasses. A range in compositions representing the range to be produced (as described by the producer in the Waste Compliance Plan and Waste Qualification Report) will be used. All related confirmation testing will be conducted under this activity (refer to milestones in Section 8.3.5.10.2.4).

### 8.3.5.10.2.2.2 Subactivity 1.5.2.2.2: Materials interactions affecting glass leaching

Objectives

The objective of this subactivity is to examine a broad range of factors that may influence glass leaching and degradation. Those determined to be most important will be tested further in Subactivity 1.5.2.2.1 (Section 8.3.5.10.2.2.1). This activity will provide information on mechanisms for input to development of the glass leaching model, Activity 1.5.3.4 (Section 8.3.5.10.3.4). Both calculational and experimental techniques will be used to examine the effects of possible interactions so that no important mechanisms for glass release will fail to be considered by the testing and modeling programs.

Parameters

The information needed from other information needs includes

1. Waste glass composition.

2. Leaching water composition.

3. Temperature.

4. Ratio of water-to-glass surface area.

5. Container material.

6. Pour canister material and heat-treated canister material.

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7. Radiation effects on leachant composition.

8. Cracking and disaggregation of glass.

9. Changes in fluid composition caused by other repository components such as grout and concrete.

10. Other waste form characteristics from Information Need 1.5.1 (Section 8.3.5.10.1).

The output parameters for this activity are the effects on glass alteration rate, and on glass leaching rate and mechanism caused by the interactions of the studied materials.

Description

A large number of interactions may affect the rate of glass degradation in the repository. Among the most important are interactions involving the parameters in the preceding list. Other interactions will be identified and examined as part of this activity.

Two types of experimental work will be conducted. In the first type, leaching experiments will be performed in which the interacting material, or radiation, is present with the glass. Several leaching tests will be used including static testing, unsaturated testing, and pulsed flow testing. In the second type of testing, fluid chemistries will be altered to simulate repository influences. In both tests, EQ3/6 modeling will be used to aid in designing the experiments, and the results will then be used to aid in the development of the glass modeling EQ3/6 package of codes. Container materials will be added to the experiments based on the metal selection process (Issue 1.4, Section 8.3.5.9). Until the metal is selected, type 304L stainless steel (the pour canister material) will be used.

### 8.3.5.10.2.2.3 Subactivity 1.5.2.2.3: Cooperative testing with waste producers

Objectives

The objective of this subactivity is to conduct a cooperative testing program with the waste producers to allow for the testing of full-scale waste forms and to ensure that the laboratory-scale test results obtained by the producers are consistent with those obtained by the Yucca Mountain Project.

Parameters

The most important parameters for this subactivity are the following:

1. The effect of scale (full-scale versus laboratory-scale) tests on leaching rates.

2. Water flow and contact with glass in a pour canister.

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3. Reproducibility and accuracy of testing.

4. Glass compositional effects on leach rates.

Other parameters are the same as those listed for Subactivity 1.5.2.2.2 (Section 8.3.5.10.2.2.2).

Description

This subactivity involves cooperation with the waste form producers in designing and interpreting leach tests on laboratory-scale and full-scale waste forms. No testing under this subactivity will be performed by the Yucca Mountain Project. Testing will be performed by the waste producers on pieces cut from full-scale canisters to ensure that the laboratory-scale measurements can be adequately applied to actual leaching in the repository. The Yucca Mountain Project will provide the following: (1) assistance in designing the experiments, (2) assistance in interpreting the results including geochemical analysis using EQ3/6 and the glass modeling codes, and (3) repository relevant materials to be used in testing possible repository interactions.

The waste producers are also conducting laboratory-scale tests similar to those done by the Yucca Mountain Project. In this subactivity, those results will be compared to ensure that the waste producers and the Yucca Mountain Project both observe similar behavior in glass leaching experiments. Additional tests may be added to the other two Yucca Mountain Project glass testing activities to confirm these results or to resolve inconsistencies.

Because a large body of consistent data on glass leaching behavior is required, the cooperation with waste form producers is important to confirm that the data provided by waste-producer tests will be usable in licensing the repository.

## 8.3.5.10.3 Information Need 1.5.3: Scenarios and models needed to predict the rate of radionuclide release from the waste package and engineered barrier system

Technical basis for addressing the information need

This information need will draw together the scenarios and conditions for radionuclide release provided by information needs or investigations under the the following issues or characterization programs:

| Issue or program | Short title |
|---|---|
| 1.1 | Total system performance (Section 8.3.5.13) |
| 1.4 | Containment by waste package (Section 8.3.5.9) |

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| Issue or program | Short title |
|---|---|
| 1.10 | Waste package characteristics (postclosure) (Section 8.3.4.2) |
| 8.3.1.2 | Geohydrology |
| 8.3.1.3 | Geochemistry |
| 8.3.1.4 | Rock characteristics |
| 8.3.1.5 | Climate |
| 8.3.1.6 | Erosion |
| 8.3.1.8 | Postclosure tectonics |

This information will be combined with the models that will be used to predict radionuclide release under anticipated and unanticipated conditions for 10,000 yr (10 CFR 60.112 and 40 CFR 191.13) and under expected conditions for 100,000 yr (10 CFR 960.3-1-5).

Link to the technical data chapters and applicable support documents

The scenarios and conditions for radionuclide release are derived from the information on site geology (Chapter 1), hydrology (Chapter 3), geochemistry (Chapter 4), climatology (Chapter 5), repository design (Chapter 6), emplacement environment (Section 7.1), waste package design (Section 7.3), waste package environment (Section 7.4.1), and metal barriers studies (Section 7.4.2). Some scenarios requiring analysis will arise from information needs of total system performance assessment, which are discussed in Section 8.3.5.13.

Performance assessment models that will be used to predict radionuclide release from the engineered barrier system have been discussed in Section 7.4.5 and the interrelationships are shown in Figure 8.3.5.10-1. The design-related inputs to these analyses appear in Section 7.3. Details of activities that will develop waste package process models that will be implemented in performance assessment modeling appear in waste package environment (Section 7.4.1), metal barrier studies (Section 7.4.2), waste form degradation (Section 7.4.3), and geochemical modeling (Section 7.4.4). Model inputs are shown in Table 8.3.5.10-1.

Parameters

Input parameters for scenario development are the following:

1. Output parameters from Issue 1.1 (conditions that reflect climatic, geohydrologic, or geologic phenomena in the far field but which result in changes at the repository level computed by total system performance models) (Section 8.3.5.13).

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2. Output parameters from Information Need 1.4.4 (waste package container failure modes and times) (Section 8.3.5.9.4).

3. Output parameters from Issue 1.10 (configurations and characteristics of waste package designs) (Section 8.3.4.2).

4. Output parameters from Issue 1.11 (characteristics of the repository and engineered barriers) (Section 8.3.2.2).

5. Output parameters from Issue 1.12 (characteristics of the shaft and borehole seals that may affect waste package performance) (Section 8.3.3.2).

6. Output parameters from Characterization Programs 8.3.1.2 and 8.3.1.3 (changes in geohydrologic and geochemical conditions).

7. Output parameters from Characterization Programs 8.3.1.4, 8.3.1.5, 8.3.1.6, and 8.3.1.8 (changes in geologic, hydrologic, and geomechanical conditions that may directly affect waste package performance).

Output parameters for scenario development are

1. Identification of scenarios due to all anticipated processes and events, in terms of qualitative features and far-field or other controlling parameters.

2. Parameters of the near-field environment and of the waste package, describing scenarios due to anticipated processes and events.

3. Parameters of the near-field environment and of the waste package, describing scenarios due to unanticipated processes and events to the extent needed by Issue 1.1 (total system performance).

4. A determination of whether the parameters of the scenarios due to all anticipated processes and events fall within the design envelope assumed for waste package design and performance allocation (see Section 8.3.4.2).

Data needed for the geochemical modeling of the reaction of waste forms with water are the following:

1. The equilibrium aqueous speciation of solutes.

2. The equilibrium solid and aqueous compositions of systems consisting of mixtures of gas, liquid, and solids.

3. The thermodynamic and kinetic data for solid and liquid species required to calculate equilibria and reaction rates in gas-liquid-solid systems.

The output parameters from geochemical modeling are the following: fluid compositions, amount and composition of solids, and overall rates of reaction and approach to equilibrium for complicated systems.

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The input parameters for waste form release models are the following:

1. The waste form characterization parameters specified in Information Need 1.5.1.

2. The waste form material properties specified in Information Need 1.5.2.

3. The parameters specified previously in this information need (1.5.3).

The output parameters from waste package release models are the rates of release of radionuclides from waste packages.

The performance assessment models will require two kinds of input parameters, both of which have been described earlier in this section. First, the parameters describing anticipated and unanticipated events (i.e., the scenarios) serve to establish the range of cases for which performance must be calculated. These parameters also specify the range of conditions under which the waste package must perform. Second, the remaining parameters for release and geochemical interactions provide the mechanisms of waste release that will be integrated by the performance assessment calculation of release from the engineered barrier system. The parameters required by performance assessment will contain the probabilistic information necessary to meet the reasonable assurance standard required by the NRC.

The output parameters obtained from the waste package performance assessment model are cumulative distribution functions for the time to failure of the container, the release rates of radionuclides from failed waste packages, and release rates of radionuclides from the engineered barrier system.

Logic

Calculation of the release rates of radionuclides from the engineered barrier system requires an integrated analysis of all the significant factors affecting loss of waste package containment. Significant factors include scenarios encompassing anticipated processes and events, near-field environments due to interactions between waste packages and the scenario-driven conditions, and geochemical system states and reactions. Processes in the loss of waste package containment include the release of radionuclides from the waste form and the movement of radionuclides away from a breached package. Release begins with container failure. Gaseous radionuclides may be assumed to leave the package upon loss of containment. Solid phase radionuclides that are contained within the waste form will require contact with ground water for release to occur. Therefore, the amount and chemistry of ground water as influenced by the waste package environment, the condition of the container, the nature of the interaction between waste form and ground water, and the inventory of waste present will affect the availability of radionuclides for transport. Once the radionuclides are in solution, pathways by which the waste may leave the waste package will complete the determination of releases from the engineered barrier system.

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The information provided under this information need constitutes the basic models needed to assess waste package performance. They incorporate all applicable information from characterization, design, and performance studies.

There are five activities in this information need. Each of the first four activities addresses a specific modeling need, and they are all combined in the fifth activity (1.5.3.5) (Section 8.3.5.10.3.5).

### 8.3.5.10.3.1 Activity 1.5.3.1: Integrate scenarios for release from waste package

This activity consists of four subactivities.

#### 8.3.5.10.3.1.1 Subactivity 1.5.3.1.1: Develop scenario identifications

Objectives

The objective of this subactivity is to identify scenarios for all anticipated and unanticipated processes and events.

Parameters

The input parameters for this subactivity are listed in the combined list in the technical basis section for this information need. The input parameters for this subactivity are identified anticipated processes and events from Issue 1.1 and other processes and events that will be screened to determine whether they should be considered anticipated.

Description

This subactivity will identify scenarios in terms of far-field or controlling parameters. This will be done for a super-set of processes and events, which will include all anticipated processes and events, and additional processes and events that are to be screened as to whether they are to be considered anticipated. This subactivity will also accept scenario descriptions from Issue 1.1 (total system performance) that are to be processed for application in Issue 1.1.

The process for this subactivity will be to accept scenario descriptions for credible processes and events from Issue 1.1, to consider scenarios that may have been screened out from Issue 1.1 on the basis of consequences to Issue 1.1 rather than on the basis of probability, and to search systematically for scenarios that may be due to near-field processes in addition to those developed by Issue 1.1.

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8.3.5.10.3.1.2 Subactivity 1.5.3.1.2: Separate scenarios into anticipated and unanticipated categories

Objectives

The purpose of this subactivity is to determine the types and extent of scenarios covering all anticipated processes and events.

Parameters

The input parameters for this subactivity are listed in the combined list in the technical basis section for this information need. These include scenarios identified in Subactivity 1.5.3.1.1.

The output parameters are

1. A binary value of anticipated or unanticipated for all scenarios identified in Subactivity 1.5.3.1.1.

2. The maximum amplitude that falls within the anticipated range, for those scenario categories spanning the anticipated and unanticipated while covering a range of amplitudes (e.g., amount and timing of climate change).

Description

Issue 1.1. has a category of scenarios, expected case, that incorporates anticipated processes and events. The separation of this category is just a convenience for Issue 1.1, since probability values are attached to all scenarios and are used in constructing a complementary cumulative distribution function in Issue 1.1. This issue (1.5) will independently determine what scenarios and scenario values are to be included in the group of anticipated processes and events.

The first step is to develop a decision criterion on how to separate processes and events into the anticipated or unanticipated categories. The NRC regulation 10 CFR Part 60 defines anticipated processes and events as those reasonably likely to occur during the period to which the regulations apply. This definition is qualitative; a clear-cut decision criterion must still be developed. Practice from other fields of engineering design will be considered in developing a decision criterion.

The second step is to develop data (or bounding estimates on the data) of parameters needed for the decision criterion. Depending on the criterion developed, these may be probabilities of events, curves of amplitude versus recurrence time, other data from the geologic record, or other data from geologic or physical first principles.

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8.3.5.10.3.1.3 Subactivity 1.5.3.1.3: Development of parameters describing the scenarios

Objectives

The objective of this subactivity is to develop and assemble the parameters of the near-field environment and of the waste package describing scenarios covering anticipated processes and events.

Parameters

The input parameters for this subactivity are listed in the combined list in the technical basis section for this information need. The most important parameters include scenarios for anticipated processes and events, waste package environment, waste package configuration, and containment performance.

Description

This subactivity will develop and assemble the parameters of the near-field environment and of the waste package, describing the scenarios identified in Subactivity 1.5.3.1.2 as anticipated and also those needed by Issue 1.1 (total system performance). This subactivity will develop how parameters of the far field influence the near field, taking into account interactions of the waste package with its environment; what parameters are determined by repository and waste package design; and how parameters of the near field and the waste package evolve under processes at the waste package scale.

The near-field parameters of the scenarios will be developed in conjunction with Information Need 1.10.4 (Section 8.3.4.2.4). Issue 1.1 will identify scenarios and determine their average impacts at the repository horizon, in most instances without waste-package-environment interactions or waste-package-scale variations in properties. This subactivity will transfer the information to Information Need 1.10.4, where the interactions between the waste packages and their environment for the given scenarios will be evaluated. These evaluations will then be combined in this subactivity with waste package and other parameters to complete the scenario description in terms of its parameters.

Some parameters of the scenario will evolve with time depending on waste package and near-field processes. This subactivity, together with model applications of the waste package system model, will track the evolution of these parameters. These parameters will determine the range of conditions for which near-field, waste package, and waste form detailed calculations will have to be established in support of waste package system assessments. This subactivity will assemble and transmit to the detailed tasks the conditions under which processes will have to be evaluated. As an example, time of container failure and amount of corrosion products still present will be transmitted to the waste form alteration and release activities.

For unanticipated scenarios needed in Issue 1.1, the degree of specificity in the near-field characterization of the scenario may be less than for the scenarios in Issues 1.4 and 1.5, depending on the extent of performance allocated to the waste package in these scenarios by Issue 1.1.

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The parameter (of near-field and waste package) development for scenarios will be done both for anticipated and unanticipated processes and events. The parameter descriptions of scenarios due to anticipated processes and events will be used in evaluations for this issue (1.5) and Issue 1.4 (containment by waste package, Section 8.3.5.9). The parameter descriptions of scenarios due to both anticipated and unanticipated processes and events will be used in evaluations of radionuclide source term for use in Issue 1.1.

The parameter values will include point estimates and probabilistic characterizations. The point estimates will be either best estimates, high percentile probability estimates, or bounding values as appropriate for the application. The probabilistic characterizations will lead to probabilistic descriptions of results. These will be directly transmitted to Issue 1.1; that issue is concerned with performance in terms of an Environmental Protection Agency performance criterion phrased explicitly in probabilistic terms. The probabilistic characterizations will also be used in Issues 1.4 and 1.5 in providing evidence to support a determination that the performance issues will be satisfied with a safe margin that is, that there is reasonable assurance that the performance objectives will be met.

### 8.3.5.10.3.1.4 Subactivity 1.5.3.1.4: Determine adequacy of design envelope of waste package

Objectives

The objective of this subactivity is to determine the adequacy of the design envelope of waste package for design and testing activities.

Parameters

The input parameters for this subactivity are listed in the combined list in the technical basis section for this information need. These include scenarios from anticipated processes and events and conditions and processes in the near-field of the waste package.

Description

The design envelope for waste package design (Section 8.3.4.2) and for performance allocation (Sections 8.3.5.9 and 8.3.5.10) was selected to be an envelope of conditions for all anticipated processes and events. In selecting the envelope, due consideration was given to present uncertainties. When the anticipated processes and events and the resulting scenarios are determined, confirmation or adjustment of the design envelope will be required.

This subactivity will examine the near-field conditions determined under Subactivity 1.5.3.1.3 due to anticipated processes and events and their interaction with waste package influences on the near-field environment, and determine whether the near-field conditions fall within the design envelope assumed in Section 8.3.4.2 and the performance allocations assigned in Sections 8.3.5.9 and 8.3.5.10. If they do, this will confirm the adequacy of the design envelope for design and testing activities. If not, this subactivity will determine the range of conditions that must be considered in design and testing. The range may be amenable to more detailed specification in terms of more parameters and correlations among parameters.

### 8.3.5.10.3.2 Activity 1.5.3.2: Develop geochemical speciation and reaction model

Objectives

The objective of this activity is to further develop the geochemical modeling code EQ3/6 for use in modeling of waste form radionuclide release and the behavior of released radionuclides. The need to make long-term predictions of release rates and the fate of released radionuclides requires the use of a geochemically sound model that accounts for the perturbations that may exist within the repository and is consistent with existing thermodynamic and waste form experimental data. For use in understanding long-term behavior, geochemical modeling codes must be capable of modeling processes already identified as major factors affecting radionuclide behavior, such as dissolution and precipitation. For use in modeling waste form release, the codes must be capable of modeling the dissolution behavior of the waste in ways that are consistent with experimental data and that provide information about the important factors affecting radionuclide release.

The EQ3/6 code package, the associated data base, and the use of the code in geochemical applications have been described in Section 7.4.4. The codes have already been used to interpret the results of rock-water interactions tests, to evaluate ground-water analyses and determine whether equilibrium conditions exist, to determine solubility limits for radionuclides under various realistic conditions, and to aid in the design of laboratory experiments by identifying parameters that need to be measured to understand the chemical processes that drive the experimental system. The EQ3/6 package may be used to calculate the fluid compositions and solid phases with their amounts and their radionuclides content that would result from the equilibration of hypothetical solutions resulting from the dissolution of waste forms in water. Similarly, it may be used to calculate the changes in composition of a water as it flows through and reacts with repository rock, engineered barrier materials, or a waste form. The codes are also useful for testing the thermodynamic feasibility of proposed mechanisms and for identifying the equilibrium reactions that control a given process. Two subactivities support this evaluation.

#### 8.3.5.10.3.2.1 Subactivity 1.5.3.2.1: Develop data base for geochemical modeling

Objectives

The objective of this subactivity is to develop a supporting data base containing thermodynamic and kinetic information on aqueous species and solids that may occur in the repository.

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The application of EQ3/6 to modeling of fluids important to radionuclide release behavior requires this data base, which is accumulated through review and verification of published values and through determination and validation of new data determined to be of the highest importance in continuing the modeling goals outlined in the other investigations in this information need and in Information Need 1.5.5, (Section 8.3.5.10.5). An important aspect of this activity is a sensitivity analysis to determine which data are the most critical to achieving modeling needs described under Subactivity 1.5.3.2.2 (Section 8.3.5.10.3.2.2).

Description

EQ3/6 data files contain the standard thermodynamic data that are reported in the literature for solids, aqueous species, and gases. These values have been gathered from the available literature as an ongoing effort. Despite a doubling in the total species in the data base and many improvements in consistency, organization, and documentation, data base work has lagged behind code development. Therefore, a significant increase in effort will be directed toward improving and upgrading the data base.

Requisite thermodynamic values for aqueous species and solid phases specific to nuclear waste that are reported in the literature will be critically evaluated for instances where data are missing or inadequate for modeling needs. In these instances, laboratory work will be conducted to obtain that data. Full compatibility with the key values recommended by the Committee on Data for Science and Technology (CODATA) task group and the thermodynamic data base sponsored by the Nuclear Energy Agency (NEA) will be developed. The validation of the data base will be carried out by comparing the results of theoretical calculations using the EQ3/6 package with experimental results and field data.

The implementation of the aspects of the EQ3/6 package required to adequately model radionuclide behavior will require other types of information in addition to that currently contained in the thermodynamic data base. This information includes kinetic rate constants, nucleation rates, and sorption coefficients. When required by the modeling efforts supported by EQ3/6, these data will also be evaluated and experimentally determined.

The data base experimental and analytical activities have been divided into five principal areas of study.

1. Thermodynamic data analysis: Sensitivity, uncertainty, and estimation/correlation studies for the radionuclides of regulatory concern, and the solution and solid species affecting them (25 to 30 elements). Review existing data, make estimates and correlations to existing data for missing data, and constrain required new data in terms of uncertainties associated with the data (or lack of data). In addition to guiding experimental studies, this activity will produce an uncertainty map for the data base.

2. Actinides and technetium: Thermodynamic data for the aqueous and solid species expected to occur at Yucca Mountain. Elements are those that require high confidence (Table 8.3.5.10-3b). Actinide

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species to be considered are $U^{4+}$, $UO^{2+}$, $Np^{4+}$, $NpO^{+}$, $Pu^{4+}$, $PuO^{+}$, $PuO^{2+}$, $Am^{3+}$, and $TcO^{-}$.

3. Nonradioactive species required to support calculations for radionuclides: Make minor additions to the thermodynamic data for the Yucca Mountain ground-water aqueous species, and for a few solid species for which inadequate data exist. Elements and species to be considered (Tables 8.3.4.2-4 and 8.3.5.10-3a) are $HCO^{-}$, $OH^{-}$, $SO^{2-}$, $NO^{-}$, $PO^{3-}$, $Cl^{-}$, $F^{-}$, $SiO_2(aq)$, $Na^{+}$, $Ca^{++}$, $K^{+}$, and $Mg^{++}$.

4. Data base validation: Controlled laboratory experiments to confirm results of calculations using the data base. Several sets of measurements for the elements listed here, at several temperatures and pH values, will required.

5. Other waste radionuclides: Thermodynamic data for the fission and activation products in spent fuel, which require high confidence (Table 8.3.5.10-3b). Nuclides to be considered are Zr-93, C-14, Sn-126, Se-79, Cs-135, Pd-107, Th-230, Ra-226, Pb-210, Ni-59, and Nb-94.

(The maintenance of the computerized data base is handled under Subactivity 1.5.3.2.2. This keeps experimental and analytical work in this subactivity and computer-dependent activities in Subactivity 1.5.3.2.2.)

## 8.3.5.10.3.2.2 Subactivity 1.5.3.2.2: Develop geochemical modeling code

### Objectives

The objective of this subactivity is to upgrade the EQ3/6 package to model important chemical processes in a nuclear waste repository. These codes will then be used in the other waste package modeling efforts (this information need) to aid in design and interpretation of experimental work on waste form degradation (Information Need 1.5.2, Section 8.3.5.10.2), to model the behavior of radionuclides after release from the waste package (Information Need 1.1.4, Section 8.3.5.13.4 and Information Need 1.5.5, Section 8.3.5.10.5), and to aid in the description of the waste package environment (Information Need 1.4.2, Section 8.3.5.9.2 and Information Need 1.10.4, Section 8.3.4.2.4). Code documentation and verification will be done concurrently with development.

### Description

The following capabilities will be added to the EQ3/6 code package to achieve the objectives:

1. Upgrade data management capabilities. In conjunction with Subactivity 1.5.3.2.1, the current EQ3/6 data base will be combined with new data and stored in a relation data base package that is capable of audit tracking; controlled access; output of data files in various formats, including the EQ3/6 format; automatic conversion of units and standard states so that only values directly from the

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reports are added without hand calculations; retrieval of data by type, such as all data from one report; complete reference control on all data; flagging for data review status and quality assurance level; and reporting and pass-through of error bars and limits to use of estimated data. The entire data base will then be processed according to the data review and analysis methods used by the Committee on Data for Science and Technology (CODATA), the Nuclear Energy Agency (NEA), and the National Bureau of Standards to ensure that appropriate data are being used and that they are completely compatible with data from the international data review groups (CODATA and NEA).

2. Extend the geochemical flow model. The current flow model may be used to examine the evolution of a packet of ground water as it moves along a flow path. A different flow model is required to model the interaction of a stationary object, such as the waste form, with successive packets of water. This models the evolution of the waste form (formation of precipitates and loss of soluble elements), and provides solution compositions leaving the waste package.

3. Extend the kinetics capabilities. The current capabilities will be tested and modified to include the effects of nucleation and substances inhibiting precipitation. Kinetic data will be accumulated from published work, and the possibility of extracting kinetic information from dissolution data will be examined and implemented if appropriate. New kinetics data on important systems will be collected as part of Subactivity 1.5.3.2.1.

4. Complete model for systems open to gases. The present fixed-fugacity option will be upgraded to better model the expected scenarios in Yucca Mountain. Currently, equilibrium with an unlimited gas reservoir may be modeled. The option will be expanded to permit modeling of closed systems containing varying amounts of gas.

5. Extend solid solutions to include site mixing concept. This addition would make the modeling of intermediate-composition phases more accurate and provide a better way to handle the substitution of radionuclides into specific sites in minerals. The current method calculates properties of intermediates using both ideal and nonideal (as appropriate) mixing of end members. The new capability would deal explicitly with intrasite mixing (where an ion can occur on more than one site in a mineral) and will be selectively applied to cases where substitutions result in structural changes in a mineral that are not present in any end-member phases.

6. Add equilibrium sorption model. A model for sorption onto waste package and repository materials is required to adequately model radionuclide concentrations in water and the migration of radionuclides. A suitable model will be identified and incorporated into the EQ3/6 code package. Sorption data available in the literature will be adopted as appropriate. New sorption data on important systems will be collected as part of Information Need 1.5.5 and Characterization Program 8.3.1.3.

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7. Extend the code for compatibility with other models. Because EQ3/6 is used in support of other modeling efforts, revisions or additions for compatibility and flexibility may be required.

8.3.5.10.3.3 Activity 1.5.3.3: Generate models for release from spent fuel

8.3.5.10.3.3.1 Subactivity 1.5.3.3.1: Generate release for spent fuel models

Objectives

The primary mechanism for the release of radionuclides from spent fuel is via water that has come in contact with the waste form through a breach in a container. A few radionuclides such as carbon-14 (or krypton-85 in the containment period) can be released in the gas phase in the absence of liquid water. The objective of this activity is to develop models for the release of radionuclides from the spent fuel waste form as a function of time using the scenarios identified in Activity 1.5.3.1 of this issue. This development will require the development of submodels for oxidation and radionuclide release from spent $UO_2$ fuel, the corrosion and failure rates of Zircaloy cladding, and the release rate of radionuclides from nonfuel assembly parts. These submodels will be based on the experimental data generated under Information Need 1.5.2. The models developed in this activity will be used in resolving both Issues 1.4 and 1.5.

Parameters

The parameters required for this activity are given in the combined list for this information need. The most important input parameters are expected to be the water contact rate and mechanism, water chemistry, temperature, and time. The output parameter provided by this activity will be a model for radionuclide release from the spent fuel waste form within breached containers.

Description

Tests to determine the behavior of the spent fuel waste form under the anticipated conditions at Yucca Mountain are described in Sections 7.4.3.1 and 8.3.5.10.2 (Information Need 1.5.2). These tests form the basis for the modeling to be carried out under this activity. To extrapolate the observations made in the laboratory to the time scale for which the performance of the repository must be specified to satisfy the 10 CFR Part 60 requirements, it will be necessary to develop predictive models based on an understanding of the mechanisms involved in the degradation of the waste form. Several submodels will be generated describing the behavior of each component of the waste form that affects the release of radionuclides.

The largest reservoir of radionuclides in the waste form is the spent $UO_2$ fuel, and the primary barrier to the release of radionuclides is the fuel material itself. Thus, the most important submodel to be generated under this activity is the one for the dissolution of and radionuclide release from the spent fuel. This model will yield predictions of the concentration of

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radionuclides as a function of time in the ground water that has come in contact with spent fuel. Analyses of the available data (Section 7.4.3.1.1) indicate that the radionuclides of interest occur in two regions within the spent fuel, and are released at different rates, depending on the region. One group of radionuclides is present within the $UO_2$ fuel matrix, and therefore has a release rate that is limited by the dissolution of that matrix. The other group of radionuclides is located both in the matrix and along fuel grain boundaries or in the gap between the fuel pellets and the cladding. The grain boundary and gap inventory of these nuclides is typically 1-2 percent of their total inventory. This function is released rapidly upon contact of the fuel by water. The latter group also includes those that are present in a gaseous phase and can, therefore, be released from a breached container even without the presence of water as a solvent and transport medium. The submodels will account for all of these release mechanisms.

Development of the submodel will be assisted by the use of EQ3/6 code analyses. These analyses will provide insight into the role of solid phases in determining the equilibrium solution concentrations of elements present in sufficient quantity to saturate the ground water. The final submodel will incorporate EQ3/6 calculations of the dissolution rate and solution concentrations of radionuclides of interest in performance assessment. The usefulness of the EQ3/6 code to this effort is critically dependent upon the availability of good thermodynamic data for the chemical elements and saturating phases of interest.

Since the transport of most radionuclides from spent fuel requires that water come in contact with the fuel, the presence of undefected cladding would provide a second barrier (after the container) between the fuel and the environment. The failure rate of the cladding will have a large effect on the release rate of the gap and grain boundary inventory of the fuel as discussed previously. A second submodel will be developed that will estimate the failure rate of Zircaloy cladding as a function of time. This submodel will incorporate the results of experimental tests aimed at identifying the important modes of corrosion resulting in failure of the cladding and the rates of those modes. Mechanistic models of Zircaloy failure will be developed to extend the laboratory measurements to the time scale of the repository isolation period.

The third submodel will quantify the oxidation rate of spent $UO_2$ fuel exposed to an atmosphere containing oxygen. Because the ground-water infiltration rate at Yucca Mountain is low, if both the container and cladding fail, fuel may be exposed to the air for some time before it is contacted by water. The higher oxides of uranium may have different leaching behavior than $UO_2$. In addition, oxidation of fuel in slightly defected cladding could lead to gross failure of the cladding due to expansion of the fuel during oxidation. The oxidation submodel will be based on the results of oxidation experiments on spent fuel discussed in Information Need 1.5.2. The oxidation rate of $UO_2$ strongly depends on temperature; the model, therefore, will be time dependent. At some time after emplacement, the temperature of the fuel is expected to be sufficiently low that no significant oxidation of the fuel will occur in the time available. After that time, this submodel will not play an active role in determining the release of radionuclides from the waste form.

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The fourth and final submodel will describe the release of radionuclides from cladding and other fuel assembly hardware (mostly Zircaloy, Inconel and stainless steel).

The submodels will be combined to make a single model for the release of radionuclides from the spent fuel waste form. Obviously, there will be significant interactions between the submodels. For example, the release of most radionuclides cannot occur until the fuel is exposed by cladding failure and water contacts the fuel. Thus, the predictions of the Zircaloy submodel must be used as input for the submodel describing the release of radionuclides from the fuel.

### 8.3.5.10.3.4 Activity 1.5.3.4: Generate models for release from glass waste forms

#### 8.3.5.10.3.4.1 Subactivity 1.5.3.4.1: Generate release models for glass waste forms

Objectives

The release of radionuclides from glass waste forms may occur if water contacts a container that has breached. The objective of this activity is to design models for glass release based on the scenarios identified in Activity 1.5.3.1 (Section 8.3.5.10.3.1). The geochemical modeling codes described in Activity 1.5.3.2 (Section 8.3.5.10.3.2) will be an important part of these models. The models generated by this activity will provide estimates of radionuclide release as a function of repository conditions and will be used in resolving Issue 1.4 and this issue (1.5).

Parameters

The parameters required for this activity are given in the combined list for this information need. The most important input parameters are expected to be water contact rate and mechanism, water chemistry, temperature, time, and interactions with repository materials. The parameter provided by this activity will be a model for radionuclide release from glass waste within breached containers.

Description

The behavior of glass waste forms under the expected conditions at Yucca Mountain is described in Section 7.4.3.2. The extension to long times of the semiempirical relationships discovered by laboratory testing cannot be made without understanding the mechanisms involved and assessing the effects of factors such as the slow buildup of crystalline layers. The model to be developed will account for glass degradation and radionuclide release using geochemically sound methods that incorporate expected perturbations in the repository environment, and will be consistent with the existing laboratory and natural analog studies.

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Glass performance modeling will depend upon two basic concepts. First, the rate of release from the thermodynamically unstable waste glass is a kinetically controlled process. No formal equilibrium can exist. Second, once components are released from glass, the formation of solids and composition of fluids may be modeled by equilibrium processes. The final outcome of these equilibrium processes will be modeled, providing important limits on the behavior of radionuclides. In addition, the kinetics of these processes may be modeled to provide more accurate estimates of radionuclide concentrations in waste package fluids as a function of time throughout the life of the repository.

The model for glass degradation will incorporate the following items, presented here in the order in which they will be developed:

1. Calculation of the composition of the solutions that are in true equilibrium with the solid phases that precipitate on the surface of nuclear waste glasses.

2. Calculation of the rate of degradation of glass using kinetic rate laws based on transition state theory, deriving rate constants from experimental and natural-analog studies.

3. Calculation of the rate of formation of solid precipitates and the concomitant rate at which radionuclides are permanently sequestered in those stable phases.

4. Calculation of the effects of repository materials on the previously stated items, including heat-affected stainless steel from the pour canister.

5. Calculation of the composition of fluids leaving a glass waste package by combining the preceding items.

In each item, the appropriate analytical expressions will be identified from experimental work, from review of the glass degradation literature, and from geochemical modeling concepts incorporated in the EQ3/6 code. Calculations will be performed using EQ3/6.

Validation of the glass model will be done in two stages. First, the model will be developed in concert with experimental work and will be tested for its ability to describe accurately the experimental work. An important aspect of this is the use of modeling to aid in understanding the physical processes important in glass degradation. Second, the results of long-term modeling will be compared with extrapolations of laboratory data, and with natural analogs. This second effort will both test the validity of the model and, more importantly, examine whether the experimental work has examined all the important geochemical interactions that are predicted to occur over long periods of time.

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8.3.5.10.3.5 Activity 1.5.3.5: Waste package performance assessment model development

Three subactivities support this performance evaluation.

8.3.5.10.3.5.1 Subactivity 1.5.3.5.1: Development of system model

Objectives

The development of the system model for waste package performance assessment is the subactivity that integrates into a single deterministic model the submodels of processes that affect waste package releases. Models for waste form degradation and radionuclide release will be combined with mechanistic representations of waste package environment, waste package design features and mechanical models, and container degradation models. The resulting waste package system model will calculate (1) the performance-related parameters used in evaluating compliance with Issue 1.4, substantially complete containment (Section 8.3.5.9), and (2) the release rates of radionuclides from failed waste packages as a function of scenario inputs, for use in evaluating compliance with this issue and Issue 1.1.

The design objectives for satisfying Issue 1.4 recognize that among the tens of thousands of waste packages there will be differences in individual performance. The design objectives are set in terms of percent of waste packages, or releases or release rates summed over the set of waste packages. To reflect these differences in individual waste package characteristics in the modeling, the deterministic system model can be executed a number of times with different inputs. Alternatively, the probabilistic system model may be used with inputs appropriate to the problem and to the use in issue resolution, as discussed in the following paragraphs.

The design objectives for satisfying Issue 1.4 admit either a partially probabilistic interpretation (with probability distributions of key input waste package parameters supported by measurements to be done) or a deterministic interpretation (with established distributions of key input parameters; for example, the heat loading per package could be established based on the projected waste form characteristics). In either interpretation, the approach to resolution of the issue will use only those distributions that are established or well supported by documentation or measurements. Other input parameters that may have distributions will be represented by bounding distributions or bounding values; the purpose of this is so that the calculated result will be a found on the true result. The calculated bound will then be compared to the limiting value set in the design objectives for the issue.

The system model will be constructed from simplified submodels of the processes affecting waste package life and performance. These submodels will be derived from studies performed under this investigation as well as those

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satisfying needs under Issues 1.4 (Section 8.3.5.9) and 1.10 (Section 8.3.4.2). They will be derived through sensitivity analysis of the processes modeled, and, therefore, will be composed of relationships incorporating the most sensitive parameters. Each submodel will be subjected to verification and validation exercises.

Parameters

The system model will combine the submodels to calculate waste package integrity as a function of time, before and up to containment failure. After failure, release rates will be determined for each radionuclide. This model will produce deterministic predictions of radionuclide release for a set of parameters describing a given scenario. The submodels that make up the system model will include the following models (refer to the listed information needs and sections for parameters of the models):

1. Waste package geometry and thermal/mechanical properties.

   a. Waste package geometry

      i. Waste package and borehole configuration and dimensions (Information Needs 1.10.2 and 1.10.3 in Sections 8.3.4.2.2 and 8.3.4.2.3, respectively)

      ii. Waste package contents (materials, mass, elemental and isotopic composition) (Information Needs 1.10.2 and 1.5.1 in Sections 8.3.4.2.2 and 8.3.5.10.1, respectively)

      iii. Changes to waste package geometry over time.

   b. Radiation (Information Needs 1.5.1 (Section 8.3.5.10.1) and 1.10.2 (Section 8.3.4.2.2) and calculations as part of this activity.

      i. Gamma ray source.
      ii. Gamma ray attenuation.
      iii. Heat source from radioactive decay.

   c. Heat transfer (thermal) (Information Needs 1.10.2, 1.10.4, and 1.5.1 in Sections 8.3.4.2.2, 8.3.4.2.4, and 8.3.5.10.1, respectively) and calculations as part of this subactivity.

      i. Heat transfer from waste forms to host rock, temperature field effects.

   d. Mechanical (Information Needs 1.10.2, 1.10.4, and 1.5.1 in Sections 8.3.4.2.2, 8.3.4.2.4, and 8.3.5.10.1, respectively) and calculations as part of this subactivity.

      i. Loads (external, internal, thermal).
      ii. Yielding (ductile and brittle failure).

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2. Container degradation (corrosion) (Information Needs 1.4.2 and 1.4.3 in Sections 8.3.5.9.2 and 8.3.5.9.3, respectively).

   a. Corrosion modes in aqueous conditions.
   b. Corrosion modes in unsaturated conditions.

3. Water package environment (ground water movement and chemistry) (Information Need 1.10.4 in Section 8.3.4.2.4).

   a. Flow surrounding the engineered barrier system.

   b. Flow mechanisms for water contacting the waste package.

   c. Flow mechanisms for water within the waste package after loss of containment.

   d. Water volume available for contact with waste package and waste form.

   e. Transport in near-field host rock.

   f. Temperature in host rock at borehole wall.

4. Radionuclide release from waste forms (this activity).

   a. Spent fuel waste form.
   b. Glass waste form.

Description

The system model will be constructed in a computational efficient manner so that a set of scenarios and conditions sufficiently large to span the range of anticipated and unanticipated events can be considered. After formulation and initial testing of the system model is complete, verification and validation exercises will be performed on the system model as a whole. Verification exercises will concentrate on the numerical accuracy of the logic linking together the system model components. Validation of the system model, in the sense of comparing system model output to experiments that represent an integration of the processes expected to be active in the repository, will not be possible because of the long time scales required. Therefore, validation of the system model will be performed ultimately by peer review.

8.3.5.10.3.5.2 Subactivity 1.5.3.5.2: Development of uncertainty methodology

Objectives

Because of heterogeneities in the environment and in components of the waste package, deterministic calculations of performance alone may not suffice to provide resolution of Issue 1.4 and this issue and to provide the reasonable assurance standard required by the NRC. Therefore, a method for analyzing waste package performance that addresses these uncertainties must

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be developed. The objective of this study is to develop such a method so that the performance assessment calculations for the waste package will provide probability distributions for individual package and ensemble performance parameters, incorporating these uncertainties in conditions and package parameters.

With appropriate structuring of the questions and input distribution values, the uncertainty methodology can be used to answer these types of questions: (1) how does the performance of the different individual waste packages roll up to form the performance of the set of waste packages, to be compared to regulatory requirements and (2) what are the probability distributions of the performance measures of the set of waste packages. The answer to the latter question can provide a part of the support for a reasonable assurance that the performance requirements will be satisfied. A third type of question is what are the probability distributions for the release rates over time of radionuclides from the waste package. This last answer will be provided to Issue 1.1 to help resolve that issue of the EPA limit on cumulative releases to the accessible environment. That EPA requirement is stated in explicitly probabilistic terms.

Parameters

The input parameters are the same as those developed under Subactivity 1.5.3.1.1 (Section 8.3.5.10.3.1.1). The output parameters are cumulative distribution functions for performance measures and for radionuclide release from the engineered barrier system as a function of time.

Description

The uncertainty methodology will use the deterministic system model as a means to predict performance from a given parameter set. Through appropriate sampling procedures, parameter sets will be assembled that represent the anticipated and unanticipated events, as well as variations in waste package components. Examples of this overall approach include the Latin Hypercube and Monte Carlo methods. By repeatedly computing performance with the system model for the sample of inputs, representative probability distributions for release performance may be computed. Because the number of sensitive parameters affecting waste package performance is not expected to be small, the number of performance calculations using the system model is expected to be very large, perhaps several thousand simulations. Therefore, the derivation of a practical but representative sampling method is central to prediction of release distributions.

The uncertainty methodology will be part of the waste package performance assessment model. Therefore, verification and validation of the methodology will be required. After development, verification exercises will be conducted to ensure the mathematical accuracy of the methods. Validation of the model will be accomplished by validation of the system model and through other means as available. As for the system model, the final validation of the uncertainty model will be performed through peer review.

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8.3.5.10.3.5.3 Subactivity 1.5.3.5.3: Water flow into and out of a breached container

Objectives

Although the capillary barrier of the unsaturated zone will normally prevent liquid water from contacting the waste container, under some conditions water flow in the unsaturated zone can result in several mechanisms for water contact with the container. The most likely mechanism for expected flow rates is by wicking from the partially saturated rock where the container is in direct contact with the rock. A second possible mechanism, which would operate at higher flow rates, would be by dripping of water from a fracture onto the container. The objective of this activity is to determine what fraction of the water that drips onto a container would enter the container through a breach in the container wall.

Parameters

Input parameters for this subactivity include the following:

1. Water drip rate.
2. Water temperature.
3. Container orientation.
4. Container breach location.
5. Container breach geometry.

The output parameter will be water flow rates into and out of a failed container and quantity of water that can accumulate in a failed container.

Description

Experiments will be conducted under this subactivity to determine the effect of each of the parameters on the fraction of water that drips onto a container but does not enter a breach in the container. The initial experiments will be conducted using small metal cylinders that contain a well-characterized defect that has been intentionally induced into the cylinder. The fraction of water that enters through the breach will be determined as a function of breach size, breach location, drip rate, and orientation of the container (and breach) relative to the water source. Results of these experiments will be modeled to predict how the results should scale with size. On the basis of the results of the model calculations, some larger scale tests will be designed and executed. The effect of water temperature will be studied in a separate series of experiments using one or two configurations selected to be most probable for the repository situation.

The information developed in the study will be used in the determination of the concentration of radionuclides in solution in failed containers, dilution factors, and release rates of radionuclides from the engineered barrier system for anticipated and unanticipated processes and events (Information Need 1.5.4, Section 8.3.5.10.4). These results in the form of distributions of releases will form a part of the source term for calculation for total system performance assessment conducted to satisfy Information Needs 1.1.5 and 1.1.6 (Sections 8.3.5.13.5 and 8.3.5.13.6).

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8.3.5.10.4 Information Need 1.5.4: Determination of the release rates of radionuclides from the waste package and engineered barrier system for anticipated and unanticipated events

Technical basis for addressing the information need

Link to the technical data chapters and applicable support documents

The bases for the models required to perform these calculations have been discussed in Section 7.4.5. The studies that develop data, parameters, and models necessary to perform the calculations are described in Sections 8.3.4, 8.3.5.9, and 8.3.5.10.

Parameters

Parameters needed for the calculation of waste package releases include

1. Waste package design (Information Need 1.10.2; Section 8.3.4.2.2).

2. Waste package design features affecting radionuclide release (Information Need 1.5.1).

3. Waste package system model and uncertainty methodology (Information Need 1.5.3).

4. Release scenarios for anticipated and unanticipated events (Information Need 1.5.3).

5. Performance of the waste forms under the scenarios in item 4 (Information Need 1.5.2).

6. Probability distributions for system model inputs (Information Need 1.5.3).

Logic

After the scenarios for release resulting from anticipated and unanticipated events have been assembled and the models for predicting release have been developed, verified, and validated, releases from the engineered barrier system may be calculated. These releases will be calculated for each radionuclide using both deterministic and uncertainty models. These releases will form the source term to be provided to the analyses for the total system performance assessment. Further, during the earlier waste package design phases, these integrated performance calculations will provide input to later waste package designs.

Two activities will be performed under this information need. The activities will respectively exercise the deterministic system model and the uncertainty methodology developed for waste package performance assessment in Section 8.3.5.10.3.

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8.3.5.10.4.1 Activity 1.5.4.1: Deterministic calculation of releases from the waste package

Objectives

The objective of this activity is to use the waste package system model developed in Activity 1.5.3.5 (Section 8.3.5.10.3.5) to predict waste package performance using the scenarios developed in Activity 1.5.3.1 (Section 8.3.5.10.3.1).

Parameters

The input parameters needed for this activity are given in the combined list in the technical basis section for this information need. The output parameters obtained are the predicted waste package release performance under specified scenarios.

Description

The calculations of waste package performance will be made in three phases: (1) for the design concepts discussed in Section 7.3, (2) for the advanced conceptual design, and (3) for the license application design. The later phases will use modeling concepts developed in the previous phases, and therefore are difficult to discuss at this point. However, it is likely that analyses in all phases will incorporate many of the same elements.

The analysis of waste package designs will proceed by assembling sets of model input parameters developed in Information Need 1.5.3 and executing the system model to obtain predictions of waste package release. Releases will be calculated for scenarios that represent both anticipated and unanticipated events. Some of the calculations will represent bounding performance calculations, but the bulk of the analyses will support the uncertainty analysis required for probabilistic calculation of releases. In addition, in the earlier phases of waste package designs, information developed in the system model calculations will be available as input to later design phases.

8.3.5.10.4.2 Activity 1.5.4.2: Probabilistic calculation of releases from the waste package

Objectives

Because of heterogeneities in both the environment and components of the waste package, deterministic calculations of performance alone may not be sufficient to provide the performance measure for the set of waste packages for this issue and to support the reasonable assurance standard required by the NRC. The objective of this activity is to provide a probabilistic analysis of waste package performance addressing these uncertainties and the probability distribution of release rates for use in Issue 1.1 (Section 8.3.5.13), using the uncertainty model developed in Activity 1.5.3.5 (Section 8.3.5.10.3.5).

Parameters

The input parameters needed for this activity are given in the combined list in the technical basis section for this information need. The output parameters obtained are cumulative distribution functions for radionuclide release rates from the engineered barrier system as a function of time, and for the maxima over time of the annual release rates.

Description

The uncertainty methodologies developed in Activity 1.5.3.5 will be employed using the system model to assess the reliability of waste package release predictions. This task will be accomplished in concert with the phases of system model development and application. The exact procedure to be followed in these analyses is partially the subject of activities under Activity 1.5.3.5. However, the most likely approach will be to exercise the system model for a range of model inputs selected by a procedure for sampling from distributions of input random variables. From the system model simulations it will be possible to construct the probability distributions for engineered barrier system releases required by the reasonable assurance standard.

The uncertainty calculations will be performed for each of the design phases although they are only required for the license application design analysis. This procedure will allow testing on the early design phases and modification of the methodology during later phases. At least two types of uncertainty will be addressed. First, the uncertainty in the predicted release rates as a result of uncertainties in the fabrication and environment of the waste package will be calculated. Then the secondary uncertainty, that is the confidence in the best estimate of complementary cumulative distribution function for releases, will be assessed. Together with the deterministic simulations for bounding case releases, these results will provide a source term for total system performance assessment and will support the reasonable assurance standard set by the NRC.

## 8.3.5.10.5 Information Need 1.5.5: Determination of the amount of radionuclides leaving the near-field environment of the waste package

Technical basis for addressing the information need

Link to the technical data chapters and applicable support documents

Section 7.4.1 discusses the fluid flow model to be developed in Information Need 1.10.4. The model validation efforts are also discussed in Section 7.4.1 and collected in studies under Information Need 1.10.4. Data acquisition for radionuclide transport properties was begun in FY 1986. No published results were available at the time of the writing of Chapter 7.

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Parameters

The data needed for determination of the amount of radionuclides leaving the near-field environment of the waste package are

1. Scenarios for release events (Information Need 1.5.3).

2. Hydrologic parameters for host rock (Investigation 8.3.1.2.2) and Information Need 1.10.4 (Section 8.3.4.2.4)).

3. Waste package environment tests (Information Need 1.10.4, Section 8.3.4.2.4).

4. Near-field flow and transport model (Information Need 1.10.4).

5. Radionuclide release predictions (Information Need 1.5.4, Section 8.3.5.10.4).

The output parameters are transport properties of radionuclides and radionuclide concentrations in the near-field environment.

Logic

The purpose of this investigation is to provide a source term for use in the 100,000-yr described in 10 CFR 960.3-1-5.

Several processes may act locally to retard the movement of waste in the first few meters of host rock after the radioactive material is released from the engineered barrier system. These processes include sorption mechanisms and, under some conditions, matrix diffusion. Depending on the scenario for transport from the waste package, either or both of these processes may be effective for many waste species.

The source term derived from the release calculations performed for Information Need 1.5.4 will not account for sorption and matrix diffusion effects occurring in the first few meters of rock surrounding the waste package, without modification of the waste package environment component to account for these effects. Flow and transport calculations will be made as described in Information Need 1.10.4 (Section 8.3.4.2.4) to include the hydrologic and geochemical environment immediately surrounding the waste package. These calculations are required to understand the boundary conditions required for the reliability analysis of the waste package. These calculations will not, however, include the transport of radionuclides through the near-field rock following release from the waste package.

To perform these calculations, parameters describing transport mechanisms active in the near-field environment must be determined. These parameters include the formation and transport properties of radionuclide-containing colloids, radionuclide solubilities in repository ground water, diffusivities of waste species, and effective partition coefficients for waste species in Topopah Spring tuff. The colloid and solubility data will be developed in activities described in Information Need 1.5.3 and Investigation 1.14.5 (Section 8.3.1.3.5). Laboratory measurements of apparent diffusion coefficients and distribution coefficients for radionuclides will

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be done using rock wafers and rock cores. The wafers are rock samples that have been part of waste form dissolution tests discussed in the following section. The effect of transport scale on transport processes will be studied by using different size rock cores.

### 8.3.5.10.5.1 Activity 1.5.5.1: Determine radionuclide transport parameters

This activity will measure the distribution of actinides and fission products in rock samples. The rock samples will be subjected to contact with radionuclides under a variety of conditions so that the effects of degree of saturation and transport scale can be evaluated. Two subactivities support this analysis.

#### 8.3.5.10.5.1.1 Subactivity 1.5.5.1.1: Radionuclide distribution in tuff wafers

Many of the waste-form dissolution tests include pieces of the Topopah Spring tuff in the test solution. These tuff pieces are included to determine the effect of the presence of rock on the dissolution rate of the waste form. The rock sample is, therefore, sitting in a solution of dissolved waste form and simulates the condition where local saturation of a portion of the repository occurs. For long-term tests, the solution concentrations are relatively constant, and the test conditions approximate those of steady-state diffusive flow. At the conclusion of the test, the rock wafer is examined with an ion microscope to determine the location of the radionuclides as a function of depth in the sample. A brief description of the method and some preliminary results are in the paper by Finny et al. (1986). The position of the radionuclides in the wafer can be combined with the test duration to calculate the effective diffusion rate for the radionuclide. The concentration of the radionuclide at the surface of the rock can be combined with the solution concentration to give an effective distribution coefficient. The shape of the diffusion profile near the surface of the rock can be used to assess whether sorption or precipitation are controlling the retardation process.

#### 8.3.5.10.5.1.2 Subactivity 1.5.5.1.2: Radionuclide distribution in tuff cores

The tuff-wafer experiments just discussed only examine transport on a small scale (a few micrometers) and under saturated conditions. To determine the effects of transport scale on transport properties, tuff core samples are being used. Solutions of radionuclides are forced through the core sample at a fixed water-flow rate. The water flow is monitored using tritium. Solutions of other radionuclides are compared with the flow rate for tritium to determine retardation parameters. Plans to evaluate retardation properties as a function of water flow rate and solution composition are in the development stage.

Transport properties may also be a function of degree of saturation of the rock. To investigate this possibility, the tuff core experiments will be modified to allow for unsaturated flow. In these experiments the effects of flow rate and degree of saturation will be studied. The goal is to develop a sufficient understanding of the unsaturated flow properties to allow coupling of these experiments with the unsaturated flow studies described in Information Need 1.10.4 (Section 8.3.4.2.4). This will allow monitoring of water movement with resistivity imaging techniques (the impedance camera), tracking of solute transport by the radioactivity (for gamma emitters), and finally, location of the radionuclide distribution in the rock using the ion microscope.

## 8.3.5.10.5.2 Activity 1.5.5.2: Radionuclide transport modeling in the near-field waste package environment

This activity will use the flow and transport model for hydrologic representation of the near-field host rock developed under Information Need 1.10.4 (Section 8.3.4.2.4). The model will be validated using data from integrated testing activities and tracer tests planned in the exploratory shaft.

### 8.3.5.10.5.2.1 Subactivity 1.5.5.2.1: Validation of near-field transport model using laboratory and field experimental data

The hydrothermal flow and transport model developed for detailed analysis of the near-field waste package environment will require validation before it is used to determine releases from the near field. This subactivity will provide model validation by comparison to hydrothermal tracer experiments performed on laboratory core samples, and in situ tracer experiments currently planned for the exploratory shaft (Section 8.3.4.2). In performing these comparisons, split sample techniques will be used to provide for both model calibration and validation.

### 8.3.5.10.5.2.2 Subactivity 1.5.5.2.2: Application of near-field transport model to waste package releases

After model validation, the near-field flow and transport model will be applied to simulate transport of radionuclides through the first few meters of rock surrounding the waste package. Predictions of package release from the waste package system model will be used as the source term. Particular attention will be given to the effects of sorption processes and diffusion of material into the matrix. The interaction of flow in the rock matrix and flow in fractures is expected to be an important factor in assessing potential transport paths for radionuclides. The degree of importance of fracture flow will be coupled to the scenarios examined in Section 8.3.5.10.4. Final results of these calculations will determine the most likely source term for total system performance calculations.

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## 8.3.5.11 Plans for assessing seal system performance

The seal system is composed of the shafts and ramps, exploratory boreholes and their seals, and the sealing components associated with the underground facility. The portions of 10 CFR Part 60 that are related to the shafts, boreholes, and their seals are Sections 60.134(a) and (b). Section 60.113 relates a performance criterion on the engineered barrier system. Because the engineered barrier system comprises the waste package and the underground facility, the seals in the underground facility are indirectly affected by this performance criterion. The approach to establishing the performance goal for the Yucca Mountain Project seal program is described in Section 8.3.3.2 of this document. The approach and plans for assessing the performance of seal designs is described in the following.

Figure 8.3.5.11-1 illustrates the overall logic currently being used to arrive at seal designs that can achieve a desired level of performance. This figure also correlates the six steps in the Yucca Mountain Project repository seal program with these design- and performance-related efforts. These steps are defined in the following paragraphs and are presented in more detail elsewhere (Fernandez, 1985) and in Section 8.3.3.

The first four steps are performed during advanced conceptual design and are as follows:

1. Assess the need for sealing.

2. Define the performance goals and design requirements.

3. Measure material properties.

4. Define sealing designs, assess the performance of these designs, and select the suitable sealing designs.

The last two steps are performed during the license application design (LAD):

5. Perform laboratory analyses and field testing, if required.

6. Reassess the performance of sealing designs (including reallocation of performance, if needed) and select the suitable design.

The focus of this section is to briefly summarize plans for assessing the performance of sealing components. Section 8.3.3 provides more explanation of planned seal performance analyses and information on (1) type of seal, (2) function of seal, (3) location of seal, (4) physical process by which the seal functions, (5) material properties key to seal performance, (6) performance measures for the seal, and (7) goals and desired confidence. The remainder of this section summarizes the broad outline of the overall strategy and the plans for assessing seal performance. A preliminary evaluation of the performance of sealing components will be made as part of step 2. In this step, design options will be rank ordered considering the relative performance of the evaluated design options and the ease in meeting the goal. The logic to be used in this step is presented in Figure 8.3.5.11-2. A

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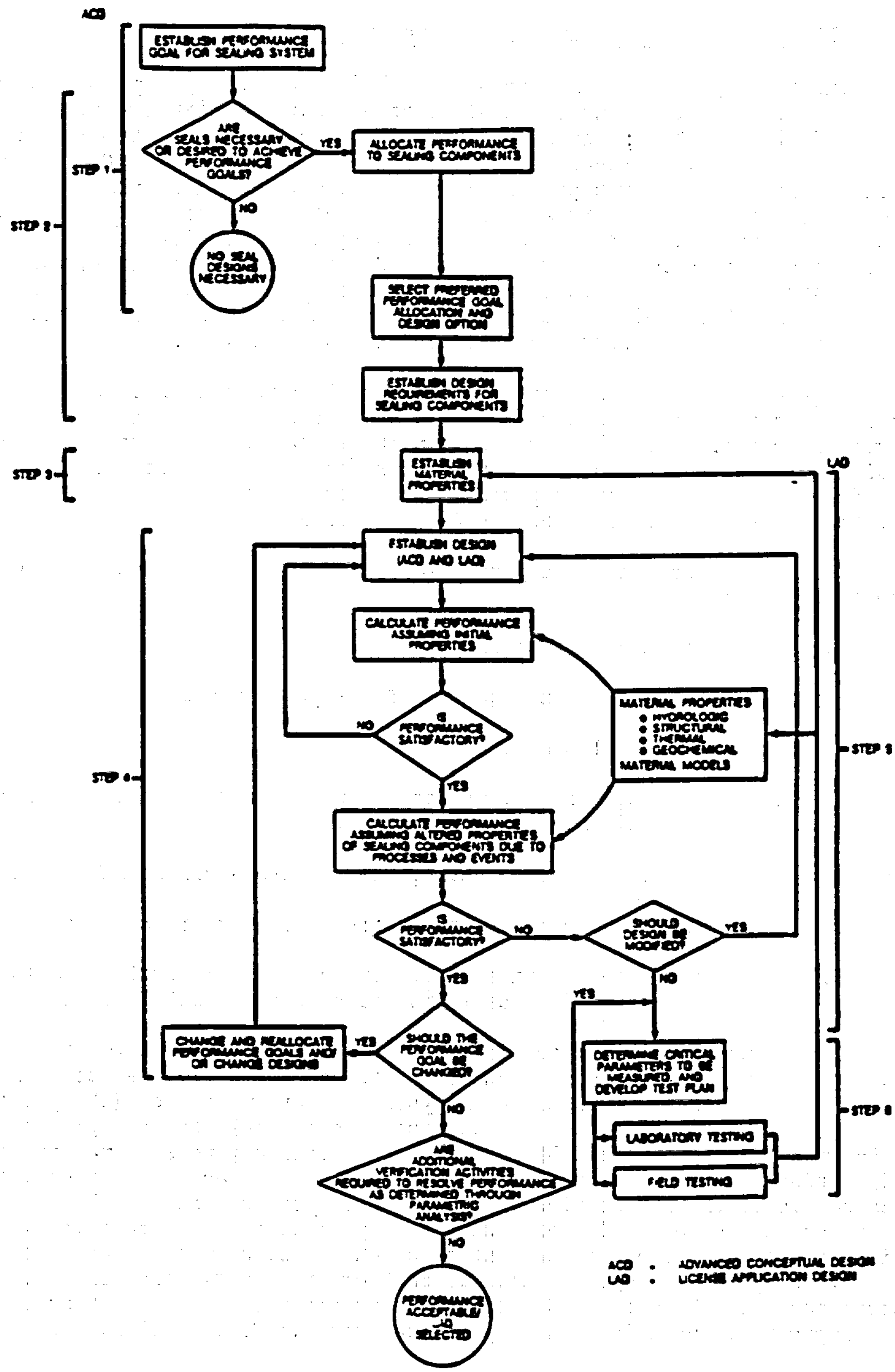


**Figure 8.3.5.11-1.** Flowchart illustrating the approach for answering the performance-related questions (modifie from Fernandez, 1985).

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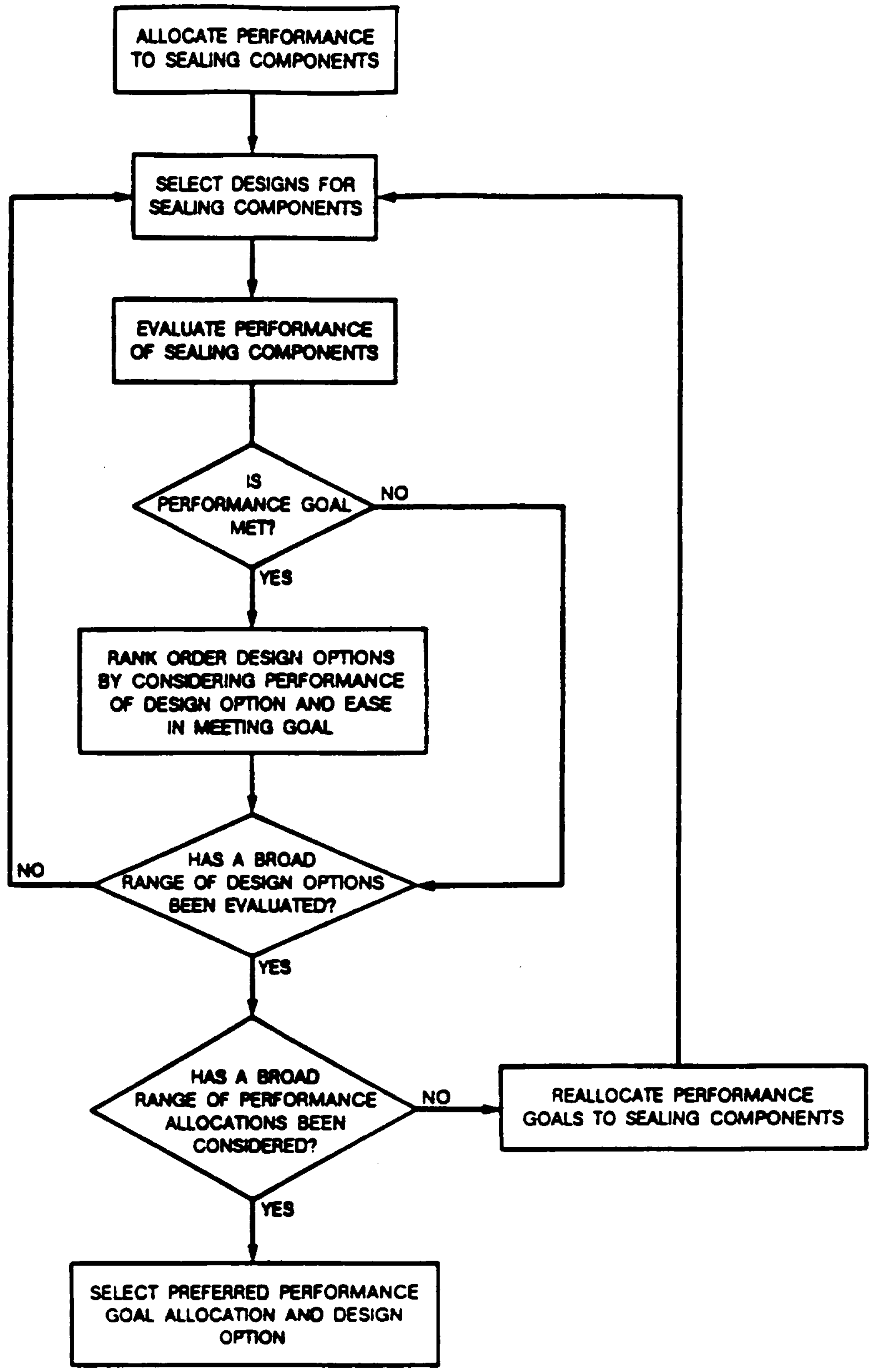


**Figure 8.3.5.11-2.** Logic to select preferred performance goals for seal program.

detailed evaluation of the performance of sealing components will be made while developing the LAD. This evaluation will consider initial and altered seal material properties. To establish these altered properties, it will be necessary to evaluate how the initial material properties are altered by processes and events.

The effect of designs on performance will also be evaluated. Following the reassessment of performance, it may be determined that reallocation of the performance goals is desired. For instance, if the performance goal is only marginally achieved, it may be prudent to reallocate performance or change the design that was evaluated. If the performance goals can be achieved, then the designs evaluated will be proposed as part of the LAD. Additional verification activities may be required if, through the parametric analyses, it can be shown that a higher degree of confidence is required to achieve the performance goal. This higher degree of confidence can be achieved through laboratory, field testing, or both. A reassessment of the designs will be made using the data obtained through verification testing.

The strategy used in the sealing program to evaluate performance of sealing components is to use analytical solutions in a sensitivity analysis and, when appropriate a combination of numerical and analytical models. The numerical and analytical approaches used to date are in Section 6.4.3.1. It is anticipated that no new fluid-flow codes will be required specifically for use in the seal program. Rather, codes that are being developed, verified, and validated for use in other hydraulic performance analyses needed for the Yucca Mountain Project will be used for sealing analyses. Input to verification and benchmark problems will be made by the seal program to ensure the applicability of the codes to the seal environment.

The following subtasks will be performed in evaluating the performance of the sealing system:

1. Develop the following matrices for the materials and designs specified in the advanced conceptual design report:

    a. Events versus processes. (Which likely natural events initiate or enhance processes affecting seal system performance?)

    b. Processes (static, dynamic, and man-induced) versus failure mechanisms. (Which processes contribute to specific failure mechanisms?)

    c. Failure mechanisms versus potential materials and designs. (Which materials or designs will resist, partially resist, or will not resist specific failure mechanisms?)

2. Use the results from the following laboratory and field testing to develop models for use in assessing seal performance:

    a. Laboratory tests to evaluate the following:

        (1) Alteration of sealing materials (cementitious materials) in contact with tuff.

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(2) Consolidation behavior of mined tuff as a function of particle size distribution (this mined tuff will be obtained from the excavation of the exploratory shaft).

(3) Early curing behavior of cementitious materials.

(4) Durability of concrete emplaced on the ground surface.

b. Field tests of the Topopah Spring Member and the tuffaceous beds of Calico Hills to evaluate the hydrologic behavior of these units (described in Section 8.3.1.2, geohydrology program).

3. Use the matrices developed in part (1) together with potential scenarios to predict the alteration of seal performance.

4. Calculate the effect of the postclosure sealing system on radionuclide release. This effort will be coordinated with the total system performance assessments (Section 8.3.5.13) for a repository at Yucca Mountain.